U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-322/83-33

Docket No. 50-322

License No. CPPR-95

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: September 13 - November 4, 1983

Inspectors: Gene Kelly — November 17, 1984
E. M. Kelly, Project Engineer — date

Approved by: Edward G Greenman for — 11-20-84
J. Strosnider, Chief, Projects Section 1C — date

Summary:

Special inspection by a region-based project engineer (106 hours) of twenty allegations related to the design, construction and testing of the Shoreham Nuclear Power Station. The allegations were part of a number of concerns outlined in letters addressed to NRC Region I during the period January-April, 1983, as clarified in an interview with the alleger at the Region I office on July 13, 1983.

Included in the allegations were the following: (1) receipt of a contaminated tool onsite; (2) elimination of the suppression pool steel liner; (3) various structural overload conditions involving the spent fuel pool, Reactor Building columns, and a PASF shield wall; (4) maintenance/surveillance inaccessibility in the drywell and steam tunnel; (5) hydrogen recombiner problems related to spare parts, controls and operation; (6) an inefficient design change ("verbal authorization") process; and (7) miscellaneous valve and piping installation discrepancies, including the use of fiberglass pipe in certain non-safety related applications.

Some allegations were found to be partially accurate descriptions of situations or events which did occur, such as spare parts difficulties for the recombiners and the radioactive tool incident. However, none of these allegations were substantiated as violations of NRC requirements, nor did they result in public health and safety problems.

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TABLE OF CONTENTS

A. Background and Summary

While many of the instances described were accurate, in part, none were a violation of NRC regulations nor did they represent a public health and safety problem. Each allegation is paraphrased from the March 9th document and the July 1983 interview, along with a listing of pertinent references, the scope of the inspection, and the corresponding findings and conclusion.

A number of allegations concerning the Shoreham Nuclear Power Station were presented in a document dated March 9, 1983, addressed "To Whom It May Concern," which contained notes from a taped discussion with a steam fitter. This document was received by NRC Region I (addressed to E. Greenman, Chief, Projects Branch No. 1) in an April 21, 1983 letter. The assertions were made by a steam fitter, formerly employed by a construction contractor at the Shoreham site, and interviewed at the NRC Region I office on July 13, 1983. Based upon the March 9, 1983 document, a transcript of the July 13, 1983 interview, and subsequent phone conversations between Region I personnel and the alleger, a list was compiled of individual allegations to be inspected.

This final special inspection was conducted to determine the validity of twenty of those allegations. The allegation numbers used in Part D of this report refer to the numbers originally assigned in the March 9, 1983 letter, and have been retained for convenience in tracking and inspection resolution. Also, the description of each alleged condition in Part D is actually a paraphrase obtained from: (1) the original March 9th document; (2) the July 13th interview, and, (3) inspection of the item. In fewer than half of alleged conditions was there enough specificity provided by the alleger to allow an immediate inspection. Most of the items involved second or third-hand information, and many of the details necessary to adequately address these allegations (such as specific equipment, people or location) were disclosed only as a product of the subsequent NRC inspection.

Part D of this report, beginning on page 6, presents the detailed findings and conclusions which support the fact that none of the allegations addressed were found to be substantiated.

B. Personnel Contacted

Long Island Lighting Company (LILCO)

S. Aikens, Technical Support
C. Albertini, Assistant Construction Supervisor
M. Chipken, Technical Support
N. DiMascio, Health Physics Engineer
M. Donegan, Health Physics Supervisor
V. Esposito, Assistant Construction Supervisor
J. Etzweiler, Nuclear Engineer
C. Gentile, Radiation Protection Engineer
R. Glazier, Field QA Engineer
J. Hall, Special Projects
T. Horner, Test Engineer
W. Hunt, Construction Manager
R. Jongebloed, Nuclear Engineer
J. Kelly, Field QA Manager
G. Laurie, Projects Office
R. Lawrence, Startup Engineer
J. Livingston, Senior Test Engineer
M. Miele, Radiation Protection Supervisor
W. Museler, Director, Office of Nuclear
E. Nicholas, Field QA Section Supervisor
R. Purcell, Assistant Startup Manager
J. Scalice, Operators Manager
J. Smith, Manager, Nuclear Support
D. Terry, Chief Maintenance Engineer
E. Tesko, Construction Supervisor
A. Todoro, QA Inspector
J. Whittaker, Startup Engineer
E. Wlock, Nuclear Procurement
E. Youngling, Nuclear Engineering Manager

Stone Webster Engineering Corporation (S&W)

P. Baker, Lead Structural Engineer, SEO
R. Chung, Maintainability Group
J. D'Amato, Power Engineer
A. Dobrzenieck', Startup Engineer
D. Dyroff, FQC Assistant Engineer
R. Jaquinto, Head, Site Engineering Office, SEO
J. Kammeyer, Assistant Head, SEO
R. Muxo, Pipe Support Engineer
C.F. Ng, Pipe Support Engineer
W. Shosho, Principal Electrical Engineer
R. Sperling, Pipe Support Engineer
R. Wiesel, Lead Structural Engineer, EMD (Boston)

General Electric Company (GE)

* C. Doyle, Manager, Customer Services, Fuel Manufacturing - Wilmington, NC
A. Ketchum, Test Engineer
* E. Lees, Manager, QA, Fuel Manufacturing - Wilmington, NC
K. Nicholas, Lead Startup Engineer
J. Riley, Operations Manager

* October 19, 1983 phone call

Dravo (October 21, 1983 phone call)

E. Reno, QA Manager
L. Harmon, QA Supervisor
T. Whitacre, QA Supervisor

Courter & Company

J. Arcuri, Project Manager
A. Musumeci, Supervisor
E. Stoudt, QC Staff Engineer

Rockwell International (Raleigh, NC) (November 1, 2 and 7, 1983 phone calls)

C. Hinnant, Radiation Safety Officer
R. Bandukwala, QA Manager
J. Kertis, Customer Service Superintendent

Rockwell Corporation, Atomics International Division Energy Systems Group (Canuga Park, California - November 3, 1983 phone call)

R. Brengle, Project Engineer - Recombiner Qualification
R. Cardenas, Recombiner Project Manager
D. Empey, Quality Assurance Director
A. Itow, Project Engineer
C. Knox, Field Service Representative
K. Sanders, Program Manager, Special Nuclear Products
S. Sarnecki, Spare Parts Manager
F. Williams, Project Manager

NUS Corporation

F. Burnhard, Spare Parts Consultant

TRW Mission Representatives (October 17, 1983 phone call)

J. McTaggart, Sales Engineer, Miller Energy Co.

U.S. Nuclear Regulatory Commission

M. Hum, Materials Engineering Branch, NRR
J. Higgins, Former Senior Resident Inspector, Shoreham
R. Hartfield, Chief, Caseload Forecast Panel
K. Manoly, Reactor Inspector, Region I
R. Nimitz, Senior Radiation Specialist
C. Petrone, Resident Inspector, Shoreham
A. Grella, Senior Transportation Specialist

The individuals listed above represent only principals contacted; the inspector also held discussions with other licensee and contractor personnel during the course of this inspection.

C. Referenced Documents

The following documents were used extensively for Shoreham design and docketed information:

- NRC Region I Inspection Reports for Shoreham
- Shoreham Systematic Assessment of Licensee Performance (SALP) Reports
- Shoreham Final Safety Analysis Report (FSAR)
- NUREG-0420, Shoreham Safety Evaluation Report (SER)
- Shoreham Plant Technical Specifications (TS) - Proof and Review Copy

D. Inspection Details

This section addresses the 20 allegations (referred to by the numbers originally assigned in the alleger's March 9, 1983 document) which were inspected. Each subsection is comprised of a paraphrase of the allegation, with the corresponding inspection scope, references, findings and conclusion.

1. Allegation No. 1 - Hydrogen Recombiner Problems

The hydrogen recombiners were alleged to: (a) require spare parts which are unavailable; (b) use control components of poor quality; and (c) have experienced "running" problems during preoperational testing.

1.1 Scope

The inspector interviewed startup test engineers responsible for preoperational testing of the recombiners, and reviewed the two preoperational test procedures which document testing completed on these units. Also reviewed were five pertinent NRC inspection reports which previously addressed the testing of the recombiners, including an outstanding open inspection item related to spare parts which has since been resolved.

Organizations onsite which were responsible for spare parts inventory, and LILCo corporate (Hicksville) purchasing personnel were contacted. A conference call was initiated with the manufacturer/supplier of the recombiners, the Rockwell Corporation (Canuga Park, California), on November 3, 1983, to discuss the availability of spare parts and the control systems employed. Also contacted were S&W engineers involved in the environmental qualification of the skid-mounted electrical components and the power cabinets associated with the recombiners.

Finally, generic NRC notification of environmental qualification difficulties was presented in IE Information Notice 83-72, and LILCo's progress in addressing those concerns was reviewed.

1.2 References

-- E&DCR-P-4088, approved July 20, 1982

-- Hydrogen Recombiner Operating and Maintenance Manual, Appendix D, Recommended Spare Parts List

-- FSAR Figure 6.2.5-1, Primary Containment Atmospheric Control System

- Purchase Order 363247-1-SSP, March 30, 1984
- Internal Memorandum dated August 12, 1983, A. Dobrzeniecki to E. Youngling
- IEEE Standard 323-1974, Environmental Qualification
- LILCo Response to TMI Action Item II.E.4.1 of NUREG-0737, Dedicated Hydrogen Penetrations
- IE Information Notice 83-72, Environmental Qualification Testing Experience, October 28, 1983
- NRC Region I Inspection Report Nos. 50-322:
  83-10 issued May 27, 1983, Detail 5.3
  83-32, issued October 27, 1983, Detail 8
  84-04 issued May 1, 1984, Details 2.2.5 and 14
  84-07, issued April 25, 1984, Details 3.3 and 3.5
  84-23, issued July 20, 1984, Detail 2.3
- PT 402.001-1 Revision 1, Approved March 15, 1983; Primary Containment Atmosphere Control System; Test results approved November 15, 1983
- CS 402.001-1 Revision 1, Approved December 13, 1982 Hydrogen Recombiner Checkout for Primary Containment Atmosphere Control, Test results approved August 12, 1983
- Shoreham TS 3/4.6.6, Primary Containment Atmosphere Control

1.3 Findings

The two hydrogen recombiners, manufactured by the Atomics International Division of the Rockwell Corporation were successfully preoperationally tested during the period from February-November 1983. Completed test procedures CS 402.001-1 and PT 402.001-1 documented a total of four test exceptions which were cleared and closed; test results were approved by the licensee for both procedures on August 12 and November 15, 1983, respectively. NRC's independent review of these procedures and test results, as part of Inspection 50-322/84-07 (as well as this inspection), found no record or evidence of recombiner "running" problems. The performance of the Shoreham thermal hydrogen recombiner system, in achieving and maintaining proper recombination temperature and containment atmosphere flow with appropriate alarm and trip signals, was demonstrated by these tests. One explanation of the alleged repeated running of the machine, and its alleged inability to build-up sufficient pressure to "blow itself clean", would have been the piping system flushes performed as part of (and documented in) test procedure CS 402.001-1. These flushes were repeated many times, sometimes for relatively long periods, to demonstrate piping system continuity, as well as cleanliness and an absence of debris. The recombiner blowers, which are fans manufactured by Buffalo Forge that achieve a maximum discharge pressure of 5 psig, were used as a source of low pressure air during the piping flushes.

The availability of qualified spare parts has been a recognized problem for the hydrogen recombiners. The problems can be characterized as a combination of: (1) minor contractual difficulty with the primary manufacturer, Rockwell International Corporation, involving subvendors which either stopped supplying or went out of business; (2) environmental qualification failures of certain control and protective components, which have necessitated replacement (in type); (3) actual performance difficulties with some hardware, principally the thermal overload breakers for the blower motors; and, (4) delays of about 15 months, in processing the spare and/or replacement part orders, by the licensee's own Technical Support and Purchasing organizations. Ultimately, the issue of a lack of spare parts for the recombiners was identified and resolved during NRC inspection of the preoperational test results for these units, as outstanding item 84-07-03. The blower motor thermal overload circuit breakers were replaced and satisfactorily tested; additional adequate spare parts were ordered on March 30, 1984; and, environmentally qualified components have been procured and installed on required portions of the recombiner system, as identified in E&DCRP-4088 and confirmed by NRC inspection.

The Rockwell Corporation is still in business, and capable of supplying qualified replacement parts for the over 100 thermal recombiner units which they have sold to date. The Shoreham recombiners are of a specific design similar to 6 other units built by Rockwell, 2 of which have been in operation in Japan for over 5 years. The Hope Creek Station recombiners are also similar to Shoreham's, the only differences being that all of their electrical components were originally environmentally qualified (whereas Shoreham's involved some retrofit), and their power cabinets are of a newer design. Two of Rockwell's former sub-vendors have since either stopped supplying or gone out of business. Honeywell, the original manufacturer of flow and temperature controllers installed in the recombiner control cabinets, no longer makes the controllers. However, the Honeywell controllers in use at Shoreham have performed satisfactorily and replacement circuit board cards are still available from Rockwell directly, as is a suitable alternate replacement controller design by Athena. Also, Visicon, the original supplier of circuit board cards for the control cabinet annunciators, has gone out of business. Rockwell subsequently bought out the rights to the Visicon annunciator design, and currently supplies these parts which they now manufacture in-house.

There are other parts that comprise the Rockwell recombiner system which were either discontinued by the original supplier, or whose supplier was changed by Rockwell, for a variety of reasons. As explained in NRC IE Information Notice 83-72 ITE-Gould circuit breakers, Timetrol SCR power controllers, and ITT-Barton pressure transmitters were reported to have failed their environmental qualification testing. All of these products were (or are) capable of replacement with qualified components, through the Rockwell Corporation.

E&DCRP-4088, if properly implemented, will effect the component replacements necessary to environmentally qualify the hydrogen recombiners in accordance with IEEE-Std. 323.

Since the Shoreham recombiners were shipped in late 1976-early 1977, the "controls" which were alleged to be of poor quality have aged, such that their design could be described as not "state-of-the-art." At the time referred to by the allegation (June-July 1982), these controls would have been more than 5 years old, and of an original a design which may have been typical of state-of-the-art 10 years agc (today). Since no specific controls were identified by this allegation, it was assumed that the alleger is referring to the Honeywell flow/temperature controllers at the control cabinet located in the main control room. A four-element thumbswitch is used to set in the desired flow control, valve position and heater temperature; analog readout is provided for both "deviation" (actual from set) and "output" (temperature or valve position). While this Honeywell flow/temperature controller arrangement has proven satisfactory, it has also been considered for replacement by a more modern digital system. The quality of these controls, however, has not been a concern or problem for the licensee's test engineers. This is also true of the Analogic digital meter used to display gas temperature and inlet pressure.

1.4 Conclusion

There have been minor problems experienced in the procurement of spare parts for the recombiners. These problems were principally due to environmental qualification of certain electrical components. Also, due to the timing of the original order for these units, the controls are not of a modern, "state-of-the-art", digital design. However, the required component replacements and controls design for the Shoreham recombiner units have neither affected the recombiner system's ability to satisfactorily perform (as demonstrated by preoperational testing), nor are they expected to present an operational problem in the future. Plant Technical Specification 3/4.6.6, when effective, will require regular surveillance testing of the recombiner units and will ensure reliable operation.

2. Allegation No. 2a - Recombiner Check Valve Flapper

A hydrogen recombiner check valve, whose flapper was found to be hung-up, was alleged to be never repaired or replaced. Further, a similar uncorrected problem was alleged to exist for not only the redundant recombiner, but other such valves supplied by the same manufacturer - Velan.

2.1 Scope

The alleged check valve is a 6-inch swing type check manufactured by Velan Engineering Company (Serial No. 218). The valve is installed horizontally in the "A" hydrogen recombiner's discharge line, and is used to prevent back-flow through the recombiner unit. Maintenance records of work on this valve were reviewed, along with flushes and preoperational tests later conducted for the "A" recombiner. The identical valve on redundant recombiner "B" was similarly checked.

NRC Information Notices and Bulletins, which address problems with Velan valves, were compared with programs at Shoreham which identified/trended valve problems. Finally, the hearing record for Contention SC-11 as part of the ASLB proceedings, which addressed passive mechanical valve failure was reviewed for its applicability to swing check valves.

The inspector observed the installed condition of the swing check valves on each recombiner's discharge.

