UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of LONG ISLAND LIGHTING COMPANY (Shoreham Nuclear Power Station, Unit 1)	Docket No. 50-322 (O.L.)

PREPARED DIRECT TESTIMONY OF
RICHARD B. HUBBARD
ON BEHALF OF SUFFOLK COUNTY

REGARDING

SUFFOLK COUNTY CONTENTIONS 12, 13, 14 AND 15

QUALITY ASSURANCE/QUALITY CONTROL

June 29, 1982

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SUMMARY OF TESTIMONY

ON SC CONTENTIONS 12, 13, 14 AND 15

Suffolk County contends that LILCO and its major subcontractors have not developed and implemented a QA/QC program for Shoreham in compliance with NRC regulations and LILCO's FSAR commitments.

Rather, there have been major QA/QC breakdowns in the past. Such breakdowns are not isolated or minor but, as indicated by the NRC Staff's February 1982 CAT inspection, include violations of at least eleven of the eighteen Appendix B criteria. The cited QA/QC breakdowns involve a large number of systems, components, and equipment important to the safe operation of Shoreham, leading to the conclusion that there has been a serious and widespread pattern of breakdowns in the QA/QC program implementation for design and construction.

Based on my experience in QA/QC matters, it is my opinion that, given the succession of QA/QC deficiencies already disclosed, further investigation of other safety systems will almost certainly reveal more errors and the violation of other QA criteria. Indeed, this is also suggested by the fact that repeated QA/QC program breakdowns have been identified in all areas subject to the NRC's narrow CAT reinspection program. The evidence of LILCO's pattern of QA/QC breakdowns demonstrates that LILCO has failed to satisfy the requirements of 10 C.F.R. Part 50, Appendix B, which constitute the linchpin of the NRC's defense-in-depth approach to nuclear plant safety.

The Shoreham QA/QC deficiencies are significant also because they went undetected by LILCO and NRC Staff inspectors for years. The NRC's regulations, particularly the eighteen QA requirements of Appendix B to 10 C.F.R. Part 50, are specifically designed to detect such defects on a timely basis and thus to ensure that nuclear plants are designed and constructed under disciplined procedures that systematically assure compliance with regulatory requirements. The QA/QC breakdowns of the pattern discovered at Shoreham should have been detected by LILCO, and on a timely basis, if the Appendix B requirements had been properly implemented. The fact that many QA/QC breakdowns were not discovered until recently -- and only then by a brief NRC Staff inspection rather than by LILCO QA personnel -- is further evidence that the LILCO QA/QC program has not been established and thus not implemented in accordance with regulatory requirements. In my technical judgment, where such a pattern of QA/QC breakdowns occurs and the breakdowns are not discovered by the licensee, the QA/QC program is deficient and the plant cannot be shown to be in compliance with regulatory requirements.

The deficiencies in the QA/QC program at Shoreham cannot be minimized by reference to Staff's QA/QC inspection program. While that program has discovered QA/QC breakdowns at Shoreham, the NRC program is far too limited to provide any assurance that all QA/QC deficiencies have in fact been found. Indeed, the opposite conclusion must be reached: the large number of QA/QC deficiencies found by the Staff's limited I&E program is strong additional evidence that LILCO has not satisfied 10 C.F.R. Part 50, Appendix B.

The NRC's I&E program should provide verification that LILCO has acted systematically to assure that Shoreham structures, systems, and components important to safety are designed manufactured, installed, and operated in accordance with NRC requirements and LILCO's own commitments. The I&E program has not fulfilled this intended function, and the result at Shoreham is that no judgment can be made that Shoreham complies with NRC regulations. First, the I&E program is limited in resources and scope, with I&E auditing only a small portion of the LILCO QA/QC program. Nevertheless, despite these limitations, I&E has found widespread QA/QC breakdowns at Shoreham. These preclude any finding that LILCO has established and implemented an adequate QA/QC program. Surely, if LILCO had such a program, the limited NRC audits would not find a pattern of QA/QC breakdowns.

