U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-322/83-41

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: December 15-23, 1983

Inspectors: Gene Kelly — July 5, 1984
E. M. Kelly, Project Engineer — date

Approved by: E. C. McCabe — 7/10/84
E. C. McCabe, Chief, Projects Section 1C — date

Summary: December 15-23, 1983 (25 hours)

Special inspection by one region-based inspector of four allegations related to the design, construction and testing of the Shoreham Nuclear Power Station. The allegations were part of a number of allegations made in letters addressed to NRC Region I during the period January-April, 1983, as clarified in an interview with the alleger at Region I office on July 13, 1983.

Alleged were (1) an inadequately sized suppression pool, (2) encouragement to shortcut QC hold-points during construction, (3) missing as-built tubing diagrams for channels used to check the weld integrity of the suppression pool liner floor plates, and (4) excessive leakage characteristics of containment isolation valves as experienced during the reactor pressure vessel cold hydrostatic test conducted on September 21-22, 1979.

None of these allegations were substantiated.

1. Summary

1.1 Background

A number of allegations were initially 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 formally transmitted to Region I (addressed to E. Greenman, Chief, Projects Branch No. 1) in an April 21, 1983 letter. The allegations were made by a steamfitter, 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 of forty allegations was compiled. This special inspection was conducted to determine the validity of four of these allegations.

1.2 Allegations Inspected

The following allegations were inspected:

1.2.1 Allegation No. 4 - Lost as-built information for tubing connected to suppression pool liner floor plate weld test channels and grout caps, such that the liner welds will not be testable at some later date.

1.2.2 Allegation No. 7 - Excessive valve leakage was experienced during the reactor vessel cold hydrostatic test, which could compromise containment integrity. Further, containment isolation valve allowable leakage values may have been revised to accomodate the excessive leakage experienced as part of the vessel hydro.

1.2.3 Allegation No. 21 - When the Shoreham generating capacity was increased by 50%, the size of the suppression pool was not changed, such that its design margin in accomodating post-accident pool swell phenomena is questionable.

1.2.4 Allegation No. 28 - The alleger was asked by his supervisor to shortcut certain construction inspection hold-points.

1.3 Conclusions

None of the allegations listed in paragraph 1.2, and investigated as part of this inspection, were found to be substantiated.

1.3.1 Suppression Pool Liner Test Channel Location

The tubing connected to grout caps and leak test channels for floor plate welds on the suppression pool liner had its field-run, as-built routing depicted on Courter and Company Drawing Nos. LKS-001 and 002. These as-built diagrammatic drawings of the completed leak test tubing installation were prepared as required by Engineering and Design Coordination Report (E&DCR) F-6347S, approved on January 31, 1977. However, the weld (and grout cap) integrity pressure tests, which were initially done in late 1973 - early 1974 and documented by Pressure Test Reports, were intended as a one-time construction test of the primary containment floor plate. Retesting was, and is, not required. The tubing which was field-run was a non-safety related (QA Category III) installation.

At the conclusion of leak channel and grout cap tests, the tubing was plugged at the test connections, abandoned in-place, and is currently practically inaccessible. A 12-inch reinforced concrete cover slab was later poured over top of the liner floor plate. The cover slab provides protection for the liner as well as bearing for various internal suppression pool structures, and is in turn covered by suppression pool water to a depth of 18 feet.

1.3.2 Containment Isolation Valve Leakage

No record of unusual or unacceptable leakage associated with containment isolation valves during conduct of the reactor vessel and main steam line hydrostatic test ("cold hydro") was noted in preoperational test procedure CS136.001-1. The cold hydro was successfully conducted on September 21-23, 1979 in accordance with the ASME Boiler and Pressure Vessel Code, to verify the integrity of the reactor vessel, its connecting piping and welds, and portions of the main steam lines. Visual inspection for leakage at all welds, joints, connections, mechanical fittings and regions of high stress intensity was performed, with no problems noted.

Excluded from this leakage, as allowed by the Code, was minor leakage from equipment seals, valve packing and gasketed joints as long as it neither masked the detection of unacceptable leakage at surfaces of interest, nor exceeded the capacity of the pressure source to achieve and maintain required hydrotest pressure. Two of three available hydrotest pumps were used; the only problem encountered during the test, as highlighted by the approved test procedure, involved a temporary delay to replace a ruptured suction hose on one of the pumps. The cold hydro was characterized as efficient, concise and problem-free by the licensee's test engineers. The hydro was witnessed as part of NRC Inspection 50-322/79-15 and found to be in accordance with approved procedures, with test acceptance criteria properly met. Unquantified leakage was apparently experienced with steam line valve packing and/or seats near the feed pump turbine, downstream and outside of containment (in the turbine-side steam tunnel), but this leakage was not a principal concern during the test nor did it mask the detection of unacceptable leakage or exceed the hydrotest pump capacity (0-40 gpm per pump) to maintain test pressure. Therefore, the number of hydrotest pumps required, and the presence of any acceptable valve leakage, were not major problems for the Shoreham cold hydro, nor are they a principal concern during a Code hydrotest.