2.2 References

2.2.1 SWEC Dwg. No. FM-52A-12, Rev. 12;
Primary Containment Atmospheric Control System, Line Nos. 6 inch-GR-105/106-151-2, Check Valves VCW-15A-2,
-- T48*06V-004A, Velan Serial No. 218
-- T48*06V-004B, Velan Serial No. 455

2.2.2 Repairs to Velan Check Valve T48*06V-004A:
-- Nonconformance Report (NCR) 1899
-- Courter DCO No. 9167, September 28, 1981
-- Courter Disassembly/Reassembly Release No. A-0454; Requested September 28, 1981; completed March 9, 1982
-- E&DCR-F-39656, March 1, 1982
-- E&DCR-F-39656A, March 4, 1982
-- Velan Engineering Company Field Service Report (by M. Bukowski, Service Representative), March 8, 1982

2.2.3 Repairs to Velan Check Valve T48*06V-004B:
-- Courter Deficiency Correction Order (DCO) No. 9169; issued September 28, 1981, closed February 26, 1982

2.2.4 Recombiner Flushing Operations
- -- Test Procedure CF402.001-1; Primary Containment Atmospheric Control System Flush, July 6, 1982
- -- Repair/Rework Form T48-45A; April 5 - Oct. 11, 1982
- -- Repair/Rework Form T48-47B; March - October, 1982
- -- Startup Flush Report; October 11, 1982
- -- OQA Verification Report; October 14, 1982

2.2.5 Other Velan Check Valve Problems
- -- Courter QAP-12.1, Nonconformances, August 20, 1980; Section 3.7, Trend Analysis
- -- Courter Memorandum (Royce to Schmidt), May 13, 1983; First Quarter 1983 Courter Trend Analysis
- -- LILCo Memorandum (A. Dobrzeniecki to E. Youngling) dated August 12, 1983
- -- LILCo to Velan Letter dated July 16, 1980
- -- SWEC Memorandum (Brabazon to LILCo Project Manager), April 6, 1981
- -- NRC Region I Inspection Report Nos. 50-322:
  80-19, Detail 3; issued January 9, 1981
  81-08, Detail 7; issued June 9, 1981

2.2.6 NRC IE Information Notices
- -- No. 81-30, Velan Swing Check Valves, September 28, 1981
- -- No. 80-41, Failure of Swing Check Valve, November 10, 1980
- -- No. 79-04, Incorrect Weights for Velan Valves

2.2.7 Passive Mechanical Valve Failures
- -- Affidavit of John A. Rigert before Shoreham ASLB; Detection of Passive Check Valve Failures, November 28, 1983
- -- ASLB Partial Initial Decision (LBP-83-57), September 21, 1983; Section II-C, Passive Valve Failure (SC Contention 11)
- -- LILCo Letter (SNRC-859), Smith to Denton, April 15, 1983; IST Program, Rev. 3 - NES Document No. 80A2903
- -- LILCo Letters to NRC Region I (Smith to Murley), dated June 30, 1983 and August 31, 1983
- -- NRC Region I Letter (Starostecki to Pollack), dated April 1, 1983

-- NRC IE Bulletin 83-03, March 10, 1983, Check Valve Failures

2.2.8 Shoreham FSAR Section 6.2.5, Combustible Gas Control in Containment

2.2.9 Shoreham Technical Specification 3/4.6.6, Primary Containment Atmosphere Control

2.3 Findings

The "A" recombiner check valve internals were removed for a piping hydrostatic test, and upon reassembly of the valve in September 1981, the disc ("flapper") was found to interfere with the valve body, such that it hung-up when moved by hand. Courter Disassembly/Reassembly Release Form 0454 initiated this work (the valve was being cleaned for the piping hydro); Nonconformance Report 1899 then dispositioned that rework be performed, in accordance with Engineering & Design Coordination Report (E&DCR) F-39656. This E&DCR authorized the millwright shop to reposition the valve's ring hinge. Ultimately, 0.046 inches were machined off of the original disc diameter, in the presence and under the supervision of a Velan service representative. Liquid penetrant testing revealed satisfactory fusion between the disc's stellite surface lay and cast base, and the valve was reassembled and found to open and close position without noticeable restriction. A Velan Engineering Company Field Service Report dated March 8, 1982, was filed by M. Bukowski, the Velan Field Service Representative. The valve was satisfactorily reassembled on March 9, 1982 and the NCR was closed on March 11, 1982. This valve was later used for recombiner system flushing, as a convenient flush connection point (i.e. further disassembled and reassembled), until the end of flushing operations in June 1982. No further failures of this valve have been experienced to date.

The redundant recombiner's discharge line also has a 6-inch Velan (Serial No. 455) swing check valve installed. This valve has not experienced the same (or any other) operational problems, such as its redundant counterpart. This valve's internals were also removed for cleaning during the September 1981 hydrotesting; however, it was not found to have disc-to-body interference. A Courter Deficiency Correction Order (DCO) was issued on the same day, September 28, 1981, as the other check valve disc interference was noted - this DCO required installation of a new flexitallic gasket, since the old gasket (pre-hydro) should not have been re-used upon reassembly following the hydro. Valve reassembly was witnessed by SQA on February 18, 1982, with no discrepancies noted. This valve was also later disassembled, and used as a flush connection. Successful conduct of recombiner flushing was documented in procedure CF402.001-1, which was approved on July 6, 1982. Proper reassembly was verified by OQA in a report dated October 14, 1982.

Both recombiner systems were preoperationally tested during February-November 1983, with no problems associated with either discharge check valve noted. The 6-inch discharge check valves on each recombiner are currently installed and operable.

Other Velan check valve problems have been identified and documented, by both the licensee and the NRC. None of these problems involved the disc-body interference experienced by the "A" recombiner 6-inch discharge check valve. NRC Information Notice 81-30, identified internal damage found on Velan 6-inch swing check valves involving disc nut lock-wire and hinge pin problems at a number of operating plants. The licensee identified problems with Velan swing check valves in two documented instances: eighteen, 4-inch check valves were found (during hydrotesting) to have a forged-body machining error, and were sent back to Velan on July 16, 1980 for repair. The disc stopper assemblies were reduced in thickness (from 3/8 to 1/4-inch) to correct jamming or wedging open against the back wall of the valve body. This solution was evaluated in E&DCR's, and resolution was documented in an April 6, 1981, memorandum from Stone & Webster to LILCo's Project Engineer. NRC inspection coverage of this problem was documented as part of Reports 80-19 and 81-08.

NRC Bulletin 83-03, described a disc-hinge separation problem, apparently caused by vibration/ corrosion, and questioned the adequacy of forward-flow testing in assuring the integrity of check valve internals. The related issue of passive mechanical valve failure, and the capability of LILCo's Inservice Test (IST) Program to detect and correct this problem, was addressed as part of Contention SC-11 during the NRC Atomic Safety and Licensing Board (ASLB) hearings for Shoreham. The Brenner Board found that the single-failure and redundancy incorporated in Shoreham's safety-related systems, together with the component reliability and low frequency of undetected valve failure (specifically disc-stem separation), are such that there is reasonable assurance that LILCo's IST program is adequate to permit fuel loading and operation up to 5% rated power. However, the sole issue of the adequacy of single-direction testing for certain check valves remained unresolved, and in the ASLB Partial Initial Decision issued on September 21, 1983, the Board sought a statement from LILCo as to the adequacy of its current IST program to detect and prevent such failures. In an Affidavit filed by LILCo on November 28, 1983, the rationale behind the proposed forward-flow IST, to be conducted once every six months for the recombiner discharge check valves, was presented. Forward-flow IST of these valves, performed pursuant to ASME Section XI, was argued to be sufficient to demonstrate their operability, since:

- these valves are infrequently operated, and in a system normally in a standby mode
- the valves are neither in a corrosive or erosive environment
- recombiner system operation will not induce abnormal thermal or vibration stresses
- industry experience shows no history of these check valves failing

Shoreham Technical Specification (TS) 4.6.6.1 will require a hydrogen recombiner system functional test every 6 months, to exercise the recombiners' Velan check valves at full flow when the TS become effective.

2.4 Conclusion

There was a documented problem with the "flapper" of a recombiner check valve. The "hanging-up" of a 6-inch Velan check valve, installed on the "A" hydrogen recombiner, was discovered in September 1981 and was repaired. The redundant "B" recombiner discharge check valve did not experience any disc-to-body interference, nor has there been any generic problem, similar to that interference, with Velan check valves at Shoreham.

Regularly-conducted IST and Technical Specification system surveillance will exercise hydrogen recombiners discharge check valves, and should allow for detection of valve failures such as a hung-up disc.

3. Allegation No. 2b - Grouted Shear Pin Holes

The alleger questioned the integrity of a grout-concrete interface, and the associated shear pin installation, for a large (initially unspecified, except for general location) support whose baseplate was alleged to have had shear pin holes drilled too large. The holes were alleged to have been then grouted, re-drilled (presumably the concrete - not the plate) to the correct size, and the shear pins re-installed and possibly welded to the floor plate. The support was later identified, in a July 18, 1983 phone call with the alleger, as RHR hanger number E11-085, located at Reactor Building elevation 63, on the north face of column 12.

3.1 Scope

The specific hanger identified by the alleger was not a support (or snubber) which employed either baseplate or shear pins, nor was it tied to a Reactor Building column. Hanger E11-085 was part of the RHR system; however, it was not large and was located at a lower-than alleged Reactor Building location. It was, therefore, not of interest.

There were three larger RHR supports which did resemble the location and description alleged, and which employed large baseplates, with numerous shear pins added in the course of extensive redesign of those supports. The inspector interviewed FQC personnel familiar with UNICO inspection of those supports, during installation in early 1980 through May, 1983. Stone and Webster Site Engineering Office structural engineers who processed many of the E&DCR changes related to those supports were also interviewed, as was the S&W chief structural engineer (R. Wiesel) from Boston responsible for Shoreham.

The support packages for each of these hangers were reviewed. These contained the many drawing revisions, deficiency reports, UT examination reports, FQC inspection reports, repair/rework authorizations and E&DCR's which document the historical design evolution of a support. The interviews of cognizant FQC inspectors and S&W/SEO support engineers, and the review of each support's extensive documentation, were performed with attention to the addition of shear pins to baseplate - specifically for oversized holes - which were grouted.

3.2 References

3.2.1 FQC Pipe Support Package for Hanger 1E11-PSSH-085; Bergen-Paterson Vertical Spring Can (in vicinity of Reactor Building Column C-11, between elevations 22-29, supporting 18-inch RHR pipe, from overhead structural steel).

- -- Bergen Paterson Dwg. No 1E11-PSSH085-7

3.2.2 FQC Pipe Support Package for Hanger 1E11-PSR-054; (RHR Restraint, supported in part of West face of Reactor Building Column C-12, at elevation 63.

- -- LILCo Dwg. No. M-12197-13 (Rev. 1, 6/75 to Rev. 4, 7/79)
  SWEC Dwg. No. BZ-8F-13-12, (Sheets 1-6), August 5, 1983.

- -- E&DCR F-23699 Series (changes to support E11-054); Original (9/19/79) thru Rev. K (6/16/83).

- -- FQC Inspection Reports:

  Shear Pin Installation, October 10, 1979
  Final Phase I Rework, December 7, 1981
  Rework (using Devcon), May 18, 1983
  UT Examination Report, February 24, 1980

- -- Repair/Rework Request E11-276, June 7, 1982

- -- E&DCR F-23699K, June 16, 1983, including:
  LILCo Deficiency Report (LDR) 1349, May 17, 1983
  FQC Inspection Report, July 29, 1982

3.2.3 FQC Pipe Support Package for Hanger 1E11-PSA-055; (Restraint for 24-inch vertical RHR pipe, in vicinity of, but not directly tied to, Reactor Building Column C-12, elevation 42)

- -- SWEC Support Drawing BZ-8F-14-10

- -- E&DCR F-42391, August 18, 1982

- -- E&DCR F-9151E, incorporates verbal authorizations dated:

  9/8/82 - maximum shear pin embedded length
  9/14/82 - relocated shear pin locations
  9/21/82 - shear pin driller inaccessibility
  9/29/82 - baseplate add-on pad dimensions

- -- Repair/Rework Request Form, August 31, 1982
- -- FQC Inspection Report, September 8, 1982

3.2.4 FQC Pipe Support Package for Hanger 1E11-PSA-025; (Anchor for RHR piping, off Reactor Building Column C-4, elevation 63)

- -- LILCo Dwg. No. M-12197-10-8
- -- SWEC Support Dwg. BZ-8F-10-8, Sheets 1-10
- -- SWEC Nonconformance & Disposition Report (N&D) No. 5222, October 12, 1982
- -- E&DCR F-9977, Repairing Enlarged Holes in Pipe Support Baseplates (Devcon)
- -- E&DCR F-44550, December 15, 1982
- -- FQC Inspection Report, May 23, 1983

3.2.5 E&DCR F-22227E, May 20, 1980; Procedure for Shear Pin Installation

3.2.6 E&DCR P-2940G, approved September 14, 1979; Installation of Drilled-in-Concrete Anchors, Section 5.0, Installation Tolerances, Subsection 5.1.5, Abandoned Holes

3.2.7 NRC Region I Inspection Report 83-34, issued December 21, 1983; Detail 2.2.3, Allegation No. 25, Verbal Authorizations

3.3 Findings

A Residual Heat Removal (RHR) system pipe support (hanger E11-085), allegedly located on the north face of Reactor Building Column 12 at elevation 63, was alleged to have had shear pins added with holes drilled too big, and then grouted and re-drilled to a smaller size. That alleged support does not exist.

RHR support 1E11-PSSH-085 is actually a vertical spring can (or hanger), located between Reactor Building elevations 22 and 29 in the vicinity of Column C-11, and supported from above, off of an overhead structural steel member. The hanger is manufactured by Bergen Paterson Pipesupport Corporation, and its design does not include either baseplate or shear pins.

Three larger RHR pipe support assemblies, none of which were specified by the alleger, but each of which resemble the qualitative description given by the alleger ("pretty big . . .lots of bolts and shear pins . . . re-designed several times"), including relative location ("as you enter the truck bay . . . in the valve alley above . . . one on each side . . . to your left . . . the biggest thing in town . . . Reactor Building elevation 63"), were inspected for baseplate shear pin modifications and, in particular, grouting of possible under or over-sized shear pin holes. The RHR supports so identified were:

-- Restraint 1E11-PSR-054 at elevation 63 off the West face of Column C-12

-- Restraint 1E11-PSR-055 at elevation 42 in the vicinity of (but not tied directly to) Column C-12

-- Anchor 1E11-PSA-025 at elevation 63 in the vicinity of (but not tied directly to) Column C-4

These supports are three of the five largest supports in the Reactor Building - all were re-designed extensively, and have baseplates containing many bolts and shear pins, including numerous additions or relocations.

3.3.1 Restraint PSR-054

This is a large, complex support with a number of baseplates employing shear pins, and tied directly to Column 12. At least ten or more formal E&DCR design changes, spanning the period September 1979 - July 1983, were documented; many involving shear pin relocation or addition for various reasons. Three baseplates are part of this support; two are located on the floor slab, 1 on nearby Column 12. The west plate, of dimension 44 X 55 inches and located on the floor slab, had one identified instance whereby 1-inch diameter shear pins (as called for on the installation drawing) were found to be, by inspection, 1 1/8-inch (as actually installed). This condition was reviewed by support engineers and found to be acceptable in E&DCR F-23699, Revision F, dated October 29, 1981. Rather than a smaller-than-intended pin size (as alleged), this was just the opposite or a larger-than-intended situation. No mention or instances of grouting were found for this support, from original to current design.

3.3.2 Anchor PSA-055

This support is in the vicinity of, but not directly connected to, Column 12 in the Reactor Building. This is, in the estimation of most cognizant structural engineers, the largest support in the building, even through its at a lower-than-alleged elevation (40 vs 63). It too underwent some redesign, including shear pin additions and relocations (not an uncommon modification for support installation). One group of verbal authorizations, pr3ceeding approval of E&DCRF-9151E in September - October 1982, referred to rebar interferences encountered while drilling shear pin holes, and approved new locations and embedments for these pins.

Another verbal authorization, dated September 8, 1982, specified a maximum embed depth of 5-inches (one inch deeper than the nominal standard depth of 4-inches) for certain shear pins whose holes had been drilled too deep. The field request for the deeper embed was preferable to grouting and re-drilling, and was approved. Another verbal request, associated with one pin which could not be located properly, authorized a 4 3/4-inch maximum embedment. No instances of grouting/redrilling were found.