Second, the I&E program focuses only on those systems, structures and components classified as safety-related. However, the NRC's QA requirements clearly apply to items which are important to safety as well. Thus, there is an entire area of QA review which the Staff has never even addressed. This deficiency, as well as evidence that the LILCO operational QA program also focuses only on safety-related items, leads me to conclude that (1) there has been a failure to satisfy Appendix B requirements and (2) the evidence does not support a finding that the facility has been designed and constructed in accordance with LILCO's FSAR commitments.

A linchpin in the NRC's defense-in-depth approach to nuclear safety is its emphasis upon QA and QC. The evidence of QA/QC breakdowns and the limited I&E program regarding QA/QC have established substantial uncertainty in the actual quality level achieved in design, construction, and installation of structures, systems, and components at Shoreham. In my opinion, this uncertainty prevents a finding that LILCO has met regulatory requirements and FSAR commitments.

The serious pattern of breakdowns at Shoreham and the accompanying doubt concerning Shoreham's compliance with applicable criteria results in the facility being unlicenseable at the present time. If LILCO, nevertheless, seeks to overcome this lack of demonstration of requisite quality assurance, in my opinion it can do this only by performing a design verification and physical inspection of all Shoreham structures, systems, components, and other features important to safety by an independent consulting firm. The results of such a complete design review and physical inspection, as well any necessary changes to plant design and construction, should be subject to the scrutiny of the Board and the meaningful involvement of all parties in the ongoing Shoreham licensing proceeding. Only by such steps is it possible to ascertain the actual quality achieved at Shoreham and thus attempt to compensate for LILCO's failure to comply with Part 50, Appendix B.

Finally, LILCO has failed to demonstrate that its QA/QC program for operation of Shoreham will be implemented in compliance with NRC requirements. LILCO's operating QA program is only summarily described in the FSAR and, from review of the LILCO QA manuals, its scope is far too limited, particularly since it excludes items which are important to safe operation of Shoreham. Thus, I conclude that the operating QA program does not satisfy Appendix B and GDC 1 of Appendix A. This deficiency is all the more significant because, unlike during construction where LILCO essentially provided an audit function, during operation LILCO personnel will be directly responsible for conducting a majority of the QA/QC program activities. LILCO QA/QC personnel have little direct experience in implementing a nuclear grade quality program at an operating nuclear station. Thus, the lack of necessary detail or scope of the operating QA program precludes a finding that LILCO will meet regulatory requirements.

<u>LIST OF ATTACHMENTS</u>

1	Supplement, Experience and Qualifications of Richard B. Hubbard
2	Appendix I to SC Contention 12
3	Testimony of NRC Chairman Palladino and NRC Operation's Director Dircks before the U.S. House Subcommittee on Energy and the Environment, November 19, 1981
4	IE Report 50-322/82-04, As Built Inspection of the Residual Heat Removal System, May 12, 1982 ("CAT Inspection")
5	Summary of Shoreham "Violations" and "Deviations" Cited by NRC IE Program (1974 to February 1982)
6	Staff Report on Quality Assurance by the President's Commission On the Accident at Three Mile Island, October 1979
7	Report For the Combined Utility Assessment of the Adequacy of the LILCO QA Program for Nuclear Application, February 27, 1981
8	LILCO's News Release and Viewgraphs Describing Its Shoreham Physical Inspection Program
9	LILCO's Independent Design Review of the Core Spray System
10	Guidelines for Independent Inspection, Task C, QA/QC Programs

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PREPARED DIRECT TESTIMONY OF
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QUALITY ASSURANCE/QUALITY CONTROL

I. INTRODUCTION

This testimony, which addresses the Shoreham Quality Assurance/Quality Control (QA/QC)[1/] program for design, construction, installation, and operation activities, was prepared by Richard B. Hubbard. A statement of my qualifications and experience has been separately provided to this Board. Supplemental information related to my QA/QC experience is provided in Attachment 1.

II. STATEMENT OF CONTENTIONS

The purpose of this testimony is to address Suffolk County ("SC" or "County") Contentions 12, 13, 14 and 15 as admitted by the Board as follows:

1/ "Quality Assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes "Quality Control," which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements. (*See* "Introduction" to Appendix B to 10 C.F.R. Part 50.)