The meaningful measure of containment isolation valve (CIV) leaktightness is the Type C local leak rate testing (LLRT) performed as required by 10 CFR Part 50, Appendix J. The initial Type C LLRT for the 215 CIV's at Shoreham was performed during the period March 1982 - February 1983 by methods described in preoperational test procedure PT654.003-1. Such tests were witnessed as part of numerous NRC inspections (e.g. Report Nos. 50-322/82-10, 82-17 and 82-32), all of which ascertained compliance with licensee committments and Appendix J requirements. The Type A primary containment preoperational integrated leak rate test (CILRT) was successfully performed on December 9-10, 1982, and witnessed by NRC as part of Inspection 50-322/82-32. The overall leakage was determined to be approximately 0.27% per day (by weight of containment air and at peak calculated accident pressure) - within the Appendix J acceptance criterion of 0.375%, and well within the design/maximum value of 0.5% assumed in accident analyses. A Summary Technical Report providing the results of the preoperational CILRT and LLRT was submitted to NRC by LILCo in a letter dated March 10, 1983.

The main steam isolation valves (MSIV's) are the only CIV's with individual leak rate limits (11.5 SCFH), and this is excluded from the summation for Shoreham's LLRT and CILRT since an exemption to Appendix J was granted by NRC. Justification for that exemption is documented in NUREG-0420, the Shoreham Safety Evaluation Report. All eight MSIV's were initially tested in Fall 1981 at Shoreham, and problems were encountered in meeting the 11.5 SCFH limit on four of the valves. All MSIV's were eventually reburbished and successfully passed their LLRT during the period October-November 1982.

Finally, Appendix J and the plant Technical Specifications require periodic verification of primary containment leaktight integrity by regular surveillance of CIV's. Type C tests will be conducted during each reactor shutdown for refueling, or at least every 2 years, and the performance of a Type A CILRT will be performed three times, at approximately equal intervals, during each 10-year plant service period.

1.3.3 Suppression Pool Size

At the request of the Shoreham Atomic Safety and Licensing Board (ASLB), the licensee submitted a response dated June 25, 1982 regarding primary containment and pressure suppression pool sizes (Item 1, initial transcript reference 1157). Containment design was subsequently addressed as part of contention SC-21 during the ASLB hearings for Shoreham. The Board found the margins inherent in Shoreham's Mark II design to be adequate, and concluded in its Partial Initial Decision issued on September 21, 1983 that LILCo had met its burden of proof on all aspects of this contention, with the exception of one concern for the operation of the RHR heat exchangers in the steam condensation mode (which is not related to this allegation). The Board's requisite finding was that: "...there's reasonable assurance that the Shoreham containment is designed with adequate conservatism to protect the public health and safety.."

The containment's ability to accomodate hydrodynamic loads, with margin, following a design basis accident, is documented in the Shoreham Design Assessment Report (DAR). The DAR was reviewed and approved by the NRC staff, as stated in the Safety Evaluation Report (SER). Design modifications to internal suppression pool structures were described in the DAR, evaluated and approved in the SER, and reviewed as part of NRC Inspection 50-322/83-34.

While comparison between the Shoreham and Zimmer containment design does show some differences, the size of the suppression pool water volume is not the only parameter to be considered when assessing pool swell phenomena and its effects. The Shoreham total containment volume is 10% greater, and design pressure 3 psi higher, than that at Zimmer. However, while the more severe blowdown from the larger Shoreham BWR-4 primary system serves to "drive" the suppression pool harder, other factors (reduced downcomer submergence, shallower pool depth, larger wetwell airspace) tend to mitigate the blowdown. While the pool represents a relatively small surge volume, it also represents a large active heat removal system. The lower initial pool temperature and Service Water cooling to the RHR heat exchangers, both parameters attributable to the cool Long Island Sound climate, along with a 15% greater RHR heat exchanger overall heat transfer capability, serve to enhance the Shoreham design's capability to perform long-term accident suppression pool cooling and to limit bulk pool temperature.

Finally, the Shoreham containment internal design volumes were enlarged at the time a decision was made in 1968 to increase plant generating capacity (by 52%) from 540 to 820 MWe. This included a 20,000 cubic foot (32%) increase in pool water volume, a 6-foot increase in pool depth (12 to 18 ft.), and almost double the number of downcomer vents (from 45 to 88) in order to accomodate the increased blowdown expected from the larger reactor.

1.3.4 Elimination of Construction Inspection Hold-Points

Telephone interviews, with the only two Courter and Company managers who served as the alleger's supervisors during his tenure as a supervision of crafts, were held in the presence of licensee representatives.

While each former supervisor stated that productivity was an important consideration which warranted attempts to "streamline" the schedule, such improvement would not be considered at the sacrifice of quality. Labor over-run and schedule slippage were part of Courter's responsibility, and end productivity was a practical priority; however, Courter was held equally responsible for efficient and safe construction which was within ASME Code requirements. While it would not have been unusual to have held discussions with the alleger in his one-time role as supervisor of Courter crafts regarding improvement of job productivity, each former supervisor stated that, while neither could recall specific discussions with the alleger, each would "not have been surprised" if they had asked the alleger his opinion on what steps could be taken to streamline a particular job's schedule.