3.3.3 Anchor PSA-025

While one of the five largest supports in the Reactor Building, this support was not near Column 12, even though it is at elevation 63 near the "valve alley" referred to in the alleger's interview. A large (5 ft. x 5 Ft. x 1½ inch thick) baseplate associated with this support and containing nine shear pins, was found by licensee inspection (N&D No. 5222, 10/12/82) to have 1 1/8-inch diameter pins installed versus the intended 1 1/4 inch design. This was dispositioned "accept-as-is". No mention of grout was made, and its surmised that the added 1/8-inch tolerance (1/16 on either side of pin) was judged to be acceptable.

3.3.4 Shear Pin Installation Procedure

E&DCR F-22227E (5/20/80) specified procedural details for shear pin installation in baseplates: the normal embed depth of 4 inches, total 1/8-inch lateral tolerance, and a locational tolerance of ± 1/2 inch on the plate were general criteria meant to serve all situations commonly expected. It was not unusual to deviate from these values, although formal E&DCR approval was required. The verbal authorization program was tailor-made for these types of deviations; for example, PSR-054 was subject to 19 verbals and PSA-055 had 16 verbals. More than 50% of these involved shear pin installation.

The shear pin is designed to load the concrete, in which its embedded, in bearing - not tension. The bearing is close to the surface (top of concrete - pin hole) and, while some pin deformation is inevitable (from top to bottom), the pin will resist a shear stress (between baseplate and concrete).

The use of grout was an approved standard practice, on a case-by-case basis, at Shoreham. Approved grout material would require a strength which is greater than the bearing capacity of concrete. Thus, while no grouted/re-drilled shear pin installations were found for the three supports inspected, the potential for this alleged condition would not be of engineering concern since the principle function of the shear pin would not be affected. The shear pin would simply load the surrounding grout in bearing, the way its designed. The grout-concrete interface would be equivalent to (if not better than) the standard hole.

All three of the large supports were inspected for installation and dimensional accuracy. Shear pins were generally difficult to directly observe, since they are covered with a pipe cap which is welded to the baseplate, in most cases.

3.4 Conclusion

RHR support 085 had no baseplate or shear pins, and was not located as alleged. This was confirmed by a visual inspection.

The three most obvious supports, which fit the allegation were found to be: 1) large, 2) re-designed a number of times, 3) full of shear pins, and 4) in the generally alleged location. However, no evidence of smaller-than-intended diameter shear pin installations (grouted and re-drilled) were found by either visual inspection or records reviews. Instances of deeper-than-intended, larger diameter pins were found; but neither instance employed redrilling or grout, and both were acceptable as-is.

Finally, even if the alleged situation were to have occurred, the shear pin would still conceptually function as intended since the pin imparts only a bearing load onto the concrete.

4. Allegation No. 9 - Fiberglass Piping in Screenwell

Fiberglass piping, originally installed in the Screenwell, was alleged to have experienced failures during testing and was replaced with ductile iron pipe. However, portions of this fiberglass pipe, which could not be reached (and currently inaccessible), were allegedly left as-is. The remaining fiberglass pipe is now alleged to be the system's "weak point", in its most inaccessible spot.

4.1 Scope

The details behind the design, preoperational testing, and modification of fiberglass piping used in the Screenwash system were investigated as part of NRC Inspection Report 50-322/82-29 issued on November 18, 1982. This inspection was conducted to follow-up on an allegation received in September 1982, which was similar (and practically identical) to the one above.

Both allegations refer to faulty piping in the Screenwash system, a non-safety related system located in the Screenwell which is used to clean the traveling screens in the intake as well as to educt water from the screenwell for de-watering. This system is not required for the safe operation or shutdown of the reactor, nor would its failure affect equipment that is.

Responsible test engineers familiar with the N71-Screenwash system operation and testing, were interviewed. The inspector then reviewed Inspection Report 82-29, including its pertinent referenced documents, and discussed the finding and details with the responsible senior resident inspector. Finally, inspection of installed fiberglass piping in the Screenwell was performed.

4.2 References

-- NRC Inspection Report 50-322/82-29, Details 3.5.1 and 3.5.2; conducted October 1-29, 1982, issued November 18, 1982

-- E&DCR Nos. F-27971 and F-20268 (entire series)

-- LILCo/S&W Dwg. No. FM-35B-8; Screenwash P&ID

4.3 Findings

Problems were experienced with screenwash fiberglass piping, in the period 1980-1982, with hydrostatic test leakage and dynamic (water hammer) operational problems. Also, corrosion of certain valves in the East and West Screenwell valve pits was experienced. The valves were replaced or overhauled with Monel internals which will resist saltwater corrosion.

The piping problems resulted in extensive redesign of supports, pump impellers, valve/pipe configuration and partial change-out of the fiberglass piping (manufactured by Seisbergage and supplied by A. O. Smith Company) with concrete-lined ductile iron pipe. Those portions of fiberglass pipe which remain, employ field-fabricated joints (150 pound or 300 pound fittings) which are reinforced. The system underwent successful pressure testing in March 1982.

The most critical and problematic portion of fiberglass reinforced piping was the 16-inch discharge line from the pump to the header. This was aggravated by the pump's characteristic flow and it's discharge valve's closing time. Shut- off head was reduced the discharge valve stroke time was lengthened and an approximate 20-25 lineal feet of 16-inch fiberglass pipe (on each discharge line leading to the header) was replaced.

Based upon discussion with cognizant test engineers and the inspector's review of the latest system drawings, it was estimated that a total of 50 feet of 16-inch fiberglass discharge piping was replaced; but, there still remains some 280 feet of 8 and 10 inch fiberglass pipe which is installed and accessible, should future replacement be required.

4.4 Conclusions

Only 15% of the Screenwash system fiberglass pipe was replaced with concrete-lined ductile iron pipe. This was not a large portion. Most of the remaining piping, which is still fiberglass, is accessible and has not required replacement - it has remained functional, has not "blown-out", and does not represent a system "weak-link".

The principal problems experienced with the initial operation of the Screenwash system involved dynamic water - hammer effects. These were corrected by hydraulic adjustments of pumps and valves, and replacement of a 25-foot portion of fiberglass discharge line most affected by the velocity head of the pump. Seawater and/or corrosion were not a contributing factor in the dynamic problems experienced with the fiberglass joints.

Finally, the failure of the Screenwash fiberglass piping has no effect upon the safe operation of the Shoreham reactor. These same conclusions were reached, in part, in Detail 3.5.2 of NRC Inspection Report 50-322/82-29.

5. Allegation No. 10 - Misorientated Check Valve

A 30-inch Mission check valve, installed in the Screenwell in either the Service or Circulating Water systems, was alleged to be incorrectly rotationally oriented. A Mission manufacturing representative, who was aware of this misorientation, was allegedly unsuccessful (and discouraged) in correcting this problem.

5.1 Scope

The purchase order specification for Mission check valves was reviewed to determine all 30-inch applications. The valve was then identified, using P&IDs and cognizant FQC engineers, as to system and plant location. Inspection of valve condition and orientation was then performed.

There are no larger check valves in the plant, and there are another 27 Mission insert check valves in the range of sizes 10 to 20-inches, installed. The two 30-inch TBCLCW check valves are in the vertical "up" position - there is no preferred rotational orientation in this configuration.

TRW Mission representatives were contacted by phone on October 17, 1983, and the Equipment Instruction Manual was reviewed to ascertain proper valve installation, orientation and operation.

5.2 References

- LILCo/S&W Dwg. No. FM-36A-17
  Turbine Bldg. CLCW Flow Diagram
- Specification No. SH1-306, Rev. 1 (8/27/81)
  Insert Type Check Valves (2-inches and larger)
  TRW Mission Manufacturing Co. (PO No. 310585)
  Mechanical Bill of Materials
- Equipment Instruction Manual 306-1,
  TRW Mission Insert "Duo-Check" Valve

5.3 Findings

There are only two, 30-inch Mission check valves installed at Shoreham; these are in the P-43, Turbine Building Closed Loop Cooling Water (TBCLCW) system. The valves (mark nos. VCI-15F-4; sequence nos. 9535 and 9540) are located in the Turbine Building, in a vertical run of discharge piping which is 7 feet above each TBCLCW pump, at elevation 15 and near the North wall.

The Mission "Duo-Check" valve uses 2 plates and 2 torsion springs which are hinged on a single pin. This pin is "full-floating" and carries no load with the plates closed. The dual plates (or flappers) are of use in checking reverse flow.

Normal installation for horizontal flow conditions would be with the pin vertically-oriented, in order for the dual-plates to swing properly. This rotational preference is unnecessary for a "vertical"-up application. The direction of flow is indicated by cast arrows on the valve body, and these two valves were found to be properly installed.

While the vertical-up installation required only a proper flow direction, it is recommended (but not essential) that the pin (horizontal in this configuration) be oriented parallel to the nearest horizontal run of upstream piping in order to minimize hydraulic disturbances and pressure drop across the valve. The position of the hinge pin may be determined by locating plugs with threaded-inserts, positioned 180-degrees apart, at the location of the pin retainers on either end of the hinge pin. These were also found to be properly installed.

TBCLCW is a non-safety related system, which has been successfully operated for the last two years.

5.4 Conclusions

There are no 30-inch Mission check valves found in the Screenwell. The only two such valves found onsite are in the non-safety related TBCLCW system, in a vertical-up piping run at each pump discharge.

The valves were found to be properly-orientated for flow direction, and hinge pin/dual-flapper configuration should additionally ensure operation at minimum pressure drop across the valve.

6. Allegation No. 11 - Control Building Fiberglass Pipe

Fiberglass piping carrying salt water in the Control Building was alleged to "blow apart" in testing. Rupture of this piping could allegedly flood the area, and damage the computers in the Building as it cascades down. Holes for drainage channels were allegedly drilled through the walls, to divert the potential flooding, and run it out of the building away from the computer room.

6.1 Scope

As with allegation number 9, (detail 4), the redesign, test and modification of that portion of the Service Water system which employs fiberglass piping in the Control Building Chiller Room were previously investigated as part of NRC Inspection 50-322/82-29. That inspection found the allegation received in September 1982, which is identical to the one above, to be unsubstantiated.

The alleged fiberglass piping is used to cool non-safety related heat loads of the Main Chilled Water system, and is isolable from the safety-related Service Water system. However, its rupture, while not having a direct effect on safe plant operation, could indirectly impact that operation by flooding of essential equipment located nearby. Rather than seismically qualify the piping, LILCo elected to provide floor protection for adjacent equipment.

As with allegation No. 9, responsible engineers familiar with fiberglass piping problems with this subsystem of Service Water in the Control Building were interviewed. Also reviewed were pertinent details of Inspection Report 82-29, including the resolution of NRC open item 82-29-01 (closed in Report 83-21) regarding a consolidated flooding analysis for elevation 44 in Control Building. The findings and details of these inspections were discussed with the responsible senior resident inspector. A walk-through of the Chiller Room on Control Building elevation 44 was conducted by the inspector.

Finally, the above were compared with analytical results and protective provisions presented in FSAR Subsection 3C.5.4.1 (page 3C.5-6).

6.2 References

6.2.1 NRC Inspection Report Nos. 50-322:
- -- 82-29, Detail 3.5.3 (p.8); issued 11/18/82
- -- 83-21, Detail 2.1.2 (p.3); issued 8/15/83
- -- 83-28, Detail 6.2 (p.7); issued 9/14/83

6.2.2 Shoreham FSAR, Revision 30 - June 1983:
- -- Subaction 3.8.4, Control Building
- -- Subaction 3C.5.4.1, Protective Measures to Mitigate Flooding Effects

6.2.3 S&W Letter to LILCo dated 7/10/81
Stress Analysis and Supports for Fiberglass Pipes, Main Chilled Water System Condensing Water

6.2.4 E&DCR P-3895

6.2.5 S&W Letter to LILCo dated 12/6/82
(LIL-21836)

6.3 Findings

A number of problems were experienced with the installation and testing of fiberglass piping in the Control Building, both with pressure and dynamic effects. Initial leakage problems were satisfactorily corrected; however, rather than seismically qualify the fiberglass, an engineering decision was made to protect other nearby equipment from fiberglass pipe rupture and flooding. This included: 1) core-drilled drains in the East wall at slab elevation 44 ft.; 2) 12-inch high curbs to contain flooding within the room; and, 3) protection of nearby electrical components from splash/spray by fire doors and sealed conduit. These provisions were described in FSAR Subsection 3C.5.4.1. There are six 8-inch diameter drain holes designed to eliminate potential flooding at lower elevations from this fiberglass piping by passing sufficient flow to limit water depth to 1 foot following severance of a 12-inch diameter Service Water line.

The modifications to mitigate flooding at Chiller Room elevation 44 were inspected, and found to be installed and appropriate. During conduct of Inspection 83-28, a damper on the Chiller Room floor drain was found to be stuck-closed due to recent painting. The problem was subsequently corrected.

6.4 Conclusions

The alleged (and solutions thereto) problems with fiberglass piping in the Control Building Chiller Room were accurately described. This situation was first alleged in September 1982, and investigated shortly thereafter.

The fiberglass piping is part of a non-safety related system, and was not seismically qualified. The potential consequences of its rupture - flooding on Control Building elevation 44 - were taken into consideration by LILCo and properly designed against, using drain holes, curbs and splash protection.

7. Allegation No. 14 - Reactor Building Accessibility

Maintenance, maintainability of equipment and surveillances which will be required during the plant's operating lifetime were allegea to be neglected, underestimated or approved. Examples given were snubber stroke-testing and weld ISI. Later clarification of weld conflicts (where ISI was alleged to be impossible to perform) was provided in a July 18, 1983 phone conversation, alleging 20 (total, unspecified) welds in the steam tunnel and various RHR system welds in the drywell.

Regarding accessibility, the drywell and steam tunnel were generally alleged to be cramped and confined. Maintenance and accessibility were suggested by the alleger for consideration prior to plant licensing; otherwise, it was alleged that given the alternatives of "servicing" versus "shutting down", the "history of the industry" dictates that the plant will not be shut down.

7.1 Scope

Three separate programs were reviewed, discussed with cognizant personnel, and then inspected: (a) The Maintainability Task Force (MFT); (b) The PSI/ISI weld programs; and, (c) snubber TS surveillance requirements, preoperational stroke testing and the related resolution of open inspection item 80-18-03 (see Section E of this report).

Inspector of various drywell locations and the Reactor Building steam tunnel was conducted. These areas are alleged as being confining, difficult to perform weld NDE or snubber surveillance, and a potential excuse for LILCo to request exemptions from such and stay on-line (vice shutdown for test).

Corrective action proposed and implemented by the MTF (established early in 1979), such as: lifting lugs, eyebolts, catwalks; recommended re-design and/or relocation of equipment; and the creation of the "yellow-painted" restricted space throughout the plant were all reviewed and observed by the inspector. Members of MTF were interviewed, and specific corrective actions proposed or implemented by the MTF for ISI weld and snubber stroke test accessibility were reviewed.

The PSI Program Plan, prepared by Nuclear Energy Services (NES) Inc., was used to locate various PSI weld locations.

7.2 References

7.2.1 Maintainability Task Force

-- MTF Problem Identification Sheets:

No. 0765 approved 5/9/80;	add platform in steam tunnel
No. 1370 approved 8/5/81;	install break flanges on MSIV leak off lines
No. 0587 approved 6/24/81;	add 2-ton pull points on steam tunnel ceiling for MSIV removal
No. 2745 approved 9/9/83;	provide grated walkway to preclude snubber dismantlement for equipment movement

-- LILCo Internal Memoranda:
March 16, 1979; Woffard to Novarro
March 16, 1979; Woffard to Dye
August 2, 1983: Chung to Hunt

7.2.2 Snubber Surveillance

-- TS 3/4.7.5, Snubbers, Revision 13, 6/15/83 ("Proof and Review Copy")

-- LILCo Internal Memoranda:
Smith to Jaquin, 9/29/83, (Feasability of "Scalloping")
Gentile to Miele, 9/12/83 (ALARA Study)
Giannattasio to Higgins, 7/19/83 (Bulletin 81-01)
Gallagher to Smith, 7/14/83
Smith to Higgins, 1/21/83

-- NRC Inspection Report Nos. 50-322:
80-18, Detail 5.b (p.4); issued 1/21/81
83-10, Detail 2.2.1 (p.5); issued 5/27/83

-- IE Bulletin 81-01, Surveillance of Mechanical Snubbers, issued 3/4/81

-- E&DCR F-24491 Series
Revision L, 3/31/83, Additional Snubber Stroking

-- Suppression Pool Hydraulic Snubber B21-PSSP-177 (Mark # S/N 379, replaced with S/N 380 spare) LDR 1335 (5/19/83), 1365 (6/1/83) and E&DCR F-45641A (7/11/83)

7.2.3 PSI and ISI Programs

- Preservice Inspection (PSI) Program Plan; prepared by NES Incorporated, Weld Map Drawings
- August 27, 1981 LILCo Letter to NES, Inc., Proposed Section III Class Welds to Include in PSI
- August 26, 1982 LILCo Letter (SNRC-759) to NRC, Smith to Denton; PSI Relief Requests
- NUREG-0420, Shoreham SER, Supplement 4, September 1983
- ASME Code, Section XI, ISI

7.3 Findings

7.3.1 Maintainability Task Force (MTF)

This group was created in mid-1979 to address existing plant accessibility/maintainability problems, and to prevent future problems from occurring as construction progressed. The group has been actively involved in the later stages of construction, redesign, and the typical interference difficulties experienced near plant completion. As of August 1983, and covering a 50-month stretch, the following measure of MTF work (i.e. MTF Problem Identifications or "MTF's" initiated and acted upon) is presented:

Total Identified:	2744
Corrective Action Implemented:	2623 (95%)
Physically Completed:	1446 (52%)

Approximately one-third, (or 1000), of all MTF's address problems in the Reactor Building. UNICO construction priority has been to close as many MTF problems within the drywell as possible, and as of November 1983, less than 10 were still open.