SC 12: QA/QC - DESIGN, CONSTRUCTION, AND INSTALLATION

Suffolk County contends that LILCO and the NRC Staff have not adequately demonstrated that the quality assurance program for the design and installation of structures, systems, and components for Shoreham was conducted in a timely manner in compliance with the pertinent portions of 10 C.F.R. 50, Appendix B, Sections I to XVIII, and 10 C.F.R. 50, Appendix A, GDC 1. There has been a pattern of QA/QC breakdowns at Shoreham. This pattern has covered Appendix B Criteria 2, 3, and 5-18. Accordingly, Suffolk County contends that LILCO has failed to comply with 10 C.F.R. Part 50, Appendix B, as particularized in Appendix I attached hereto. 2/

SC 13: QA/QC - OPERATIONS

Suffolk County contends that the QA program description for the operation of Shoreham, as provided in the FSAR, does not comply with 10 C.F.R. 50.34(b)(6)(ii) and 10 C.F.R. 50, Appendix B, Sections I to XVIII, with regard to:

(a) Failure to address, at a minimum, each of the criteria in Appendix B in sufficient detail to enable an independent reviewer to determine whether and how all the requirements of Appendix B will be satisfied;

(b) Failure to provide for the adequate identification, reporting, and analysis of all equipment failures discovered during operation and maintenance at Shoreham and at other operating BWR stations with similar equipment;

(c) Failure to ensure that replacement materials and parts of systems classified as components "important to safety" will be equivalent to the original equipment, that replacements will be installed in accordance with adequate process procedures, and that the repaired or reworked structures, systems, or components will be adequately inspected, tested, and documented in "as-built" drawings; and

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2/ Appendix I to SC Contention 12 is Attachment 2 hereto.

(d) Failure to provide an adequate number of qualified QA/QC personnel on the operating staff, including the availability of QC personnel on off-shifts.

SC 14: QA/QC - NRC I&E PROGRAM

Suffolk County contends that the NRC Staff's Inspection and Enforcement (I&E) Program has not adequately verified that LILCO's quality assurance program for Shoreham has been implemented in accordance with the requirements of 10 CFR 50.34(a), paragraph 7 and 10 CFR 50, Appendix B, Sections I through XVIII, in that:

(a) The I&E Program has identified only the symptoms of the Shoreham quality deficiencies as nonconformances, and has not required LILCO to initiate corrective action to resolve the root causes;

(b) The I&E Program's reliance on LILCO for primary inspections at Shoreham with NRC officials serving as auditors is not adequate because the same practice has recently proved to be inadequate in timely identifying quality deficiencies at other nuclear facilities (e.g., Browns Ferry, North Anna, Davis Besse, and Rancho Seco); and

(c) The I&E Program has no baseline criteria against which to measure quantitatively the effectiveness of the Shoreham quality program. In order to draw conclusions from random inspections just such a statistical base is desirable. In addition, the I&E program has no means of determining improvements in, or the effectiveness of, corrections as no comparative measure are used.

Therefore, no general conclusions as to the adequacy of the LILCO program can be drawn by I&E.

SC 15: QA/QC - REVIEW AND INSPECTIONS

Suffolk County contends that the Shoreham quality assurance/quality control program has involved inadequate review and physical inspection to verify compliance with 10 CFR 50, Appendix B. The inability to verify full compliance with Appendix B is based on inadequacies discovered independently at Diablo Canyon, Zimmer, Midland and South Texas, and on statements by NRC Chairman Palladino. Similarly, the random, non-systematic approach taken by NRC I&E to verify quality programs is inadequate to provide appropriate assurance of compliance. Also the random checks being conducted by the NRC resident inspector at Shoreham indicate lapses, breakdowns, and inconsistencies that do not provide credible public assurance of an operable QA System. For example, see NRC Inspections 50-322/79-05, 80-03, 80-06, 80-08, 80-14 and 8. 02. Because the NRC reports do not indicate what changes, if any, were made in procedures to correct for failures that have occurred, it is not possible to judge the adequacy of corrective actions. Finally, there are no quantitative measures used to assure that NRC I&E and LILCO audits can be correlated statistically to provide verification of the adequacy of the QA system to detect system or equipment errors or distinguish between random errors and systematic failures.