However, as stated by each of the alleger's former supervisors during the phone interview, no verbal or written direction was ever given by them to the alleger to eliminate or shortcut a QC inspection hold-point. Both of these former supervisors are currently employed by Courter and Company, but at job site locations different than Shoreham.

Interviews with licensee representatives who performed FQC inspections (during that time-frame when the alleger was a supervisor) identified no instances of any deliberate attempts by Courter supervision to bypass a required inspection checklist item. During the height of construction activity, FQC manpower was stated as being a typical "beef" on the job - QC inspector response time was improved at one point by instituting the "area concept" of stationing a QC inspector in the building where work was underway, to minimize delays in work progress at the hold-point (e.g. pre-weld inspections). However, since Courter did not receive an ASME stamp (and hence did not require their own QC) until January 1978, which was after the alleger's tenure as a supervisor, Courter could not write their own nonconformance reports (NCR); Stone & Webster FQC would have had to generate such reports during the period of interest. Courter welder training sessions were stated by the alleger's former supervisors as accordingly underlining either the observation of a hold-point or the obtaining of an NCR, for proper progress of construction.

2. Details

2.1 Principals Contacted

Long Island Lighting Company (LILCO)

J. Smith, Manager, Nuclcar Support
R. Glazier, Field QA Engineer
G. Gisonda, Compliance Engineer
E. Nicholas, Field QA Section Supervisor
D. Terry, Chief Maintenance Engineer
R. Lawrence, Startup Engineer
J. Livingston, Senior Test Engineer
S. Aikens, Technical Support
G. Laurie, Projects Office
E. Stoudt, Field QA Engineer

Stone Webster Engineering Corporation (S&W)

*J. Metcalf, Power Engineer, EMD (Boston)
P. Baker, Lead Structural Engineer, SEO
R. Jaquinto, Head, Site Engineering Office, SEO
V. Mehta, Structural Engineer, SEO
*C. Malovrh, Lead Engineer, EMD (Boston)
W. Smith, Power Engineer, EMD
R. Wiesel, Lead Structural Engineer, EMD (Boston)
D. Misiaszek, Assistant Licensing Engineer

General Electric Company (GE)

J. Riley, Operations Manager
A. Ketchum, Test Engineer

Courter & Company

J. Arcuri, Project Manager
*J. Pecoraro, Vice President, Construction
*A. Czarnomski, Former Supervisor

U.S. Nuclear Regulatory Commission

C. Petrone, Resident Inspector, Shoreham
C. Anderson, Chief, Region I Plant Systems Section
*F. Eltawila, Containment Systems Branch, NRR
*W. Guildemond, Senior Resident Inspector, LaSalle
*P. Gywnn, Senior Resident Inspector, Zimmer

The inspector held discussions with other licensee and contractor personnel during the course of this inspection.

*Denotes telephone contact.

2.2 Reference Documents

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

- Shoreham's Mark II Design Assessment Report (DAR)
- Shoreham Final Safety Analysis Report (FSAR)
- NUREG-0420, Shoreham Safety Evaluation Report (SER)
- NUREG-0808, Mark II Containment Program Acceptance Criteria
- ASLB Partial Initial Design, LBP-83-57
- Shoreham Plant Technical Specifications (TS) - Proof and Review Copy

The references which follow were specific to the individual allegation inspected.

2.2.1 Allegation No. 4 - Pool Liner Test Channels

a. E&DCR-6347S, approved 1/31/77

b. Specification SH1-75, PDM PO-310103; Field Erection of Steel Plate Liner, 3/3/82

c. TS 3/4-6.1.5; Primary Containment Structural Integrity

d. Pressure Test Reports H.155, 157, 158; 3/8/74

e. DAR Section 7; Containment Liner Assessment

f. LILCo Dwg. M-10167-11; Liner Floor Details

g. Courter and Co. Dwg. Nos. LKS-001 and 002;
Reactor Building Concrete Slab - Leak Detection System, (11/9/79)
Primary Containment Liner - Grout Hole Capping (Rev. M)

h. PDM Dwg. HLI, 9-13-73; Containment Liner - Halide Leak Test Record

i. NRC Region I Inspection Report Nos. 50-322:
82-17, issued 6/22/82, Detail 4.2.2
82-15, issued 8/30/82, Detail 10
83-11, issued 5/27/83, Detail 3