Typical of an MTF issue was a problem with access to the Feedwater valves inside the highly-congested steam tunnel. The MTF solution, implemented via E&DCR was to install a catwalk within the tunnel, from platform-to-platform, to facilitate access. As another example, MTF-2624, was written to add supplemental rigging points (e.g. eye-bolts, lifting lugs, monorails, etc.) to aid in inboard MSIV removal for maintenance within the drywell. This was in response to an ASLB Settlement Agreement for Contention SC-26 (ALARA).

The MTF was utilized to review snubber accessibility as part of LILCo's resolution of NRC open inspection item 80-18-03 (see Section 7.3.2 and Part E of this report), includirg the attendant ALARA evaluations. Notable contributions of MTF include reviews of support re-design under the Stress Reconciliation Program, and the "yellow-lining" of plant areas as out-of-bounds for last minute storage or re-location of equipment.

Plant tours of problem areas, identified (and fixed) by the MTF, were found to be representative of the proposed solutions and were effectively resolved.

7.3.2 Snubber Surveillance

NRC Inspection Open Item 80-18-03, closed in Part E to this report, questioned the lack of preoperational functional testing for safety-related mechanical snubbers which were listed in the Shoreham TS and would require stroke-testing once the plant began to operate. Such a preoperational test would demonstrate immediate operability, as well as the capability to actually perform the surveillance. While records indicated that snubbers were satisfactorily stroked at the factory by the vendor, and then by LILCo personnel upon warehouse receipt. The time elapased between that inspection and TS operability testing was potentially too long.

In response to Item 80-18-03, LILCo re-stroked random samples of all sizes of snubbers (30 of total 230 in drywell; all 7 hydraulic snubbers in suppression pool; 50 of an approximate 200 total in Secondary Containment and BOP; and 3 non-safety related INS snubbers). This representated a total of 90, or over 20%, of all snubbers. All 90 were satisfactorily tested, although 2 in the suppression pool (the only hydraulic snubbers in the plant) required replacement because of oil leaks.

To further address the NRC open item, LILCo's MTF undertook a study of the 100 large size snubbers (35 and 100 kips) installed within the drywell for accessibility and removability. While 84 of these were determined to be relatively easy to remove, the remaining 16 were found to be difficult, if not questionable, as to removal. Another snubber, on an RWCU line in the steam tunnel (G33-PSSP-244), was also characterized as questionable.

Upon identification of these 17 difficult snubbers, the Radiation Protection Group performed an ALARA study on the work expected in servicing a snubber for maintenance or surveillance. They were aided by the MTF, in estimating the time required for the five-phase process of mobilization, disassembly, stroking, re-assembly and demobilization. The assembly/re-assembly phases typically account for over 60% of the exposure, and the total average exposure times ranged from 5 to 7 hours per snubber. The man-rem predictions were, in almost all cases, less than one man-rem. The group concluded that, from an ALARA standpoint, removal of drywell or steam tunnel snubbers "did not present any remarkable radiation concerns." Further, employment of standard ALARA techniques (e.g. lead blankets, decontamination) would significantly reduce predicted exposures. All snubbers were therefore considered by that study as capable of access and removal.

Finally, S&W Site Engineering Office (SEO) was requested to estimate the cost of relocating the four worst (inaccessible and highest man-rem) of the 17 difficult snubbers. The feasibility and cost of "scalloping" certain members to facilitate snubber removal was also requested. Three of the 4 worst-case snubbers were concluded to be "impossible to relocate" by SEO; the fourth (1B31-SSA21) was estimated to require 2000 man-hours of work with a projected cost on the order of 100 thousand dollars.

It is concluded that, even if the LILCo projections were in error by one order of magnitude (for example, a cost of 10 thousand dollars for a reduction of 1 man-rem), that cost outweights the benefit of subber relocation. Also, conservative but well-conducted man-rem studies, coupled with actual pre-operational stroke testing of over 20% of all installed snubbers, demonstrate that all snubbers should be capable of being functionally tested per TS 4.7.5.e.

Snubbers Identified by MTF
and Analyzed (ALARA) by HP
as Difficult to Remove

Mark #.	Drywell Location	Projected Man-rem	Obstruction/ Comments (relocation)
1B21-SC8	el 98'	1.475	CRD lines (impossible)
1B21-SB8	el 98'	1.120	CRD lines (impossible)
1B31-SSB11	el 74'	0.815	Recirc. valves
1E11-PSSP904	el 76'	0.741	RHR conduit (imposs.)
1B31-SSA21	el 65'	0.520	Rupture Restraint
G33-PSSP-244	el 78' (Steam tunnel)	0.190	RWCU Piping (possible)

7.3.3 PSI/ISI Program

Residual Heat Removal (RHR)

RHR is one of the most extensive systems which occupy the drywell and Reactor Buildings. That portion which occupies the drywell has a total of 120 welds which are Code Class 1 included in the PSI program; 3 exemptions in this category were requested, and granted by NRR, due to a pipe-to-elbow part geometry obstruction which prevented 100% volumetric examination. Partial unaccessibility was experienced; however, a substantial percent coverage (ranging from 54 to 81% of scan) was achieved.

More current versions of ASME Code Section XI require less weld volume to be examined so that, for ISI during plant operating life, the above three PSI exemptions will actually receive 100% of the Code-required volumetric examination.

The welds are located near drywell elevation 90', at the loop A and B RHR recirculation inlet and outlets off of the mainsteam lines:

-- Weld E11-298D
-- Weld E11-303D
-- Weld E11-290A

Code Class 2 welds were not required to be included as part of PSI at the time of Shoreham's program development in the early 1970's (e.g. 1971 edition of Section XI). There are, however, 175 Class 2 welds included in the Shoreham PSI Program, of which about 90 are RHR piping welds. These are included since ISI is anticipated to require sampling examination of this Class. The RHR Class 2 piping welds selected are located in both the drywell (off mainsteam lines) and Reactor Building (off of RHR heat exchangers). Two RHR Class 2 exemptions were requested, and granted by NRR, due to a branch connection obstruction which allowed for only 84-88% scan coverage. The welds are located at pipe-to-reducer locations on the drywell spray headers, at elevation 70':

-- E11-IC62-FW6
-- E11-IC62-FW20

Therefore, of the over 200 RHR piping welds (Code Classes 1 and 2) which are included in the Shoreham PSI Program, only 5 were unable to be fully UT-examined. Further, the partial scans were better than 50% completed at the time, and will be fully volumetrically covered when later ISI Programs are in effect. There are estimated to be about 40 RHR hangers which were also examined as part of the PSI Program.

Steam Tunnel

That portion of the pipe tunnel which is part of secondary containment is of rectangular cross section, 28 ft. wide and 15 ft. high, and traverses 21½ ft. horizontally from the drywell to the Reactor Building boundary wall. The surrounding walls are 4-foot thick to protect against the high radiation fields caused by the four 24-inch mainsteam and two 20-inch feedwater lines contained within the tunnel. Other smaller lines in the tunnel include MSIV leak collection, Reactor Water Cleanup (RWCU), and HPCI and RCIC discharge piping. Of the estimated 9000 cubic feet of gross tunnel volume, a large percentage is actually occupied by equipment. The steam tunnel is very confined, and not strictly designed, for access when the plant is shutdown). It should be noted that, during reactor operation, the tunnel is unaccessible because of the prohibitively high radiation fields present.

There are estimated to be over 52 piping welds which received PSI examination in the main tunnel; an estimated 8 of these required some relief request from full 100% volumetric examination. The following breakdown of those welds was compiled from NES PSI weld map information:

System	Total PSI Welds	Exempt. Requests	Weld #.	% Scan Compltd
RWCU	20	1	G33-IC-1508-FW3	76%
Feedwater	16	2	B21-IC-173-FW1	96%
			B21-IC-175-FW1	95%
Leak Collection	2	0		
Main Steam	10	3	B21-N5001-BW15	97½%
			B21-N5003-BW12	97½%
			B21-N5004-BW12	97½%
HPCI	4	2	E41-IC-182-FW-2	96%
			E41-IC-183-FW-3	97%

All of the exemptions, involving pipe-to-valve branch connection obstruction, were granted by NRC's Office of NRR as documented in SER Supplement 4. The 8 instances of incomplete accessibility represent about 15% of all PSI welds in the steam tunnel and, with the exception of 1 RWCU weld, completed greater than 95% of the required UT scan volume.

General Discussion

The Preservice Inspection (PSI) Program, provides baseline information, prior to plant service or operation, for comparative purposes with later ISI. PSI is performed, via volumetric NDE-ultrasonic testing (UT), one time prior to commercial operation. ISI is then designed to detect generic problems in a system; individual defects are ideally picked up during construction inspection. Hence, ISI is not designed to examine every weld in the plant - if defects or problems are discovered, then the program requires increased or augmented inspection.

The earlier version of ASME Section XI-1971, to which Shoreham's PSI Program was patterned, called for 100% percent of all Class 1 welds to be inspected, but none of the Class 2 welds. LILCo did commit, however, to a sampling of 175 Class 2 welds since current versions of Section XI (e.g. 1980 Edition thru Water 1980 Addenda) do require Class 2 inspection. The latest Code requires "repeat" sampling of Class 1 welds and "composite loop" sampling of Class 2 welds. The former picks one loop of representative Class 1 piping and inspects one fourth of the same loop welds every 10 years (or 4 identical samples). The latter selects a composite loop of Class 2 piping and, over the 40-year plant design life, inspects a different 25% of the loop every 10 years (or 4 different samples) such that all composite loop welds are each inspected once. The samples selected should be "representatively" distributed among redundant trains, and "similar" in the sense of pipe diameter, wall thickness, geometry and line configuration. Welds thus selected are also considered, if possible, for ease of inspection and ALARA - both being accessibility considerations. Therefore, while the aim of the ISI Program is to assess the effects of operating stresses upon the as-constructed system (using baseline PSI data), not every weld is (or must be) inspected, nor does every weld have to be completely accessible.

An important distinction between PSI and ISI weld selection is than more welds receive PSI examination than do ISI - the PSI is expected to encompass the required ISI. Important attributes of the ISI weld examination program are as follows:

-- ISI is an accelerated program, with provisions for increased sampling as defects are found

-- Those inaccessible weld locations, where only a partial volume is examined, have documented exemptions

-- The program is a "living document" - 10 CFR Part 50.55a (g) requires the Shoreham ISI Program to be updated at the end of the first 10-year inspection interval, to meet future Section XI Code changes

-- The program allows for augmented inspection of special weld problems, experienced during Shoreham (or other generic plant) operation.

7.4 Conclusions

The Maintainability Task Force has been in existence since March 1979, and comprised of not only engineering and construction representatives, but maintenance and services personnel including craft foremen. The MTF has been concerned with the "feasibility of maintaining things", so that equipment was (and remained) accessible for surveillance and maintenance. The MTF was to identify existing plant accessibility problem areas, and prevent future interferences as construction was completed. So that future refueling downtime (especially the first scheduled outage) could be properly considered, the group allowed for meaningful planning by "the people who will do the work".

A measure of the group's progress is the initiation and implementation of MTF Problem Identifications. Over a 46-month period, beginning in the Fall of 1979, an average of 60 MIF's were generated monthly. A measure of LILCo management support of this program is the statistic that over 95% of all MTF's identified have had corrective action appropriately initiated.

A measure of the group's worth was that, as of August 1983, more than half of the total recommended field modifications (roughly 1400 of 2600) had been physically completed. The inspector concluded that the MTF:

-- demonstrated notable LILCo Management initiative
-- was an effective and useful program
-- did a good job with the right people
-- prepared the physical plant adequately for maintenance, Technical Specification surveillance, and general accessibility during operation and scheduled outages
-- utilized an appropriate balance of reactor safety, ALARA and cost-benefit analyses

Therefore, maintainability and plant accessibility have been given due consideration, up-front, prior to reactor licensing and operation.

Access to certain drywell locations, and within the main steam tunnel area, was found to be difficult in some cases. These areas are cramped and confining; however, required entries for maintenance, ISI weld examination, or snubber surveillance would not ordinarily be attempted while at power. The required ISI weld examination is performed every ten years, typically, unless higher incidence of defects or generic problems would be found. Similarly, the required functional stroke testing of snubbers is usually done for a random representative 10% sample of all types every 18 months, and this sample size and test frequency are increased with successive failures which may be found. Visual inspections of both ISI welds and installed snubbers are also performed regularly, every 12 months or less, in addition to the required functional testing. Both of these surveillance programs are capable of change or augmentation, if a generic problem is suspect, and both have provisions for escalated testing upon the discovery of failures. Neither ISI examination nor snubber stroking would be ordinarily performed during operation for drywell and steam tunnel equipment - thus, no decision between "servicing versus shutting down" is involved. Some amount of engineering/operating judgement is applied toward selecting those welds or snubbers to be inspected, and accessibility is a consideration in that regard, but this is required surveillance per TS and 10 CFR Part 50, and not an optional activity.

As for the ability to remove and/or functionally test snubbers, the licensee's thorough and responsive approach to the resolution of NRC open item 80-18-03, demonstrated that a representative 20% sample were pre-operationally functional (and by virtue of those tests, removeable), and that none of the over 450 installed snubbers were inaccessible from an ALARA standpoint. Predicated occupational exposures incurred in removal were shown to be typically on the order of 1 man-rem for the worst cases, and an order or two of magnitude less for all others. The MTF identified 16 drywell and 1 Steam tunnel snubber as "difficult", and their relocation was shown to be not cost-beneficial when compared with a value of 100 thousand dollars per man-rem reduction.

The inspector considered the results of these analyses to be sufficient for resolution of NRC open inspection item 80-18-03, and demonstrative that all snubbers were capable of access for maintenance and test, within reasonable radiological exposure constraints.

8. Allegation No. 16 - Buried Circulating Water Pipe

Twelve foot diameter, fiberglass-reinforced piping manufactured by Corban Industries and installed, underground onshore as circulating water system discharge lines, was alleged to have been changed from the originally intended pre-cast concrete. The alleger, in reviewing the Corban Installation instructions, pointed out a problem if the piping were to be installed "in the wet" (in a water-filled, as opposed to dry hole). The Corban requirements, which recommended that the bell-spigot mechanical joints (and rubber gasket) be kept free from dirt, sand and scale, could not be allegedly met with this installation. At a conference held between S&W, Corban and LILCo representatives, it was then allegedly decided to install the pipe "in-the-wet" for economic reasons, and to allegedly relax the cleanliness requirement. The alleger expressed a concern that, over time, the gaskets would become scored or scarred.

8.1 Scope

The installation of this non-safety relating piping was discussed with cognizant UNICO personnel, and pertinent records reviewed.