Therefore, there is no assurance that LILCO has complied with 10 CFR 50.55(e) and 10 CFR 50, Appendix B, Sections XVII and XVIII. Suffolk County contends that NRC I&E and LILCO cannot provide assurance of compliance without systematically auditing QA documentation against physically inspectable structures and components. This physical audit should be sufficiently detailed to provide statistically valid data to permit projection of the audit results to systems beyond those systems and QA records inspected.

This testimony is organized into five major sections (III through VII) and a conclusion section (VIII). Background data, including the importance of QA/QC in nuclear plant safety, are covered in Section III. Recently disclosed breakdowns in the implementation of the Shoreham QA/QC program for design, construction,and installation are summarized in Section IV, while the failure of the NRC's Inspection and Enforcement (I&E) program to demonstrate that the root causes of the Shoreham QA/QC breakdowns have been identified and resolved is presented in Section V. Section VI covers areas of non-compliance in the Shoreham operating QA/QC program. Finally, the need for an independent design review and physical inspection to determine whether Shoreham design and construction, as implemented, complies with applicable regulations and the license application is covered in Section VII.

III. BACKGROUND AND IMPORTANCE OF QA/QC

The Shoreham Nuclear Power Plant, like other nuclear power facilities, is a complex installation with numerous design and construction interfaces, with each aspect, particularly the interfaces, creating the opportunity for an error in communication or implementation of requirements. Thus, the QA/QC program exists first to ensure that the plant is designed and constructed in compliance with the prescribed regulations as committed by Long Island Lighting Company (LILCO) in the license application (Final Safety Analysis Report ("FSAR") and, where appropriate, the PSAR) and second to document this compliance.

QA/QC requirements for nuclear power plant design, construction, and operation have expanded greatly since the late 1960's when the Shoreham project was initiated. Early methods of ensuring quality were largely informal, guided by a modest set of codes and standards. Thus, prior to 1970, there was no explicit regulatory requirement for a QA/QC program, but good industry practice would have resulted in a detailed QA/QC program. Starting in 1970, however, a new approach was developed, requiring adherence to a number of specific, detailed NRC rules and industry standards. There is no question that the NRC's QA/QC criteria had a significant impact on utility work practices. For example, the Vice President of Engineering for LILCO stated: 3/

3/ Direct testimony of Dr. Matthew C. Cordaro, Vice President of Engineering for LILCO, NY PSC Case # 27563, Nov. 2, 1981, p. 160.

> ... on fossil facilities construction engineers had considerable latitude in resolving installation problems. This latitude is not allowed for a nuclear plant. According to Appendix B of 10 CFR 50, 'design changes including field changes must be approved by the organization which performed the original design.' As a result, the construction organization at Shoreham has generated over 33,000 design changes reports that required disposition and approval by the Engineering organization.

The QA requirements which evolved during the early 1970's were pervasive, reaching into every aspect of nuclear plant design, construction, and operation. For example, the owner of Diablo Canyon, which was completed during the same time frame as Shoreham, described the effect of the evolving QA/QC requirements on construction as follows: 4/

> "We did not....anticipate the detail in documentation and independent inspection of workmanship which would be required by the NRC. For instance, simple field changes to avoid physical interference between components (which would be made in a conventional plant in the normal course of work) had to be documented as an interference, referred to the engineer for evaluation, prepared on a drawing, approved, and then released to the field before the change could be made. Furthermore, the conflict had to be tagged, identified, and records maintained during the change process. These change processes took time (days or weeks) and there were thousands of them. In the interim the construction crew must move off of this piece of work, set up on another and then move back and set up on the original piece of work again when the nonconformance was resolved."

4/ Brand, Donald A., Vice President of Engineering, PG&E testimony before the California Public Utilities Commission, Application Nos. 58911 and 58912, June 6, 1979, pp. 17-18.