2.2.2 Allegation No. 7 - Isolation Valve Leakage

a. Test Procedure CS136.001-1, (9/18/79); Reactor Pressure Vessel Hydrostatic Test

b. Test Procedure PT654.003-1, Test Summary - Primary Containment LRT-Type C; 3/7/83

c. LILCo Dwg. No. FM-83A, Reactor Pressure Vessel Boundary

d. Specification SH1-412, WO-80-48923, 8/1/83;
Technical Requirements for Pressure Testing Installed Piping,

e. Courter Disassembly/Reassembly Release Forms; VRB-100, VRB-143, VGW-15A-2

f. ASME Code, Sections III and XI, Subsection NB 2121

g. NRC Region I Inspection Report Nos. 50-322:
79-15, issued, 11/9/79;
82-10, issued, 5/24/82, Detail 3
82-17, issued 6/22/82, Details 3, 4 and 5
82-32, issued 1/18/83, Details 3 and 4

h. Code of Federal Regulations, 10 CFR Part 50,
Appendix J; Primary Reactor Containment Leakage Testing

i. LILCo Letter SNRC-856, Smith to Denton, 3/10/83;
Preoperational Integrated Leak Rate Test Summary

j. Main Steam Isolation Valve Maintenance Report,
Prepared by C. R. Clark, GE-I&SE, 12/82

k. LILCo Response to TMI Action Item III.D.1.1 of NUREG-0737,
Leakage Reduction and Control Program

l. Shoreham TS 3/4.6.1; Primary Containment Integrity

m. ANSI/ANS-56.8 - 1981, National Standard - Containment System Leakage Testing Requirements

n. NUREG-0420, April 1981, SER Section 6.2.5

2.2.3 Allegation No. 21 - Suppression Pool Size

a. LILCo Response to ASLB May 1982 Request, dated 6/25/82; Item No. 1, Primary Containment and Suppression Pool Sizes (Initial Transcript Reference 1157)

b. ASLB Partial Initial Decision (LBP-83-57), 9/21/83; Section II-F, Mark II Containment (SC Contention 21)

c. Shoreham FSAR Table 6.2.1-1, August 1983; Containment Design Summary

d. Zimmer FSAR Table 6.2-1, Jan. 1983, Containment Design Summary

e. LaSalle FSAR Table 6.2-1; Dec. 1982; Containment Design Parameters

f. DAR Section 6, Primary Structures Assessment, Subsections 6.5.2 and 3, Design Margins and Conclusions

g. DAR Appendix B, Containment Structure Design Margin

h. NRC Region I Inspection Report No. 50-322/83-34, issued 12/21/83, Details 2.2.1 and 2.2.4

i. Shoreham FSAR Section 6.2, August 1983; Containment Systems

h. NUREG-0420, SER Supplement 1, Sept. 1981, Section 6.2.1.8, Pool Dynamics

2.2.4 Allegation No. 28 - QC Holdpoints

There were no reference documents reviewed by the inspector which would apply to this allegation.

2.3 Bases for Findings

2.3.1 Suppression Pool Liner Test Channel Locations

The floor liner is comprised of ¼-inch thick carbon steel plates, arranged in a rectangular array and joined by continuous full penetration welds. The liner provides a vapor-tight/water-tight barrier for the primary containment structure, and is set on top of a 10-foot thick reinforced concrete foundation mat. The test channels were placed above each floor plate weld seam, and tracer gas-tested with halogen. At floor locations near the reactor vessel pedestal wall, small holes were drilled to grout underneath certain plate locations, in order to correct for proper floor bearing. The grout holes were later closed by welded plugs, which were in turn capped.

E&DCR F-6347S, approved on 1/31/77, provided UNICO Construction with information regarding the leak test tubing for the pool liner floor test channels. The tubing and fittings used were QA Category III (non-safety related), with specific routing to be determined in the field. The location of, and connection to, test plates was also clarified in this E&DCR and item 8 required that an as-built diagrammatic drawing be made of the completed installation. Courter drawings LKS-001 and 002 satisfy that requirement. The test channels and tubing were then required to be retested prior to the placement of the concrete cover slab over the pool liner floor.

The capped grout plugs and weld seam leak channels were connected by ¼-inch stainless steel tubing which was field-run to test plates in each of the four floor quadrants. Courter drawing numbers LKS-001 and 002 depict the specific routing of tubing from grout caps and weld channels to the 16 test connections at each test plate. At the conclusion of the halide leak testing, the tubing was plugged at the test plates and abandoned in-place, and is currently practically inaccessible. In May-June of 1977, a 12-inch concrete cover slab was poured over the liner floor, and provides protection as such. On top of the covering concrete is suppression pool water at a normal depth of 18 feet. Although as-built test area/tubing configuration is available which would enable additional weld integrity testing of the pool liner floor plate seams and grout hole plugs, retests are neither planned nor required.

The floor liner weld seam leak channels and grout caps, including connecting tubing and test plates, are all located within the primary containment structure. The liner serves only as a leak-tight barrier for the containment, and the containment pressure - retaining boundary at the pool floor is comprised by the ten-foot thick reinforced concrete foundation basemat (not the liner plate). The Shoreham Containment Structural Acceptance Test (SAT) was successfully performed during July 3-10, 1982, and verified the integrity of the primary containment by pressurization (with air) to 117 percent of design pressure. Virtually no cracking of containment concrete was evidenced, and strain/deflection measurements were found to be within the expected limits. The test was witnessed as part of NRC Inspection 82-15 and test results were reviewed and found to be acceptable in Inspection Report No. 83-11.