8.2 References

-- Courter Installation Procedure N-71, Circulating Water On-Shore Piping (7/25/77), Section 8, Joint Assembly

-- Courter Dwg. PC# Mk-144-750-10A N71 - Discharge Pipe (Ref. Corban Dwg. 9.15-2B; FP-310-6B)

-- Courter Installation Inspection Checklist, 11/10/77 (including Pressure Tests)

-- Notes of Conference No. F-473; SNPS Construction Office, 3/31/77

-- Notes of Conference No. F-444; SNPS Construction Office, 1/19/77

-- Corban Industries Fiberglass Reinforced Pipe Specification 432

-- NRC Inspection Report Nos. 50-322: 78-16, Detail 13 (p11), issued 11/28/78; 79-24, Detail 16, issued 4/28/80

-- Shoreham FSAR Section 10.4.5

8.3 Findings

The 144-inch onshore (buried) Circulating Water discharge piping was installed in the Fall of 1977. The pipe was originally intended to be pre-cast, concrete-encased, ductile iron pipe, but because of bidder and schedular conflicts, was re-specified as fiberglass reinforced pipe made by Corban industries. The 144-inch discharge pipe is downstream of the discharge lines from the main condenser, and was installed as twenty-one, 60-foot long spools, terminating at the Long Island Sound shoreline.

Spools were joined by a mechanical fit, employing a bell-over-spigot assembly with a double-gasketed joint. The pipe was installed in a 20 foot wide trench (4 feet of clearance on either side), which was typically filled (naturally) with 2 to 3 feet of tidal/ground water. Divers were used to remove sand from the piping and clean the bell and gaskets (if necessary) prior to fitup - the spigot end would be tilted up, out of the water, prior to its joining with the next spool's bell. Two requirements for the joint of any two pipespools were: (1) a maximum articulation of 1½ degrees; and, (2) 4 inches of pull-back. After placement of the piping (prior to backfill), a 33 psi pressure test was conducted on each joint as well as inspection for proper gap, alignment and gasket installation. Two such tests, dated November 10, 1977, were discussed with cognizant UNICO Construction personnel. All joint leakage tests were stated to be successful.

Conferences were held on January 19 and March 31, 1977 to discuss the Corban contract, including installation procedures. The alleger was present at the latter conference, during which R. Johnson, the Corban representative discussed cleanliness requirements. Conference Report No. F-473 indicated that:

- -- water-borne silt was not a problem
- -- precautions should be taken to prevent the sand from cutting the gasket
- -- sand in the bell should be removed

UNICO representatives who witnessed the installation of this pipe stated that some cover over the spigot end gasket was provided, until joining with the next piece. This spigot end was kept up, out of the water, until just before the joint was made. Divers removed sand from the bell at that point, so that the gasketed joint was free of sand. The inspector could not identify evidence or documentation which either proves or refutes this description of installation.

NRC Inspection 78-16, conducted in October 1978, reviewed portions of this piping installation.

8.4 Conclusion

The Circulating Water discharge piping is a non-safety related system, and the possible deterioration of the gasket used on any of the 20 mechanical fittings or bell-spigot joints in the buried 144-inch pipe poses no concern for the safe operation of the Shoreham reactor. Parts of the piping are installed below the tidal table and therefore most probably submerged, at various times, in the high tidal ground water in the vicinity of the Long Island Sound shoreline. Therefore, a water seal would be present and serve to prevent leakage out of this pipe. The deterioration of the gasket would not only be of no concern, but also possibly expected to occur.

Based on the discussions and documents referred to, there is indication that appropriate care was exercised in the fitup of these pipe joints, since:

- the spigot end was kept raised, out of the water
- the gasket was apparently covered until fitup
- divers removed sand from the bell-end
- pressure tests were successful
- no problems have been experienced in the last six years of system operation

Therefore, the installation "in the wet", and cleanliness requirements actually applied, were acceptable.

9. Allegation No. 17a - Fuel Load Date

LILCo allegedly contacted their (unspecified) "fuel loading company" in March of 1982, six months prior to their projected September fuel load date, in March of 1982. This unspecified company's representatives ("inspectors") allegedly stated that "...There's no way LILCo would be ready to load fuel". LILCo then allegedly decided to load fuel themselves, since they could not afford the "load company".

9.1 Scope

This allegation was stated to be a paraphrase of third-hand information obtained by the alleger, allegedly from a Courter & Co. reactor area supervisor. Based on vague, non-specific, subjective information, the alleger concluded that LILCo didn't "have a handle" on the project due to their alleged inability to accurately assess a fuel load date.

The identification of the "load company", and on what basis the alleged statements were made is difficult to ascertain. Further, the prediction of an NTOL's fuel load date, more than a year ahead, is a complex and often revised task.

While the allegation that an inability to accurately project fuel load implies an incompetency on the part of a utility would be hard to prove, there are valid indicators to judge Shoreham's readiness to load fuel. These include, in addition to construction completion:

-- NRC SALP Evaluations
-- Shoreham Master Punch List (MPL) Status
-- NRC Caseload Forecast Panel Reports
-- Preoperational Test Program progress
-- Closure of NRC open inspection items
-- Resolution of Licensing (NRR) items
-- Reconciliation of "As-Built" conditions
-- RAT Inspection 83-02

Since this allegation cannot be directly addressed, the above documents were reviewed to assess the realism of LILCo's September 1982 fuel load projection. A phone conversation with the responsible Chief of the NRC's Case Load Forecast Panel was also held.

9.2 References

-- NRC Region I SALP Report (Feb. 1983 - Feb. 1984) issued 5/14/84; Readiness For Operation (p 27), and Engineering and Design (p6).

-- NRC Inspection Reports Nos. 50-322:
82-26, Detail; 4, issued 10/29/82
83-02, Detail 10.6, issued 1/20/83
83-35, Detail 2.1, issued 12/14/83

-- Summary of 8/26/82 meeting held between LILCo and NRC Region I to discuss prerequisites for OL issuance.

-- Shoreham Project Fuel Load Schedule (as of February 1983)
-- March 29, 1982 LILCo Submittal to ASLB; Fuel Load Schedule

9.3 Findings

The NRC's Caseload Forecast Panel has visited Shoreham twice; initially in May 1980, and in August 1982. The second visit was a 3-day tour which found the following status:

-- 93% Construction completion
-- 63% Preop. Test completion
-- "Soft" schedule

According to the Panel's evaluation (which was never issued), the so-called "pacing" items remaining to be completed were:

-- TSC completion
-- LLRT, ICRT and Containment Structural Acceptance Test
-- Drywell painting
-- PASS Installation
-- Piping insulation/cable wrapping
-- Stress Reconciliation Program (calculations and hanger repairs)
-- Block wall modifications
-- Radiation Monitoring and Security systems

Review of the most recent SALP report and the RAT Inspection 83-02 indicated that over 300 open inspection items and 3000 MPL items (in need of prioritization) still remained to be resolved at that time (winter 1983).

The status of Shoreham's readiness for an operating license was discussed during an 8/26/82 meeting with LILCo at NRC Region I offices. The issue of readiness to load fuel at Shoreham was brought before the attention of the ASLB in march 1982. LILCo asserted a "strong likelihood" that Shoreham would be ready by September 20, 1982, citing ten milestones or prerequisite activities along the critical path in support of fuel load.

9.4 Conclusion

This allegation might well have been dismissed since it was subjectively vague, non-specific, third-hand information, and of no apparent safety significance. The "fuel load company" could not be determined, nor was the reason for the company's alleged statements even given (such as incomplete construction or pre-operational test problems).

Significant and unpredictable forces can come into play to influence the best-planned schedules; examples are Shoreham's TDI diesel problems, extended ASLB hearings, and the summer-1984 strike. Recent typical critical path items for Shoreham were completion of: (1) construction; (2) preoperational testing; (3) MPL open items, and (4) unresolved NRC inspection items.

Assuming that the fuel load company was GE, there is an Extended Services Contract which provides for GE STD&A engineering assistance during fuel load. However, Shoreham fuel load will be conducted by LILCo personnel because of their training and qualification (rather than for the alleged financial considerations). LILCo plant staff will perform refueling bridge work under the witness of LILCo nuclear engineers and supervised by a licensed Shoreham SRO, as required by plant Technical Specifications.

The Shoreham fuel load date, projected and published by LILCo, has changed a number of times. The most recent delay is attributable, in part, to the ASLB hearings and the TDI diesel failures. At the time of the alleged "unrealistic" September 1982 projection, an NRC Management Meeting (8/26/82) and Case Load Forecast Panel site visit (8/11 thru 13, 1982) were held. These found the LILCo schedule to be in need of refinement to better reflect incomplete activities such as construction, preop. testing, and closure of open items. However, while hindsight proves that September 1982 was most probably not a realistic projection (6 months prior), it did not indicate incompetence on the licensee's part. Rather, it was more likely indicative of their aggressive pursuit of a "soft" schedule. The RAT Inspection conducted in January 1983 found that the plant would not be ready to load fuel for a period of at least 5 to 6 months (at that time), the principal problems being a higher reject rate experienced by final FQC inspection and approximately 30% of all systems remaining for preoperational turnover. It should be also noted that the subsequent projected fuel load date, for internal planning, became June 1983 - another date which later had to be extended.

Recent Region I SALP evaluation found adequate staffing, aggressive closure of open MPL and inspection items, and concluded that the plant and its personnel were prepared for operation. A detailed NRC Region I staff evaluation will be prepared, prior to any recommendation for a license, which will consider the licensee's readiness to load fuel and operate the plant.

10. Allegation No. 17b - Misstacked Fuel Assemblies

New fuel assemblies were alleged to have been stacked on the refueling floor greater than four-high. Two separate third-hand accounts were presented by the alleger; one report alleging eight-high, the other five-high. This handling was alleged to be in excess of signs on the new fuel "boxes" which were marked "do not stack more than four high."

10.1 Scope

This is a third-hand, "hearsay" allegation, with no distinction between the wooden (outer) shipping container (WSC) and the metal (inner) shipping container (MSC).

Records of LILCo OQA procedural handling, audit and inspection of new fuel assemblies at Shoreham during the period July-August 1982, were reviewed and discussed with responsible personnel. GE's Wilmington, NC Fuel Manufacturing Department was contacted by phone on October 19, 1983 to discuss labelling and recommended handling of new fuel.

10.2 References

-- GE Fuel Manufacturing, Wilmington, NC, Nuclear Safety Release Instructions 4.8.12, 13, and 14

-- Station Procedure SP No. 58.001.01 (Rev. 5, 4/30/82), Receipt Inspection and Channelling of Unirradiated Fuel, Precaution 4.1

-- LILCo OQA Surveillance Plans (7/28-8/3/82), New Fuel Storage on Reactor Building Elevation 175 ft., Item No. 9, "Assure that the MSC are Stacked in the Required Sequence".

-- LILCo OQA Inspection Reports (7/23/82 and 8/9/82), New Fuel Inspection

-- LILCO Deficiency Reports LDR-0772, 7/26/82, Hold Tags; LDR-0791, 8/16/82, Cleanliness

-- NRC Inspection Report Nos. 50-322/82-15, Detail 7 (p. 12), issued 8/30/82; 83-03, Detail 3 (p. 3), issued 2/15/83

10.3 Findings

LILCo Operational Quality Assurance (OQA) maintained round-the-clock inspection and audit of new fuel handling and storage on the refueling floor at Reactor Building Elevation 175. Extensive procedural direction and precautions were contained in SP

58.001.01; precaution 4.1 prohibited stacking MSC more than four high (no wooden containers were lifted to el. 175). This warning was not placarded or otherwise marked on the MSC. OQA surveillance plans, inspection reports, and an estimated 25 LDR's written, were reflective of a carefully controlled activity. No record of over-stacking was found in these documents, and OQA personnel were unaware of any such instances.

GE Wilmington, NC representatives were contacted, and explained that GE was licensed to stack the MSC four-high, and the WSC five-high. The outer WSC is marked that way; the inner MSC is not. The bases behind the stacking height is of significance for the containers toppling over, rather than a criticality concern.

NRC inspection coverage of new fuel receipt and storage was documented in Report 82-15; review of documentation was accomplished as part of Inspection 83-03.

10.4 Conclusions

No evidence of misstacked new fuel, either in the WSC or MSC, was found. Only the WSC were marked (five high, not four) with a maximum stacking precaution. The MSC/WSC stacking precaution is for physical protection as opposed to a criticality concern. LILCo OQA provided a extensive and thorough procedural direciton and inspection coverage of new fuel receipt and handling. This was augmented with previous NRC inspection of new fuel activities, which did not identify any problems with the receipt, handling and storage of new fuel.

11. Allegation No. 20 - Overloaded Reactor Building Column

A column (unspecified) in the Reactor Building was alleged to be overloaded, and "nine months" of previous hanger work had to be redone to redistribute column loads. Although not specifically alleged, a single large support was apparently the cause of this problem. The alleger questioned the column's design margin.

11.1 Scope

No specific column or support was identified in the allegation. There are twelve rectangular columns in the Reactor Building, and their load calculations were reviewed with the S&W-Boston Chief Structural Engineer. To associate a specific support with this alleged condition, although difficult, the inspector assumed it to be RHR restraint PSR-054.

A distinction should be made between actual overloading of a column or member, versus the potential for a theoretical overload. Cognizant structural engineers identified Reactor Building Column-12 as the column whose shear capacity required reconfiguration of a support.

11.2 References

-- S&W Reactor Building Column Check, 2/8/83 (pages C16-2456, 2464, 2434)

-- Interoffice Memo (Wiebel to Glazier), 10/4/83

-- AISC and ACI Code

11.3 Findings

Current Column-12 design loads are enveloped by the larger Column-10 loads. S&W calculations conclude that AISC minimum reinforcing steel is adequate to resist these loads, and that significantly more than required rebar was used.

Calculation Page C16-2456

Column-12 is a 50 x 50 inch square, reinforced concrete member, over 100 feet in height, from Reactor Building elevation 8 to 112'. It is expected structural loading is as follows:

-- P_u = 2294 kips axial load

-- M_u = 979 kip-ft moment

Calculation Page C16-2454

The current (as of November 1983) largest theoretical column loading is for Column-10:

-- Pu = 3267 kips axial load

-- Mu = 1777 kip-ft moment

About 85-90% of a column's axial loading is typically the result of slab dead-weight. The above calculations were performed on February 1, 1983 by S&W structural engineering for new loads added to Column-10; which is identical in size to Column-12. The new Column-10 loading was as stated above. Calculation of the critical moments for this column, and comparison with Code strength requirements (i.e., required amount of rebar) showed that the total amount of steel reinforcement (A_{st} = 64 in^2 of rebar) provided for the column is 2½ times more than the minimal amount required. Therefore, the Column-10 strength is in excess of its theoretical maximum - predicted loading.

Since Column-12 is identical in design to Column-10, and its predicted loading is less (by 70% axial; 55% moment), it was less critical and therefore also adequate.

Calculation Page C16-2434

A calculation check on 3/2/82, of the shear capacity of all Reactor Building columns, showed the maximum allowable shear stress per ACI formula to be 110 psi. For the smallest column size (42 x 42 inches), this results in a maximum allowable sheer loading of 194 kips; for the column size (50 x 50 inches) of interest, this value is 275 kips.

Intermediate shears, between floor slab elevations, are created in the Reactor Building columns due to pipe support loads. Shear loading at the junction of slab and column is not critical for the column, since that load is transferred by the floor slab to the Secondary Containment wall. Shear loads from supports were frequently changed, when support redesign was at its peak in the late 1970's - early 80's as a consequence of the Mark II program. The support loads are complex combinations of many variables, such as earthquake, transient Mark II dynamic phenomena, thermal growth and dead-weight.

The following six columns were found to have the worst-case, maximum expected shear values:

Column #.	Max. Expected Shear (kips)	Percent Allowable*	Design Margin
5	83.3	42%	2.4
2	77.0	39%	2.6
4	70.0	20%	4.9
12	65.0	24%	4.2
10	49.3	18%	5.6
11	48.5	25%	4.1

*based on column size and 110 psi shear stress

These values indicate that there is adequate design margin with respect to shear loading on these columns:

-- Factor of 2 or more for all columns

-- Factor of 4.2 for column 12

-- Factor of 4.0, on average, for the six highest-loaded columns

Column-12

This column is at about average shear load for the six highest-loaded Reactor Building columns, when the as-built maximum predicted shear of 65 is compared with the Code-allowable value of 275 kips.

S&W structural engineering personnel did not identify the precise time, support and corresponding maximum shear stress which resulted in the alleged redistribution of loading on Column-12. The inspector did not attempt to clarify those details. The S&W Lead Structural Engineer related that, he remembered a support which, in the course of redesign, was found to contribute an additional shear on the order of 100 kips to Column-12. An engineering judgement was made to redistribute this loading, since it was felt to be too close to the maximum allowable by Code.

If an additional 100 kips of shear were to be presently added to Column-12, the maximum shear would become 165 kips, which would be 165/275 or 60% of the allowable limit. This represents a design margin of about 70% (factor of 1.7). Structural engineering practice has been to limit the percent of allowable shear to less than 50%, thereby maintaining a design margin which is a factor of 2 or greater than the conservatively predicted Code value. Thus, the re-distribution of the additional 100 kips of shear on Colum-12 was a reasonable and prudent engineering decision.

A major shear load on Column-12 is produced from pipe restraint PSR 054, a large, complex RHR support at elevation 63, employing a number of baseplates (one directly attached to Column-12). This is a support which underwent over ten instances of documented redesign, from September 1979 through June 1983, and could have been, the support which was alleged to potentially overload Column-12 since:

-- The alleger worked on this support, as determined from interviews of cognizant personnel

-- It is supported, in part, off of Column-12

-- It's one of five largest supports in the Reactor Building

-- It underwent a number of design changes, redistributing loads by relocating the baseplate

-- Although never specifically identified by support, the alleger referred to a number of situations (e.g., column overstress, shear pin issues, verbal authorizations, etc.) which have a resemblance to this support and Column-12.

Refer to Detail 3.3.1 for a discussion of RHR support PSR-054.

11.4 Conclusions

Column-12 and all other Reactor Building support columns are found to be designed with adequate margin, in excess of a factor of two. The six highest loaded columns are "over-designed" by an average factor of 4; all twelve columns, when averaged, are designed with a safety factor of 8.

The potential overstress of Column-12 never actually occurred; rather, in the course of support redesign (probably RHR restraint PSR-054), an engineering judgement was made that the maximum predicted shear stress was too close to the maximum allowable by Code. Therefore, the configuration of PSR-054 was changed to redistribute its loads on nearby structures and reduce the shear stress on Column-12. This is typical of the engineering design process, and indicative of sound and conservative engineering judgement.

12. Allegation No. 23 - Suppression Pool Liner

The stainless steel liner in the suppression pool was allegedly never installed, and was eliminated in lieu of an epoxy paint which was placed on the walls and columns. After reading a Bechtel report concerning post-TMI cleanup conditions, the alleger stated his concern that this epoxy would peel off after an accident, and would be difficult to decontaminate. During the July 13, 1983 interview, the alleger also claimed that embedment plates were put in to receive the liner, not just on the floor and walls, but on anything that came in contact with suppression pool water, including columns.

12.1 Scope

Records for the steel liner and painting systems inside the suppression pool were reviewed.

12.2 References

-- Shoreham FSAR Subsection 3.8.1.1.3, Primary Containment System Steel Liner
-- Shoreham FSAR Figure 3.8.1-11 (Rev. 16), Reactor Containment Liner Elevation
-- Specification SH1-75 (Rev. 4, March 1982) Shop Fabrication and Field Erection of Steel Plate Liner
-- PDM Construction Procedure Sequence for Contract 10093
-- PDM QA Examination Check Lists
-- S&W Drawing No. M-10166
-- Specification SH1-228 (Rev. 2, October 1981); Section 4-1, Protective Coatings Within the Suppression Chamber
-- FSAR Subsection 6.2.1.6, Materials
-- KTA Daily Painting Inspection Reports
-- S&W Memorandum to LILCo dated 6/10/77 (LIL-10852); Stainless Steel Cladding of Suppression Pool Columns
-- NRC Inspection Report Nos. 50-322:
   79-20; Detail 9; (p.7) issued 4/28/80
   79-24; Detail 21 (p.44); issued 1/15/80

12.3 Findings

The primary containment system is a continuous steel membrane backed by reinforced concrete (except at the drywell head). A carbon steel liner, 1/4-inch thick on the floor and walls of the suppression pool and 3/8-inch thick within the drywell, acts as an impervious gas-tight membrane. The liner plates were welded together and all seams or penetrations were tracer-gas tested. The liner was then painted to protect against corrosion. The containment liner was erected in 1973-74 and later sandblasted and painted in 1977-79. The liner is painted on both the floor and walls inside the suppression pool. However, there is no steel liner installed on any other interior concrete surfaces, such as the twelve Reactor Building columns or the pedestal wall. Those exposed surfaces are painted.

During the early 1970-72 conceptual phase of design, it was proposed to install thin, 16-18 gauge, stainless steel sheet metal cladding on the columns within the pool. Continuous vertical strip plates were installed in 1975-76, on the columns (on either side), to attach the cladding. A number of engineering evaluations during 1977 concluded that the installed cost of the cladding would be prohibitive. The proposal was rejected, and the concrete columns were recommended to be protected (i.e. decontaminable surfaces) with the same painting system used on the pedestal wall and all other exposed concrete surfaces within the pool, at roughly one-fifth the cost of the cladding. The vertical embed strips were left in-place, and are still installed, today. Later-recognized dynamic phenomena associated with Mark-II containment loads, such as pool swell, would have probably caused the stainless steel column cladding to be removed.

The liner within the pool, as well as all exposed concrete surfaces, are protected by a Keeler & Long Inc. painting system which is designed for the plant's lifetime. This system is of proven high decontaminability (99.8% DF). The paint is tested and qualified to recognized standards (ANSI N101.2 - 1977 and N512 - 1974). Below elevation 30, where concrete surfaces are in contact with pool water, a high-resistant, heavy-build epoxy phenolic (Plasite-7155) is applied. Above this elevation, two coats of K&L-6548 and a white finish coat of K&L-7475 protective coating are applied for a total dry film thickness (DFT) of 8.5 - 13.5 mils. The embed plates were not coated.

Painted surfaces within the suppression pool were observed to be in satisfactory condition by the inspector. Painting procedures, records and in-progress inspection coverage were previously reviewed in NRC Inspections 79-20 and 24.

12.4 Conclusion

The suppression pool floor and walls are lined inside by painted, ¼-inch thick carbon steel plate which is a vapor/liquid-tight barrier. The protective coating system (paint) is of proven, high decontaminability.

The alleger's statement that stainless steel within the pool was eliminated, and replaced by epoxy coating, is true. However, the pool liner is carbon steel (not stainless) and is still installed. Rather, stainless cladding for the columns within the pool was originally proposed and later eliminated for epoxy. The embed strips for the cladding remain installed (and unused) today. The pedestal wall was never intended to be lined with steel, and is painted.

13. Allegation No. 26 - Contaminated Tool

A radioactive tool was alleged to have been delivered from Brookhaven National Laboratories. The tool was alleged to be a wrench used for work on the main steam isolation valves (MSIV's), located within the steam tunnel, sometime in 1982. The tool was used by Startup personnel, and allegedly discovered to be contaminated; the alleger found out about the incident because of general discussions on-site. It was alleged that the incident should have been reported to the NRC, but was not.

13.1 Scope

The inspector interviewed licensee representatives familar with this incident. GE Startup engineers directly involved in the MSIV refurbishment, and LILCo HP's who performed surveys during this incident were interviewed. Phone conversations were held with Rockwell (Raleigh, NC) representatives responsible for providing this tool to Shoreham. Finally, cognizant NRC personnel were contacted to assess the implications with respect to radiation protection, transporation and nuclear materials control.

Reviewed were Radiation Survey sheets, an MSIV Maintenance Report, and Material Receipt & Return Reports. This tool was actually a hand-held portable valve disc lapping device, received and used at Shoreham during the last week in March 1982. The contamination levels found were compared with Shoreham procedures for the receipt and handling of radioactive material, as well as with NRC guidelines and regulations.

13.2 References

-- Shoreham Radiological Survey Sheets (Nos. 0105 thru 0112)
-- Shoreham MSIV Maintenance Report dated 12/82
-- NRC Region I Memorandum (Kelly to Greenman) dated November 8, 1983
-- IE Circular 81-07 dated May 14, 1981; Control of Radioactively Contaminated Material
-- 10 CFR 30.71, Schedule B (Exempt Quantities)
-- 49 CFR 173.389(c)(5) and (e), Radioactive Materials Definitions
-- Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, June 1974
-- LILCo Repair/Rework Request B21-147
-- Eberline Hand Probe, Model HP-210, Specifications (1/78)
-- LILCo Material Receipt Report No. 82-2747 (3/22/82) and Return Material Report No. 82-0326 (3/31/82)
-- Shoreham Procedure No. 61.020.07, effective 3/19/81

13.3 Findings

13.3.1 Background

A radioactive tool was confirmed as showing up onsite during late March, 1982.

Local leak rate testing (LLRT) on the eight main steam isolation valves (MSIV's), initiated in the Fall of 1981, resulted in four of the valves failing, necessitating disassembly and repair. Inspection of the first of these showed a heavy scale accumulation. Based upon this condition, and GE-recommended maintenance for the valves hydraulic operators, it was decided that all eight MSIV's would be disassembled for refurbishing. These valves are manufactured by Rockwell (Rockwell-Eward Flite-Flow Balanced Stop valves, size 24"x20"x24", Mark Nos. 1B21*AOV-81A thru D and 82A thru D). The four outboard valves are located in the Reactor Building Steam Tunnel at elevation 80. The allegation referred to work in that area, on these valves, during this time frame.

Maintenance on the MSIV's was initiated by Repair/Rework Request B21-147, and continued throughout calendar year 1982. This work was documented by a Maintenance Report prepared by the GE Startup Test engineer.

13.3.2 Valve Disc Lapping Device

A slightly contaminated tool was identified onsite, as evidenced by surveys which had been performed from March 24-29, 1982. The tool was a hand-held inner-disc "lapping" device, provided by Rockwell directly to the Shoreham site. Rockwell personnel in Raleigh, North Carolina and LILCo/Shoreham site employees (including General Electric engineers on contract to LILCo Startup and Maintenance) identified the responsible engineer from GE who supervised the 1982 MSIV refurbhishment as Mr. Arlo Ketcham (currently lead NSSS engineer at Shoreham). Another GE employee (Charles Clark, who was contacted on November 1, 1983 by Mr. Ketcham) was the individual who arranged to have this tool sent directly to the site, without a Rockwell representative accompanying it.

Mr. Ketcham verified that:

(1) the tool arrived in a crate without any special labelling or stickers;

(2) he suggested that the tool be surveyed since it was worn looking and rusty, indicating to him that it had been used (possibly at an operating plant) and could be contaminated;

(3) surveys performed indicated a slight amount of fixed contamination, the majority of which was determined to be beta; and

(4) after approximately one week of use during the last week in March 1982, the tool was decided to be ineffective for lapping the MSIV, and was shipped back to Rockwell in Raleigh, North Carolina in the same manner that it arrived.

The tool was received at Shoreham by air freight on March 19, 1982 under Rockwell Order No. 3676522, and subsequently left the site on April 1, 1982 by air freight, all under the referenced material reports and as part of LILCo PO. No. 375359. The tool was identified by these shipping papers as portable lapping tool FM4-13 (and disc FM4-13-10 with expendable emory cloth), to be used on the MSIV's. Rockwell records for the number 10 disc-type lapping tool indicate that the only nuclear facility at which this tool was used immediately preceeding this time-frame was CP&L's Brunswick station (shipped to that site on July 18, 1981). Since there are three identical FM4-13 lapping tools owned by Rockwell, its difficult to ascertain which of the three was either at Brunswick or at Shoreham.

13.3.3 Radiological Considerations

When Mr. Ketcham requested a survey, the Shoreham Radiological Engineer initiated surveys which were performed twice per day, morning and afternoon, from 3/24 thru 3/29. These indicated the following: 1) contamination was combined beta-gamma with no alpha considered present; 2) 1000 counts per minute contact reading in all cases; and 3) all contamination was fixed (i.e. smears taken showed no detectable results) and localized to three or four spots on the tool's "cutting edge", thereby presenting no airborne radiation hazards. Interviews conducted with HP personnel assigned to monitor the work performed with this tool, as well as GE test enginners present, verified that the licensee properly surveyed, monitored, and controlled such activities to ensure no radiological problems were incurred.

Assuming that the detector used in these surveys, a Victoreen 496 GM with an Eberline HP-210 probe, has an efficiency of between 1-10% for the combined beta-gamma source which it was monitoring (most probably Cobalt-60 and Cesium-134/137 residual "crud"), then the measured count rate represents anywhere from 10 to 100 thousand dpm. This represents a total estimated activity (assumed as unknown isotope since no such breakdown was done) of between 5 and 45 picocuries (pCi). Further, the contact dose rate was approximateed as 0.20 mR/hr. The detector characteristics were based upon typical values taken from the manufacturer's specifications.

These results (5-45 pCi and 0.2 mR/hr contact) are conservatively high in that the majority of contamination was suspected to be beta. The beta efficiency of this probe is much higher, resulting in a lower estimated total activity as well as a lower projected contact dose rate, for two reasons: 1) when the detector was moved a small distance away from the tool, the current rate became undetectable, and 2) when a thin piece of paper was placed directly between the probe and tool (essentially still in contact), the count rate disappeared. Upon evaluation of these surveys, and after it had been decided to not use this tool anymore, LILCo HP determined that no furhter measures were required (such as special labeling, handling or shipping) to either control or ship this tool due to the following reasoning:

-- conservatively estimated activity and dose rate

-- this contamination level represented an exempt non-licensable quantity (per Part 30), by comparison with the Part 30 limit of 100 pCi for unspecified isotopes (excluding alpha)

-- When considering the reference DOT definition of a radioactive material ("...estimated specific activity greater than 2 pCi/gm, essentially uniformly distributed..."), and assuming the contamination to be uniformly spread over a ten pound weight (grinding wheel), the tool represented two to three orders of magintude less specific activity (than the definition) and hence did not fall under any applicable DOT requirements

-- a dose rate, and lack of removable contamination, which present no radiological hazard

-- no applicable reporting requirements in accordance with 10 CFR 20 and 71, or 49 CFR 173.

13.3.4 Procedural/Shipping Controls

This tool was authorized for shipment under Rockwell Order No. 3676522 by a Rockwell service representative. Typically, a tool which is leased (as opposed to a "loaner") is usually accompanied by a product service representative, under the cognizance and direction of Rockwell's Customer Service Supervisor, who maintains records of tool distribution and allotment as well as personnel assignments. In this particular case, the hand-held portable MSIV inner-seat lapping tool FM4-13 was shipped directly from the Rockwell facility to Shoreham, in care of Shoreham's Resident Engineering Office, for LILCo Startup. The tool, including the disc and emory cloth, was leased to LILCo at a rate of $75.30 per day, and is one of three similar tools (all serialized as FM4-13) owned by Rockwell.

The Rockwell radiation safety officer described the program in effect at Rockwell's Raleigh facility to assure that tools are properly handled. The program is based upon reliance on the licensee's who use, decontaminate, and return such equipment, coupled with spotchecks or surveys performed by Rockwell's consultant for such work (NC State University). Decisions to sample certain leased tools which are returned from operating plants to Rockwell are based upon the following considerations: 1) established reliability and performance of customers; 2) knowledge of what type of equipment is being returned (e.g. valve parts, hard-to-decon threaded equipment, etc.); and 3) from who and when it is coming. The Rockwell accept/reject criteria are based principally upon: 1) reliance on their contractor's knowledge of NRC and DOT regulations; 2) the presence of removable contamination in excess of 200 dpm/100cm^2 and dose rates typically less than 0.5 mR/hr;

and, 3) confidence in licensee programs for release of such materials. Rockwell has an industrial radiography license, and is regulated by North Carolina which is an Agreement State.

Shipping records at Rockwell's Raleigh facility were examined by their representatives to determine where (e.g. which nuclear facility) this tool may have been contaminated. This is difficult to determine since this tool (with the FM4-13-10 disc, which is what actually was contaminated) is one of three identical lapping tools. Such a tool was shipped to CP&L's Brunswick site in July 1981; this may have possibly been the same tool which was subsequently shipped to Shoreham in March 1982.

13.3.5 NRC Regulations and Guidance

IE Circular 81-07 and Regulatory Guide 1.86 present guideline levels of surface contamination below which it is acceptable to dispose of decontaminated materials to unrestricted areas. These levels are expressed in disintegrations/minute (dpm) per 100 cm^2 for fixed and removable beta-gamma activity, as well as exposure rates at 1 cm, and are as follows:

5,000 dpm/100cm^2 (average) & 0.2 mrad/hr @ 1 cm
15,000 dpm/100cm^2 (maximum) & 1.0 mrad/hr @ 1 cm
1,000 dpm/100cm^2 - removable

The tool used on the Shoreham MSIV's during the last week of March 1982 was estimated to be within these limits. Based on these considerations, no radiological hazard was presented in shipping this tool back to Rockwell.