In conclusion, although no re-testing or re-use of either the floor or wall liner leak channel system is planned, Shoreham Technical Specification surveillance 4.6.1.5 will require a visual inspection of exposed accessible interior and exterior surfaces of primary containment, including the liner plate, as part of each Type A CILRT (to be conducted approximately every 3 years during plant operation). Further, a visual inspection of accessible liner test channel plugs will be performed at least once per 18 months. This requirement does not apply to inaccessible plugs, or plugs which are tack-welded in-place; therefore, the pool floor liner plate test connections will not require these inspections.

2.3.2 Containment Isolation Valve Leakage

a. Reactor Vessel Cold Hydro

The reactor pressure vessel and main steam line hydrostatic test (or "cold hydro") was successfully conducted on September 21-23, 1979 in accordance with Startup Test Procedure CS136.001-1 and as required by the ASME Boiler and Pressure Vessel Code at 125% of system design pressure. The test verified the integrity of the reactor vessel, its connecting piping and welds, and portions of the main steam lines by gradual pressurization of the reactor coolant pressure test boundary up to a hydrotest pressure of 1562.5 psig, which was held for 10 minutes, and then reduced to the 1250 psig design pressure for visual inspection of leakage at all welds, joints, connections, mechanical fittings and regions of high stress intensity (such as transition sections). The test was concluded to be efficient, concise and problem-free by the licensee's test engineers. The test was also witnessed by the NRC as part of Inspection 79-15 and found to be in accordance with approved procedures, with test acceptance criteria properly met.

The reactor coolant pressure boundary (RCPB) was initially filled via the Residual Heat Removal (RHR) pumps from the Condensate Transfer system. RHR pump heat was utilized to heatup the vessel and its solid (water) boundary to a 180-200°F test temperature. High head, low flow (0-40 gpm) hydropumps were then used to pressurize the test boundary and control pressure. During the early portion of the test, the entire RCPB was vented (during filling) and at times the drywell was described as "raining" by personnel who were present. Later in the test, at specified pressure plateaus (50, 75, 500 and 1000 psig), UNICO construction inspection walkdowns ensured that the primary purpose of the hydro (i.e. the detection of weld leakage) was being met. Formally-required QA walkdowns to inspect for RCPB leakage at the 1250 psig design pressure were performed on the evening of September 22, 1979.

Two of the three available hydrotest pumps were used, and the only problem encountered during the cold hydro, which was highlighted by procedure CS136.001-1, involved a temporary delay to replace a ruptured suction hose on one of these pumps. Some leakage during the test, allowable by the ASME Code, was observed from equipment seals, valve packing and gasketed joints as well as minor unquantified leakage past valve seats which were part of the test boundary. However, none of the observed (acceptable) leakage masked the detection of leakage (unacceptable) at surfaces of interest, nor did it exceed the capacity of the hydrotest pumps to achieve and maintain desired pressures for the required duration. Valve packing leaks, expected in some cases at the 180-200°F test temperature, would also be expected to diminish significantly (or disappear) at normal operating temperatures of 400-500°F, and should in some cases be corrected at the higher temperatures. Valve packing leaks were stated by licensee personnel to have been experienced on steam lines near the main feed pump turbines, in the turbine-side steam tunnel, but were neither quantified nor mentioned in procedure CS136.001-1. These did not affect the conduct of the cold hydro. Therefore, the number of hydrotest pumps required to maintain test pressure, and the minor test leakage experienced with certain valves, were not a major problem for the cold hydro nor are they a principal concern for a Code hydro test. Rather, only the maintenance of a specified test pressure for some period of time, and the observed integrity of the test pressure boundary's welds and other such joints and connections, were pertinent during the conduct of the cold hydro.

b. Code Hydros-General

Construction acceptance testing of all ASME Code Section III piping and welds applied the same criteria as were used for the vessel cold hydro; the test is acceptable if no leakage is observed by inspection of joints, connections, and high stress regions, exluding that from valve packing and seats. In practice, if the excluded valve leakage were in excess of the pressure source's capability to maintain hydro test pressure, then the test results would be rendered unacceptable and a mechanical equipment Disassembly/Reassembly Release form would be generated to repair the leaky valve's seat, disc or packing. While none of these equipment release forms were found to be generated during the vessel cold hydro, many were created during the numerous system hydros performed for all ASME Section III piping systems. However, in many cases, valves within the test boundary were either pressurized in the reverse direction due to test configurations, or their internals were removed to facilitate the hydro test. The point being that the purpose of a Code hydro is to verify pressure boundary integrity, not valve leak tightness, and that valve leakage is neither meaningfully characterized nor critical during a hydro unless its such that it affects the ability to maintain test boundary pressure. It should also be noted that ASME Code

Sections III and XI specifically exempt valves from re-hydro test when replacing or repairing seats, discs, packing, seals or gaskets.