While the Shoreham procedure in effect during March of 1982, governing the release of tools and equipment from a Restricted to an Unrestricted Area, precludes such removal if fixed contamination levels are greater than or equal to 0.1 mR/hr (which was the case for the lapping tool), this area was not classified as Restricted in the radiological sense at that time, nor should it have been. This underlines the fact that this tool was not contaminated while at Shoreham. On the contrary, were dose rates higher or removable contamination found, the licensee would have most probably classified this area Restricted. No violation of this procedure is apparent in this instance, nor were any violations of NRC regulations found in Parts 20, 30, and 71 found.

There's a "gray area" in what's considered to be very low level LSA (Low Specific Activity) material versus what's legally (by 10 CFR 71 or 49 CFR 173) determined to be nonradioactive material. The contaminated tool at Shoreham cannot be technically considered as non-radioactive material; rather, its more properly classified and considered (as defined by these regulations) to be LSA or limited quantity material. This is because the contamination cannot be considered as "essentially uniformly distributed." Thus, any contaminated tool (no matter how well-decontamintated) which has a known detectable level (however small) could be argued as having to be considered "radioactive material" (per 49 CFR 173.389(e)) and hence regulated (no matter how slightly). The DOT regulations for limited quantity material in this case, for example, would require a sticker marked "Radioactive" and shipping papers identifying it as such.

The NRC recommends a lower limit for decontamination and unconditional release to unrestricted areas (i.e. 5,000 dpm/100cm^2 total and 1,000 dpm/100cm^2 removable with 0.2 mrad/hr average at 1 cm) for purposes of decommissioning work. These guidelines do not appear in any NRC or DOT regulations, IE Circular 81-07 has been misinterpreted as an acceptable release threshold; however, the limits in this circular refer to detector limits for material which is not expected or known to be contaminated (as opposed to a known contaminated or detectable tool).

An apparent minor violation of DOT requirements (49 CFR 173) may have occurred insofar as this contaminated tool: 1) cannot be considered as "nonradioactive" (per Part 173.389(c)(5)); 2) meets the definition of limited quantity material (per 173.391(a)) which would require the outside of the inner container bear the marking "Radioactive" with shipping papers which identify it as such, as a minimum; and 3) is therefore considered as regulated, however minimal the provisions must be.

The classification of this tool is complicated by 10 CFR Part 71, which clearly applies to "licensed materials" only. It is this position which the licensee assumed upon discovery of the tool; namely, that the tool did not represent a licensed Part 30 quantity and was exempt from NRC requirements.

13.4 Conclusions

A contaminated tool did show up at Shoreham in March 1982, and was associated with Startup work on the MSIV's in the steam tunnel, as alleged. The tool did not, however, originate from Brookhaven nor was it required to be reported to the NRC.

The incident was of minor radiological significance, and a violation of NRC or DOT regulations is not appropriate, in so far as:

-- This event had no safety or environmental significance in that dose rate was on the order of a tenth of an mR/hr, with no removable contamination, and involved an insignificant non-licensable amount of activity.

-- The contamination did not originate at Shoreham, nor was LILCo notified of any contamination prior to and upon receiving the tool.

-- LILCo acted responsibly and prudently in initiating a survey, providing HP coverage, and monitoring for removable activity until the tool was shipped back to Rockwell.

-- NRC and DOT regulations in this instance do not address the issue of "De Minimis" levels clearly.

-- Contaminated tools and equipment may be routinely released from operating plants at specified administrative limits similar to those encountered in this case, without any special handling or labelling, for ultimate unconditional unrestricted use.

14. Allegation No. 32 - Underdesigned PASF Walls

The alleger was told by other workers that a wall in the Post Accident Sampling Facility (PASF) was designed improperly, such that post-accident radioactivity would be expected to "penetrate the wall". Vague references were made by the alleger to a hanger design problem, generic to Stone & Webster computer calculations, which was alleged to have possibly been related to the improperly designed wall in the PSAF. To make the wall more-resistant to radiation, four 1-inch thick steel plates were bolted to the wall.

14.1 Scope

PASF design was reviewed and cognizant personnel were interviewed. FSAR Section II.B.3 describes the location of the facility, on the south side of the Reactor Building directly adjacent to the truck access lock. The facility is a seismically-designed, two-story structure, divided into four distinct areas. It houses the sampling system (PASS) and is accessible following an accident. The inspector reviewed design drawings and observed the location of the alleged wall.

14.2 References

-- FSAR Section II.B.3, Post-Accident Sampling, p. 1-8.

-- S&W Dwg. No. FM-10A, Machine Location, PASF, 11/13/84

-- S&W Dwg. No. FP-86E, Floor Wall Sleeves and Penetrations, PASF, 2/3/83

14.3 Findings

The alleged wall is an interior, poured concrete wall on the first floor (el 40'-6") of the PASF, approximately 12 feet high and 6 feet long. The wall is 4-foot thick reinforced concrete, with 1-inch thick carbon steel plates on either side, and functions as a shield between the sample skid and the PASS control panel.

The wall was originally designed as concrete encased with steel plate, for shielding of the gross gamma detector located near the sample skid. The steel was added in 1982, as four, 1-inch thick plates, bolted to the wall on either side. The steel is in preference to a thicker concrete wall (with no steel plate), since approximately three inches of steel would be equivalent to 12 inches of concrete with respect to radiation attenuation properties. The steel-encased concrete wall therefore provides access and maintainability to the sample skid, gamma detector, and other PASS equipment while affording the equivalent radiation protection of a thicker (pure) concrete wall.

The wall was observed by the inspector to be in conformance with as-built design drawings.

14.4 Conclusion

The alleged wall was never underdesigned and, while seismically designed as Category 1, did have 1-inch thick steel plates added to either side for radiation protection and equipment access considerations.

The design of the wall was not "improper", and has no relation to the S&W seismic load combination computer code problem referred to by the alleger.

15. Allegation No. 33 - E&DCR Verbal Program

While employed onsite from 1974-77, the alleger stated that Engineering and Design Coordination Reports (E&DCR's) would have to be issued before design changes could be made and work continued. Upon his return to the site in 1982, a "new system to speed up the job was in place", involving a phone authorization for a design change. The "verbal" was to be followed by an approved E&DCR; the alleger questioned whether the "paperwork ever caught up", and whether the verbal authorization was followed-through by S&W/LILCO engineering with an approved design document.

15.1 Scope

Cognizant engineers with the S&W Site Engineering Office, responsible for implementing the E&DCR verbal authorization program for support/hanger work, were interviewed.

The verbal program was the subject of allegation nos. 25 and 34, which were addressed in NRC Inspection 50-322/83-34 and found to be unsubstantiated. That inspection involved a review of approximately 10% of the total 14,000 verbal authorizations, generated since the beginning of the program in early 1978, for the most extensive systems in the containment (B21 and E11, or reactor and RHR systems). The "paperwork catching up" is assumed to mean incorporation of a verbal into an E&DCR, or rejection of a verbal at the E&DCR stage.

The ultimate end to paperwork catching up with as-built systems is the incorporation of all E&DCR's into their affected design documentation. This inspection went one step further in assessing that status, and found that other NRC inspections have identified weaknesses (open items 82-04-14 and 84-20-01) with regard to the timely incorporation of E&DCR's into pertinent drawings and documents. The status of the licensee's efforts to resolve that item was reviewed.

15.2 References

-- NRC Inspection Report Nos. 50-322:
82-04, Appendix C, issued 5/12/82
83-10, Detail 2.2.3, issued
84-14, pp 31-34, issued 7/16/84
83-34, Details 1.3.3 and 1.3.6, issued 12/21/84
84-20, Detail 2.1 (p3); issued 6/15/84

-- SEO Interoffice Memo No. 55B, 1/11/80

-- S&W Engineering Assurance Division Trip Report (S. Morss and C. Walters), Jan 10-11, 1980

15.3 Findings

Refer to NRC Inspection Report 50-322/83-34, Details 2.2.3 (p13) and 2.3.6 (p21) which address the inspection of the verbal program, in response to allegation numbers 25 and 34.

The open NRC inspection items 82-04-14 and 84-20-01 are more significant findings as to the measure of paper work catching up with as-built design. This item is being resolved by the licensee, and will be a proposed license condition.

15.4 Conclusion

The verbal program was found previously in Inspection 83-34 to be successful and efficient. The verbal authorization was required by procedure to be either: 1) incorporated into an E&DCR, or 2) dispositioned as a nonconformance, within 90 days. A greater than 98% success rate (acceptable verbal) was experienced with this program, and all outstanding verbals had been resolved at the time of this inspection.

16. Allegation No.35 - Tight Work Space

A contractor was alleged to have no personnel small enough to perform work in or about the reactor vessel in 1982. The contractor allegedly "borrowed a small, slender man "from another (unspecified) company to do his (unspecified) work. The alleger's stated concern was for accessibility to that area after an accident.

16.1 Scope

Interview of cognizant FQC and Courter personnel resulted in the identification of the small worker(s) and specific work involved.

16.2 References

- Reactor Controls Inc. (RCI), QC Data Sheets for 4 Supports (identified as Support #1 on SM009, Sheet 4 of 6)

16.3 Findings

RCI Support #1 is actually four hangers, each at an equally-spaced (90 degree) quadrant off the pedestal wall and underneath the reactor vessel. The hangers are linear supports for CRD lines and a drain line off of the bottom of the vessel. Welding was performed in October 1982 by a "small" person borrowed from either UNICO Construction or Courter & Company. The inspector observed the location of these supports, and confirmed the difficulty in moving about in that location.

16.4 Conclusion

The area in which these supports are mounted is difficult to move about in, and is "close quarters". It is acccurately described to be a tight work space, but would not be required to be accessible either during normal operation or following an accident, due to severe radiological constraints.

17. Allegation No. 36 - Spent Fuel Pool Column Loads

It was alleged that, as an "after-thought" to design, LILCo decided that there was no place to send spent fuel, and so the spent fuel pool was added subsequent to original design and after Reactor Building construction was started. The alleger referred to another allegation (see detail 11, Allegation No. 20) regarding the over-loading of Reactor Building columns, citing the addition of the spent fuel pool as a major reason for the alleged overload. The alleger questioned the "margin of error", presumably for the expected versus design maximum loads on the Reactor Building columns.

17.1 Scope

Stone & Webster's Lead Structural Engineer for Shoreham explained the means of support provided for the spent fuel pool by Reactor Building structural members. Detail 11 of this report discusses Reactor Building column loads. Structural design margin for the Reactor Building columns was determined by comparing actual expected shear stresses (principally from attached supports) versus their ACI Code allowable values.

17.2 References

-- LILCo Dwg. No. M10005-12 (S&W Dwg. FM-1F-12); Machine Location Reactor Building, Section 2-2

17.3 Findings

The columns in the Reactor Building do not provide support for the spent fuel pool or associated pool loads. The pool is located at the outer radius or periphery of secondary containment, between building elevations 137' - 175'9", and was considered in the original design of the plant. The pool is totally supported by two, 20-foot deep, horizontal concrete girders which tie-into and span (over 130 ft.) four diametrically-located secondary wall pilasters. The pilasters are integrally part of the 2-foot thick secondary containment (or Reactor Building) wall. The twelve Reactor Building columns vertically span a height of over 100 feet, ending at upper building elevation 112'9", which is still more than 25 feet below the spent fuel pool floor liner. Pool loads are not taken up by these columns, but rather, are transferred via the girders to the outer building wall.

Regarding the margin of error, the following table is a comparison of the maximum expected (calculated) shear loads against ACI Code-allowable values, based on a 110 psi shear stress limit. The corresponding design margin is defined as design limit over expected. Four findings should be considered along with the table:

-- None of the Reactor Building columns have been loaded to their maximum expected shear stress, nor are they reasonably expected to be. This load is the "worst-case" expected, and is a combination of static (dead weight) and dynamic (Mark II transients, earthquake, etc.) contributions. Therefore, once a support is attached to a column, the as-loaded column experiences a shear that is only due to dead weight, which is a fraction (albeit large) of the total expected.

-- The actual calculated values are "intermediate" column shears from those various pipe support loads which are between floor slab elevations. The shear stresses at the slab-column junctions are transferred to the secondary (outer Reactor Building) wall, similar to the spent fuel pool.

-- To give an idea for Code conservatism, the theoretical maximum allowable shear stress of 110 psi is based on unreinforced concrete of 3000 psi compressive strength. Actual column concrete (1) exceeds the 3000 psi value, and (2) is heavily-reinforced. The 110 psi value would be much higher if actual strength were considered and rebar was not neglected. Therefore, this stress represents a design limit; actual material limits are such that a column would not be expected to fail in shear until stressed beyond 200 thousand pounds-force.

TABLE 17.3
Shear Capacity of
Reactor Building Columns

Column	Size	Max. Shear Load (kips)		Fraction of	Design
Number	(inches)	Allowable[b]	Expected[a]	Allowable (%)	Margin[c]
1	44 x 44	213	37.0	17.4	5.8
2	41 x 44	198	77.0	38.9	2.6
3	44 x 44	213	34.7	16.3	6.1
4	56 x 56	345	70.0	20.3	4.9
5	41 x 44	198	83.3	42.0	2.4
6	56 x 56	345	18.0	5.2	19.2
7	42 x 48	222	19.0	8.6	11.7
8	41 x 44	198	12.0	6.1	16.5
9	42 x 42	194	21.2	10.9	9.2
10	50 x 50	275	49.3	17.9	5.6
11	41 x 44	198	48.5	24.5	4.1
12	50 x 50	275	65.0	23.6	4.2

Notes:
a. S&W Calculation C16-2434 dated 3/2/82
b. 110 psi shear stress based on ACI equation 11.3.1.1
c. Ratio of maximum Code-allowable to actual expected

17.4 Conclusions

The spent fuel pool was considered as part of the plant's original design. The pool is not structurally supported by the Reactor Building columns; its load is transferred via two large span girders to the outer Reactor Building wall.

The average margin between actual shear expected versus the design limit (Code allowable) is over a factor of 8 for all twelve columns. The average column never exceeds the maximum Code-allowable shear load by more than 20%. Therefore, there is an unusually large amount of "margin for error", or design margin, engineered into these columns.

18. Allegation No. 38 - Sponges Left In Piping

This allegation was not originally provided in the March 9, 1983 letter, but was verbally described during the July 13, 1983 interview by the alleger. When startup flushing of a piping system was performed, the alleger stated that sponges were being found in the pipe. The sponges were allegedly used as "purge dams" during shop welding, and because of alleged inadequate off-site QC, were left in the pipe and shipped in that condition. The specific piping system(s), locations and timeframe when the alleged conditions were found was not provided by the alleger, since the information was received third-hand and alleged to occur during the period 1977-1982 when the alleger was not employed onsite.

18.1 Scope

Cognizant LILCo FQC, Startup and UNICO construction personnel were interviewed, as were Courter & Co. representatives.

One Courter Supervisor remembered an instance, late in 1977 or early 1978, where a sponge was supposedly found during a cleanliness inspection of the G41-Fuel Pool Cooling system. The incident could not be recalled by any of personnel interviewed above, nor was any record of it found. Approved flush procedures for that system, and for RHR, were reviewed for record of any sponge or other foreign material found during the flushing operations in preparation for preoperational testing.

The shop-fabricated pipe vendor-Dravo (Marietta, Ohio)- was contacted by phone on October 21, 1983, to discuss their use of purge dam material and final QC inspection at the shop.

18.2 References

- Courter General Welding Procedure for ASME III Piping - Specification SH1-056, NW-056-100, Rev. 3 (4/11/80); Sec. 4.5 and 4.6, Purge Dam Materials and Adhesives
- Dravo Corp. P.O. No. 310475, Specification SH1-24, Rev. 4, 5/11/83; Shop Fabricated Piping in accordance with ASME Section III, NA-3250 (7/1/77)
- Courter & Co. QA Procedure (QAP) 6.2, Field Fabrication and Installation of Piping Systems, Rev. 2, 10/19/82
- LILCO Preoperational Test Procedure CF 707.001-1, Fuel Pool Cooling and Cleanup (G41A&B); System Flush Procedure, approved 7/13/81; including: (a) Repair/Rework G-41-53 (b) Repair/Rework G-41-68

-- LILCO Preoperational Test Procedure CF121.001, RHR System (E11); System Flush Procedure, approved 3/24/81

-- NRC Inspection Report Nos. 50-322: 77-21, Detail 4 (p6); issued 11/21/77; 80-06, Detail 2b (p5); issued 5/19/80

-- IE Information Notice 81-07 issued 3/16/81; Potential Problem with Water-Soluble Purge Dam Materials Used During Inert Gas Welding

18.3 Findings

The field-welded piping installed by Courter and Co. involved an approximate 500 pieces, most of which (est. 70%) was small bore. These spools and field fabricated pieces were welded in accordance with Specification-056 which did not explicitly allow for sponge material as a removable purge dam; rather, Section 4.5.A specified 1) expandable or flexible plugs, bags or balloons, 2) plexiglass or plywood, and 3) kiln-dried wood dams. Courter personnel stated that the most-commonly used removable dam was a blowup bag, and that sponges were not used.