c. Type C Isolation Valve Testing

The meaningful measure of containment isolation valve (CIV) leak tightness is the "Type C" local leak rate testing (LLRT) performed as required by NRC regulation in 10 CFR Part 50, Appendix J. Initial Type C LLRT for the 215 CIV's at Shoreham was performed during the period March 1982 - February 1983 by methods described in Preoperational Test Procedure PT654.003-1, the results of which were provided to the NRC by LILCo letter dated March 10, 1983. The combined leakage for all resilient seals (Type B) and CIV's (Type C) was found to be well within (by a 95% margin) the acceptance criterion of 10 CFR Part 50, Appendix J. This limit at Shoreham is approximately 4100 standard cubic feet per day (SCFD). These tests were witnessed during numerous NRC Region I inspections (e.g. Report Nos. 50-322/-82-10, 17 and 32) which included reviews of test procedures, data and calculations, and which ascertained compliance with LILCo commitments and Appendix J requirements.

d. Type A Integrated Test

The "Type A" primary containment preoperational integrated leak rate test (CILRT) was successfully performed on December 9-10, 1982. The measured overall leakage, including statistical and other corrections, was also determined to be within the Appendix J acceptance criterion (0.375% by weight per day) by a 37% margin. The CILRT was witnessed by the NRC as part of Region I Inspection 82-32, a report of which was issued on January 18, 1983, and which found test data, assumptions and results to be acceptable with some minor discrepancies noted. A Summary Technical Report was formally submitted to the NRC on March 10, 1983.

All containment leakage testing (Types A, B and C) was performed in accordance with industry standard ANSI/ANS 56.8 - 1981, including the formulae and equations contained in Appendix E of that standard for pressure and temperature corrections using the ideal gas laws.

e. MSIV Leakage Testing

The main steam isolation valves (MSIV's) are the only containment isolation valves with individual leak rate limits (11 5 standard cubic feet per hour per valve), as required by plant Technical Specifications. This leakage is excluded from the summation for LLRT, as well as the CILRT, since an exemption to Appendix J of 10 CFR Part 50 was granted by the NRC for both the method of testing and its inclusion as part of combined containment leakage. Justification for that exemption was documented in the Shoreham Safety Evaluation Report (NUREG-0420, Section 6.2.5.1), and is in part due to the Main

Steam Leakage Control System which will maintain a negative pressure between and collect any leakage past the MSIV's, should isolation be required follow an accident. This leakage source is accounted for separately in the radiological analysis of the Shoreham site, and for that reason, is not required to be considered as part of the local and integrated leakage rate limits. All eight MSIV's were initially tested in the fall of 1981 at Shoreham, and problems were encountered with four of the valves in meeting the 11.5 SCFH limit. Eventually, all eight valves were completely disassembled for refurbishment, repaired, and then successfully passed their LLRT during October-November, 1982. The results of the MSIV refurbishment and testing were documented in a General Electric Maintenance Report prepared in December 1982, and the results of the LLRT reported in LILCo's March 10, 1983 Summary Technical Report to NRC. All four main steam line penetrations were found to be within the 11.5 SCFH limit, with the largest leakage measured as approximately 7.4 SCFH (a margin of 55%) for the MSIV's on penetration X-C. MSIV leak testing will be performed at least once every 18 months when the plant Technical Specifications become effective.

f. Future Appendix J Tests

Finally, Appendix J to 10 CFR Part 50 requires periodic verification of primary containment leak tight integrity by regular surveillance of containment isolation valves and penetrations (Type B and C testing) during each reactor shutdown for refueling, and in no case at intervals greater than two years. With the exception of the MSIV's, there are no individual valve leakage limits which must be applied to CIV's. Rather, Type B and C cumulative leakage must not exceed 60% of the design/maximum value. Also, the Type A CILRT must be performed three times, at approximately equal intervals, during each 10-year plant service period. Type A integrated leakage must not exceed 75% of the design/maximum value. The Appendix J testing ensures that total primary containment leakage during an accident would not exceed the design/maximum value of 0.5% by weight per day (at 46 psig peak accident pressure) which is assumed in Shoreham site radiological analyses.

2.3.3 Suppression Pool Size

a. Increase in Reactor Size

In a June 25, 1982 LILCo response to the Shoreham Atomic Safety and Licensing Board (ASLB) request of May 1982 for information related to primary containment and suppression pool sizes (Item 1, Initial Transcript Reference 1157), the historical background behind the 1968 decision to enlarge Shoreham's generating capacity from 540 to 820 MW (net electrical increase of 52 percent) was presented.

With this increase in power level, the total containment volume was also increased by approximately 11 percent or 40,000 cubic feet. This included a 32 percent increase in pool water volume (an additional 20,000 cubic feet) and a 50 percent increase in pool depth (from 12 to 18 feet). Also, the number of downcomer vents was almost doubled, from 45 to 88, to accomodate the greater potential loss of coolant accident (LOCA) blowdown rate which would be expected at the larger reactor power. Therefore, the Shoreham suppression pool volume was expanded when a decision was made to increase reactor power level.

b. Comparison with Zimmer

In comparing the Shoreham and Zimmer designs, the rated thermal power levels are identical (2436 MW), but their Mark II containment design details differ. The Zimmer suppression pool water volume is 21 percent greater than Shoreham's; however, the Shoreham pool depth is 4½ feet shallower and its cross-sectional area is about 8 percent larger than that at Zimmer. While both designs employ 88 downcomer vents, 6 of these (the ones with vacuum breakers) have been capped at Shoreham, and because the vent diameter for Zimmer downcomers is slightly larger (24 inches vs. 23¼ inches), there's about 14 percent more total vent area at Zimmer. There are many other comparisons to be made with respect to containment designs, not only between Shoreham and Zimmer, but amongst the other five BWR Mark II containments.

Four major functions, for which the suppression pool must be adequately sized, were identified in Item 1 of the ASLB information request. Shoreham's pool is designed to limit the following post-LOCA conditions:

- bulk suppression pool temperature to less than 170°F during the blowdown phase
- containment pressure to less than 48 psig
- bulk pool temperature to less than 190-200°F, depending upon safety/relief valve (SRV) mass flux
- pool temperature to a low enough value which guarantees adequate ECCS pump net positive suction head (NPSH)

The Shoreham Design Assessment Report (DAR) and Final Safety Analysis Report (FSAR) document the results of analyses which demonstrate that the containment adequately meets the above criteria, with margin. For example, the design basis accident (DBA) recirculation suction line break produces a peak calculated containment (drywell) pressure of 46.0 psig - 4 percent below the design pressure. The corresponding peak calculated suppression pool (or wetwell) airspace pressure is 33.7 psig which represents a design margin in excess of 40 percent. By comparison, the Zimmer containment design pressure is 3 psi less than Shoreham's yet the peak calculated design margins for Zimmer's drywell and wetwell airspace (11 and 26 percent respectively) differ from those for Shoreham. The above margins reflect other design differences between these plants; for example, the Shoreham overall containment volume of 408,000 cubic feet is 10 percent larger than Zimmer's, including a 43 percent larger wetwell airspace and a 7 percent larger drywell.

c. BWR-4 Blowdown Characteristics

However, the most significant difference between these plants, directly affecting post-accident containment pressure response and structural loads, is their vintage of General Electric BWR reactor system design. While Zimmer's reactor is a BWR-5 product line with 20-inch diameter recirculation loop piping and 16,000 cubic feet of primary system volume (steam and water), Shoreham's reactor is a BWR-4 type with 28-inch recirculation piping and approximately 18,000 cubic feet primary volume. So, even though both reactors are rated at the same thermal power and temperature/pressure flow conditions, the larger recirculation piping (where the DBA pipe break occurs) and water/steam volume of the BWR-4 causes a more severe mass and energy release (blowdown) from the primary system break. For a BWR-4 such as Shoreham, this pressurizes the drywell faster and initiates higher mass flow through the downcomer vents, causing direct dynamic loading of suppression pool structures and ultimately resulting in pool swell phenomena. Therefore, the larger break area for the Shoreham/BWR-4 design (4.22 vs. 2.24 equivalent square feet) accounts principally for the calculated peak drywell pressure at Shoreham (46 psig at 9.26 sec.) being greater than that at the Zimmer/BWR-5 plant (40.4 psig at 42.5 sec.), even though the entire primary containment volume at Shoreham is actually 10% larger. While the blowdown period lasts for approximately 50-60 seconds for a DBA in either plant design, the increased BWR-4 blowdown drives the suppression pool "harder" at Shoreham; one result being the relatively high pool swell velocity, characteristic of the BWR-4 design, as compared with BWR-5 plants.

d. Other Influencing Factors

There are other factors which tend to reduce the dynamic loads associated with pool swell. These include the pool surface/vent area ratio, downcomer submergence, and pool water depth. The relatively larger pool/vent ratio, and shallower vent submergence and pool depth at Shoreham all tend to mitigate the dynamic loads from the post-accident blowdown. For example, the shallower pool results in less wetted boundary, and therefore less area over which to apply certain loads. Also, the slightly smaller and less-submerged downcomers at Shoreham have 20-30% less mass to accelerate out during the post-accident vent clearing phase preceeding pool swell.

The Shoreham containment was assessed against the generic hydrodynamic loads specified in NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria". One of the four major suppression pool loads which are postulated to occur following a LOCA is the pool swell phase, which refers to the rapid rise of the suppression pool water surface that is driven by drywell air being forced into the pool through the downcomers. The swell is predicted to last at Shoreham for 1.22 seconds following a large break DBA, and involves the acceleration of a slug of water, between the elevation of the downcomer vent exits and the initial pool water surface, due to formation of bubbles which grow and eventually coalesce at the downcomer exits. The water slug or swell continues to accelerate until its retarded by wetwell airspace compression and gravity, at which time it falls back to the original pool surface elevation. This entire process is calculated to last for 2.32 seconds, with a maximum swell height of 20 feet (above the initial surface) and a maximum surface velocity of 42.8 feet per second.