Dravo QA representatives were contacted on October 21, 1983. Shop-welded pieces were governed by Specification-024, and the commonly used dam material was a rubber-covered wooden disc. Dravo personnel stated that sponges were not used, since they would have a difficult time in getting them to properly seal the pipe during inert gas welding. There were cleanliness inspections and other activities, following the welding of stainless steel pipe, which made the possibility of leaving removable purge dam material (within the pipe) remote. These piping subassemblies would be receipt-inspected by LILCO FQA upon site arrival. A final "square-up" inspection by LILCO SQA, prior to pressure testing and at the completion of installation, was a typical holdpoint on Component Checklists for a weld. A sampling review of weld packages for Fuel Pool Cooling and RHR systems piping found no mention or evidence of sponges found. A Courter & Co. supervisor did recollect an instance, guessing it to be during late 1977 - early 1978, of a rumor that a sponge had been found in G41-Fuel Pool Cooling and Cleanup piping, during a cleanliness inspection. The inspector could find no other individuals with knowledge of that alleged instance, nor any record or evidence thereof.

Startup flush procedures, approved by the LILCO Joint Test Group (JTG), were reviewed for evidence of foreign objects found during flushing; none was found. Further, no problems were experienced during the preoperational testing of either E11 or G41 systems which could be attributable to flow blockage that is indicative of trapped foreign objects, such as sponges.

Finally, NRC Inspection 77-21, conducted during October 1977, opened item 77-21-01 concerning OQA control of special processes. Specifically, QA coverage of the control of removal/installation of parts removed for flushing operations was reviewed, including cleanliness considerations. Flush procedures were required to have OQA witness at various points, and the removal of any potential foreign objects would be documented with repair/rework procedures and Appendix 4A of the Startup Manual entitled "Maintaining Cleanliness During Repair/ Rework of Flushed Systems". The item was found to be acceptably resolved, and was closed in Inspection 80-06.

18.4 Conclusion

Dravo shop weld practices were found to not use sponge material as a purge dam for inert gas welding of stainless steel large-bore spool pieces. Similarly, field-welding of piping at Shoreham did not typically use sponge material, and no evidence of sponges was found during cleanliness inspections of piping installations. A review of flushing records for RHR and Spent Fuel Cooling systems found no record of sponge material left in these systems.

The NRC readiness assessment team inspection 83-02 involved daily plant tours and extensive records reviews by eight inspectors, accounting for over 500 inspector hours. No problems with foreign material left in piping systems were identified.

While one Courter supervisor did remember an instance where a sponge was found in Spent Fuel Cooling piping, no record could be found of that instance, and the flushing and preoperational testing of this system was successfully completed with no evidence of flow blockage or foreign material.

19. Allegation No. 39 - Copper-Nickel Pipe Joints

This was an additional item, not in the March 9, 1983 letter, but presented during the July 13, 1983 interview at NRC Region I. A problem was alleged with the first threaded-joint off of the Main Service Water system (P-41) header, a small-bore brazed connection, which "keeps snapping off". The cause and timeframe for the alleged problem was unknown, and the information was received by the alleger over the phone, second (or third)-hand. The alleged problem was ot operational (i.e. not cuased by dynamic or flow conditions), and the alleger's stated concern was for a "loss of cooling". The alleger implied that the problem might have been caused by "people stepping on" the connection, and that the utility was aware of the problem but incorrectly addressing it.

19.1 Scope

Cognizant personnel were interviewed, pertinent records for Service Water reviewed, and related problems (including NRC inspection coverage thereof) with copper-nickel piping were researched. Dravo shop representatives were contacted by phone. The inspector walked-down the 20-inch P-41 header inside the Reactor Building, and observed the condition of 3/4-inch vents and drains.

19.2 References

- Dravo Piping Isometrics E-2821, IC-130 thru 134; As-Built Drawings 40.41 - 003 P, 031R, 032U, 004Q and 039Q for Service Water (P-41) System.
- Service Water Flow Diagram FM-47A-13; Area D-6
- Specification SH1-056, ASME III, Class 2&3 Piping
- Pressure Test Package P-41-14, (hydrotest performed November 1978)
- Preoperational Test Procedure PT122.001-2, Reactor Building Service Water, approved 1/1/83.
- -- Unresolved Item 82-04-10, Carbon Steel Bolting

  (1) NRC Inspection Report Nos. 50-322:
  82-04, Detail 4.1.3 (p20); issued 5/12/82
  83-01, Detail 2.4 (p3); issued 2/8/83

  (2) LILCo letter (5NRC-743) to NRC, 7/28/82

  (3) NRC letter to LILCo dated 11/4/82
- -- Investigation Report No. 50-322/81-21, conducted Oct.-Dec. 1981; issued 1/14/82

-- Construction Deficiency Report 77-05, (10/6/82), Incomplete Penetration in Copper-Nickel Welds on Service Water Piping
(1) NRC Inspection Report Nos. 50-322:
77-23, Detail 8a (p.8), issued 12/23/77
79-03, Detail 4b(p6), issued 3/23/79

(2) LILCo letters to NRC Region I dated 11/16/77, 2/24/78, 7/6/78

(3) NRC letters to LILCo dated 12/22/77, 4/4/78, 7/27/78

-- NRC Inspection Report 50-322/83-07, Detail 3.2 (p4); issued 5/24/83

-- Unresolved Item 82-02-8, Salt Water Leakage, NRC Inspection Report 50-322; 82-02, Detail 7 (p7), issued 2/2/83; and
83-17, Detail 2.1.5 (p3), issued 7/8/83.

19.3 Findings

During normal operation, the Service Water (P41) system provides cooling water to both safety and non-safety related loads. The RBCLCW heat exchangers, the drywell cooling booster heat exchangers, and the reactor building air conditioning chilled water condensers are the principal Reactor Building loads. The main chilled water condensers, the turbine building closed loop cooling water heat exchangers and other miscellaneous loads comprise the 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 under accident conditions. The system additionally provides cooling water to the emergency diesel engine coolers, emergency makeup water to the spent fuel pool, and emergency cooling water to the ultimate cooling connection (to RHR).

The piping is ASME Section III, Class 3 (safety-related and seismically qualified) and is designed for 125 psig pressure. Normal system pressure is expected to be 70 psig, and normal flow in either 20-inch main header (loop A or B) entering the Reactor Building is on the order of 10,000 gpm. The major-safety related heat load serviced by the system, in the Reactor Building, are the RHR heat exchangers (8,000 gpm design flow to each required). The emergency makeup branches off of the main header in Reactor building, to the fuel pool and "ultimate" cooling tie-in with RHR system, are isolated with two locked-closed valves in series, and would not be normally used.

There were two documented problems with copper-nickel piping in the Service Water system, but neither resembled the alleged condition of the threaded joint snapping off. One (open inspection item 82-04-10) involved the corrosion of carbon steel bolts and external fasteners used on the copper-nickel piping which exhibited general corrosion and some galvanic activity. The licensee proposed a solution that replaced and encapsulated (with insulation) these components; and which was acceptable and therefore closed. The other issue was a construction deficiency reported to Region I on October 7, 1977 regarding incomplete field-weld penetration defects as well as shop-weld deficiencies with the 90-10 copper nickel service water piping. All accessible field welds (approximately 129, open butt root type) were visually inspected, and all shop welds on-site (918 total) were visually inspected.

The inaccessible field and shop welds, in piping already imbedded in concrete in the diesel rooms, was retired in-place and replaced. Unacceptable welds were repaired and weld procedures were revised to correct this problem. The shop welds were subsequently determined to be superficial conditions and in full compliance with Code requirements. The deficiency was therefore found to be acceptably resolved.

Another inspection item (82-02-08) identified a problem with the location and flushing of a 3/4-inch drain connection off of the copper-nickel Service Water emergency supply piping ("ultimate cooling") to the spent fuel pool. Two series isolation valves, normally locked-closed, leaked by and flooded the spent fuel pool with 2,000 gallons of salt water via the Service Water cross-connect on August 17, 1981. The normally open tell-tale drain line was relocated, and a preventative maintenance activity initiated to regularly check for leakage, with a pressure indicator added between the block valves to accomplish the same. The inspection additionally verified that all vent and drain connections are routinely flushed, and the item was closed.

An NRC investigation was conducted at Shoreham during October thru December 1981 in response to an allegation of two cracked welds in the 20-inch diameter copper-nickel supply piping between the Screenwell and Reactor Building. The headers are encased in concrete and buried; the alleged cracks were located over 150 feet away from the headers' penetration into the Reactor Building. The investigation was documented in Report No. 50-322/81-21 which was issued on January 14, 1982, and concluded that the weld defects which were encountered were properly identified and documented by QA inspections, and were subsequently repaired and successfully hydrotested. The investigation was consequently closed. Many of the investigative findings, particularly the conduct of successful hydrotesting on the Service Water piping during 1978-79, are relevant to (and therefore used as supportive bases for) the resolution of the alleged problem with the small-bore threaded joint connections.

Preoperational testing of the Service Water system commenced with turnover of the system to LILCo Startup in July 1981. Preoperational testing was successfully performed, and NRC inspection coverage of that testing was provided. Preoperational Test Procedure PT122.001-2 for Reactor Building Service Water was approved by the LILCo Joint Test Group on January 1, 1983. NRC review of that completed test was documented in Report 83-07 and found no discrepancies noted in the procedure with respect to test results, changes and exceptions, deficiencies, acceptance criteria, and final restoration of the system to normal following the test. This system has been run successfully, since, with no problems noted or known with regard to small (3/4-inch vent, drain, instrument) connections leaking or falling-off of the main 20-inch headers inside the Reactor Building. The inspector verified operation of (flow in) portions of this system; no related problems were observed.

It was observed that many of the small connections off of the header, which are now encapsulated by mirror insulation and not in high traffic areas, would have been more accessible during construction and therefore susceptible to people stepping on them.

The P-41 copper-nickel 20-inch supply headers penetrate the Reactor Building at lower elevation 12', and both lines run vertically up (along the inner wall) to elevation 25', where each loop is routed (in opposite directions) horizontally along the wall to its respective RHR heat exchanger (and other lesser heat loads). Based on a study of the piping isometrics and a walkdown of the lines, there are estimated to be no more than 25 small (less than 2½ inches diameter) connections off of each loop of the 20-inch header inside Reactor Building. The connections nearest the penetration into the building were inspected and found to be intact and not leaking. These included the following 3/4 inch vents and/or drains:

3/4" - WS-416-158-4
3/4" - WS-801-158-3
3/4" - WS-800-158-3
3/4" - WS-425-158-3

Dravo (Marietta, Ohio) was contacted and provided the information that a number of silver-brazed bronze outlet fittings termed "brazolets" (manufactured by Bonney-Forge) may have been the connections described by the alleger. Had these been used, they would have been brazed to the copper-nickel header and would accept a threaded outlet connection. However, these were not used by Dravo; they welded DB-1 fittings of their own design to the 20-inch header which is a brass boss that is welded (not brazed) to the pipe. No problems were noted by Dravo with these fittings provided for Shoreham.

19.4 Conclusion

No problem with the small-bore, 3/4-inch brazed joint connections off of the 20-inch Service Water sytem copper-nickel header inside the Reactor building could be found. Most of the piping was already insulated at the time of the inspection, but an inspection of four typical small (3/4-inch vents and drains) brazolet connections, after the system had been preoperationally tested and run, showed no problem with the integrity of the fittings.

The licensee is unaware of any such problem, as alleged, and has therefore never had to "address" this alleged problem.

The significance of a 3/4-inch leak in the 20-inch Service Water header, at 10,000 gpm and 100 psig, would be relatively minor. Not only would cooling flow be unaffected, but the leak may not actually occur, since the possiblity exists that air would be educted into the pipe (rather than salt water out) in that situation.

20. Allegation No. 40 - RHR Support Overstress of Column 12

The alleger contacted NRC Region I by phone on July 18, 1983 and stated an additional concern not brought forth in either the March 9, 1983 letter or the July 13, 1983 interview. An RHR hanger (E11-095 or 096) was alleged to have overstressed Reactor Building Column-12 at elevation 63 ft. The hanger was alleged to be tied to the south face of the column.

20.1 Scope

Details 3, 11 and 17 of this report address RHR supports and column loads. In particular, RHR restraint PSR-054 and Column-12 are discussed in detail. Loads on Column-12 were reviewed and shown to be within design limits, with margin.

The support packages for RHR pipe supports 095 and 096 were reviewed.

20.2 References

-- RHR Strut PSST-095
   Bergen-Paterson Dwg. No. 1E11-PSST-095-10
   Piping Isometric 11IC13

-- RHR Hanger PSSH-096
   Bergen Paterson Dwg. No. 1E11-PSSH-096-7
   Piping Isometric 11IC14

20.3 Findings

As concluded in Detail 11 of this report, Column-12 is loaded to less than design limits, with margin.

RHR PSST-095 is actually a strut tied into the bottom-side of a concrete floor slab at Reactor Building el. 38', and in the vicinity of Column-3, nine feet away. This strut supports a 20-inch RHR line (WR210) run at elevation 27', approximately ten feet below the ceiling.

RHR PSSH-096 is actually a spring hanger, tied into an overhead structural steel beam at Reactor Building el. 29', and in the vicinity of Column-4, seven feet away. This hanger supports the same RHR line as does PSST-095.

20.4 Conclusion

Neither of the alleged supports is near Column-12. The support which is closest to matching the alleged location is RHR restraint PSR-054, but this support is not tied to the south face of that column.

As discussed in Detail 11 of this report - while it is conceivable that a support such as PSR-054 (more than 10 formal design changes, 20 verbal authorizations associated, large and extensively re-designed) may have approached its maximum Code allowable shear stress of 275 kips at some point in the chronology of its proposed design, current loads on this column are within design limits by a factor of 4 or more.

Column 12 is not (nor has it ever been) overstressed, and it has no support connections on its south face. RHR supports 095 and 096 are nowhere near Column-12.

E. Action on Previous Inspection Findings

(Closed) Unresolved Item 80-18-03

This item addressed preoperational functional testing of mechanical snubbers identified in draft Technical Specification (TS) Table 3.7.5-1, to demonstrate initial operability prior to fuel load and the capability to perform required TS surveillance during operation.

NRC Inspection Report 322/83-10 examined records of vendor factory functional testing and initial receipt stroking by the licensee. Adequate protection was observed following installation. A sampling (minimum of 10%) of drywell snubbers were being stroked in-place at the time of Report 83-10; no failures had been experienced at that time. However, the sample was committed to be increased to include a population representative of all sizes, as well as snubbers outside primary containment including some non-safety related snubbers.

An additional accessibility review was performed for 200 more snubbers, of all sizes, throughout both primary and the Reactor Building. Seventeen were identified as difficult to remove; one of these was located near the CRD insert/withdrawal lines, and was classified questionable as far as removal or access. Another of the 17 difficult snubbers was located in the steam tunnel. All 17 were in potentially high radiation areas, and ALARA studies were undertaken to estimate total man-rem exposure for surveillance on each.

The ALARA study concluded that all 17 snubbers could be accessed for removal, replacement, or Technical Specification surveillance. The work was estimated by considering 5 phases: mobilization, disassembly, stroking, re-assembly and demobilization. The ALARA study was summarized in a September 12, 1983 memorandum from the Radiation Protection Supervisor; it stated that snubber removal did not present any "remarkable radiation concerns", and that employment of standard ALARA techniques would significantly reduce the projected occupational exposures (which were less than 1 man-rem in most cases).

Finally, the Site Engineering Office (SEO) prepared an estimate of the cost expected to be incurred in re-locating a snubber. Engineering, alone, would account for 1500-2000 manhours or an estimated cost on the order of 100 thousand dollars per snubber. To close this item, SEO then undertook a feasibility/cost evaluation of the four most difficult (ALARA or accessibility) snubbers, identified as capable of "scalloping" or cutting away of structural steel, to facilitate removal. Three were judged as impossible to be relocated. Based on the snubber sampling, ALARA studies and cost evaluation, and upon the conclusion that none of the snubbers are considered inaccessible or significant radiological problems, this item is considered closed.

F. Exit Interviews

Exit interviews were held on a number of occasions, during the course of this inspection, to discuss the findings summarized herein.