Additionally, while the pool water volume at Shoreham is 21% less than Zimmer's, the wetwell airspace is 43% larger. Peak calculated wetwell air pressure is actually 2 psi less at Shoreham (33.7 vs 35.6 psig at Zimmer). This is significant in the sense that wetwell pressure tends to "drive" the drywell, thereby mitigating the effects of the increased BWR-4 blowdown.

The peak calculated bulk pool temperature for Shoreham during the blowdown phase is 139°F, which is well below the 170°F limit at which complete condensation of DBA blowdown steam has been proven (by test) to occur. Similarly, long-term peak pool temperature is calculated to be 189°F, which is within the required limits. Other factors which will limit bulk pool temperature following a DBA at Shoreham more effectively than at Zimmer include Residual Heat Removal (RHR) heat exchanger efficiency, cooler Service Water, and a lower initial pool temperature. Primarily because of higher shell-side and tube-side flow rates, the overall heat transfer capability of Shoreham's RHR heat exchangers is 15 percent greater than that at Zimmer. Also, the relatively cool Long Island Sound climate at Shoreham allows for a lower initial temperature in the pool and colder Service Water used to cool the RHR heat exchangers, which in turn enhances the capability to perform suppression pool cooling using the RHR system during the long-term DBA period.

e. <u>ASLB Decision</u>

The Partial Initial Decision issued in September 21, 1983 by the Shoreham ASLB addressed Suffolk County Contention 21 for the Mark II containment system design, and concluded that LILCo had met its burden of proof on all aspects of this contention with the exception of one concern for the operation of RHR heat exchangers in the steam condensation mode. The ability of the Shoreham containment to accomodate the hydrodynamic loads associated with a design basis LOCA and SRV actuations, in combination with other loading conditions such as an earthquake, has been documented in the Shoreham Design Assessment Report, and approved by the NRC staff in the Shoreham Safety Evaluation Report (SER/NUREG-0420). In the SER, the NRC staff concluded that the load specifications for assessing all pool dynamic loads were conservative and therefore acceptable.

f. Conclusions

Notwithstanding, the adequacy of the design margin inherent in Shoreham's Mark II containment, in accomodating the potential effects of pool swell phenomena, was presented in the DAR and found acceptable in the NRC's SER, as affirmed by the ASLB Decision published on September 21, 1983. The design modifications to structures within the suppression pool to accomodate pool swell loads included: (1) support redesign; (2) relocation of certain structures (raised vacuum breakers, lowered downcomer bracing, removed grating); and, (3) the addition of flow deflectors and a drywell floor shear ring. None of these changes represent a reduction in the containment design margin, as suggested in this allegation. Inspection of these modifications is described in NRC Region I Inspection 50-322/83-34, a report of which was issued on December 21, 1983.

In summary, the comparison of Zimmer and Shoreham plant designs, suggested by this allegation, shows both reactors to be rated at the same power level (Shoreham is not larger), with Zimmer's pool water volume 21% larger than Shoreham (not 25%). Shoreham's suppression pool design was enlarged when the decision to increase generating capacity was made in 1968, which is contrary to the assertions made in this allegation. The size of the pool does not, however, affect whether or not post-accident pool swell will occur; hence, pool swell is not a problem simply because the pool is too small. The design assessment of Shoreham's Mark II containment, begun in January, 1976 and essentially completed by December 1981 (the initial and final DAR revisions) encompassed a six-year re-analysis/re-design period, the last three years of which being when most of the physical changes were accomplished (not, as alleged, over the past ten years). Finally, the containment design margin, alleged to be too "close," has been demonstrated by the licensee in DAR Section 6 and Appendix B of the DAR to be sufficient to sustain all accident load combinations without exceeding allowable concrete and steel stresses. In fact, containment structures, even at selected critical design sections, were shown in Appendix B of the DAR to have significant reserve capacity in sustaining internal bending moments and axial tension load components. The DAR was reviewed and accepted by the NRC staff, as documented in the SER.

More recently, containment design was evaluated as part of contention SC-21 during the ASLB hearings for Shoreham, and the margins inherent in the Shoreham Mark II design were found to be adequate such that the Board concluded that LILCo had met its burden of proof with respect to those aspects of contention SC-21 related to pool swell phenomena.

2.3.4 Construction Inspection Hold-Points

Phone interviews were conducted on December 21, 1984 with the only two former Courter and Company supervisors to whom the alleger reported to during his tenure as supervisor of Courter crafts personnel (during the period late 1974 through mid-1977). Each of these individuals served as either the Courter Project Manager or Project Engineer, positions to which the supervisor of Courter craft personnel would have reported. Both of the alleger's former supervisors stated that neither had ever given verbal or written instruction to the alleger to bypass or eliminate a required construction QC hold-point.

Former UNICO construction personnel, currently employed by either Stone and Webster or LILCo, were also interviewed during his inspection. None were aware of any attempts by Courter construction supervision to encourage, either formally or informally, the elimination of Construction Inspection Checklist items (hold-points).

3. Exit Interview

On December 21, 1983, the inspector met with R. Glazier, LILCo FQA engineer, and C. Petrone, NRC resident inspector, to discuss the preliminary findings and conclusions of this inspection.