The new NRC QA/QC rules and standards had one basic purpose: to ensure that the design, construction, and operation of nuclear power facilities such as Shoreham met high quality standards, thus providing assurance that the plants would generate power safely. In the past few years, however, a series of QA program breakdowns have occurred,[5/] which raise serious questions about utilities' compliance with NRC requirements and the ability of the NRC licensing review and inspection process to provide assurance that these QA rules have been effectively implemented. These QA/QC breakdowns include:

- -- <u>South Texas Project</u> -- multiple design flaws as set forth in the recent Quadrex report following earlier disclosure of widespread construction flaws;
- -- <u>Stone & Webster Seismic Models</u> -- incorrect seismic analysis resulting in a 5-plant shutdown followed by a series of NRC bulletins.

The Diablo Canyon and Zimmer QA/QC errors are the most recent examples. The significance of these breakdowns for Shoreham is that they demonstrate the ineffectiveness of the NRC I&E program in providing verification of utilities' compliance with Appendix B. The breakdowns also demonstrate the need for independent design and construction verification programs at these facilities and at Shoreham. These matters are discussed in Sections V and VII of this testimony. In the following portion of this testimony, the

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5/ These recent significant QA/QC breakdowns are described in the testimony of NRC Chairman Palladino and NRC Director of Operations Dircks' November 19, 1981 testimony before the U.S. House Subcommittee on Energy and the Environment. The Dircks and Palladino testimonies are included herein as Attachment 3. The author of this testimony also attended and presented testimony at the November 19 hearing.

NRC's required QA/QC measures are described which, if properly applied by LILCO, would have prevented the widespread quality breakdowns which have occurred at Shoreham, and would have identified at an early date those breakdowns which did occur.

III.A. Evolution of the NRC's QA/QC Requirements

On April 17, 1969, the Atomic Energy Commission (AEC) published in the Federal Register, Volume 34, No. 73, a proposed amendment to 10 C.F.R. Part 50 which would add an Appendix B, "Quality Assurance Criteria for Nuclear Power Plants." This Appendix was officially adopted as an NRC regulation on June 27, 1970, and published in the Federal Register, Volume 35, No. 125. Shoreham construction started in September, 1972, with the construction permit issued by the NRC on April 14, 1973.[6/] Thus, the NRC QA/QC regulations had been in place for nearly three years at the time construction activities for safety features were authorized for Shoreham.

Requirements of 10 C.F.R. 50, Appendix B, apply directly to and place responsibility on the Applicant (LILCO) for the establishment and execution of the total quality assurance program. Appendix B and Appendix A, GDC 1, require quality assurance to be applied to virtually all the activities associated with the design, construction, and operation of those structures, systems, and components "important to safety" from which satisfactory

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6/ NUREG-0030, Construction Status Report.

performance is required to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Thus, the requirements of Appendices A and B apply to a wide range of interrelated activities affecting the safety functions of structures, systems, and components, including designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

The Commission further requires each Applicant for a construction permit to include in its Preliminary Safety Analysis Report (PSAR) a discussion of how the applicable QA requirements will be satisfied during the design, fabrication, construction, and testing of the structures, systems, and components of the facility.[7/] Likewise, the Commission requires every Applicant for an operating license for a nuclear plant to include in the Final Safety Analysis Report (FSAR) a discussion of how the applicable requirements of Appendix B will be satisfied in order to assure safe operation.[8/]

In response to the AEC's adoption of the QA/QC requirements, the American National Standards Institute (ANSI) in 1970 established committees to develop formal, explicit guidance procedures for licensees and their contractors on how to implement the AEC QA regulations. ANSI is an umbrella organization for

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7/ *See* 10 C.F.R. 50.34(a)(7).

8/ *See* 10 C.F.R. 50.34(b)(6)(ii).

technical societies including the Institute of Electrical and Electronics Engineers (IEEE) and the American Society of Mechanical Engineers (ASME) whose memberships have a professional interest in and familiarity with nuclear power plants. The ANSI working groups are composed primarily of representatives of electric utilities, power plant designers, and equipment manufacturers but include nominal AEC/NRC representation. Beginning in 1971 and continuing through the 1970's, the ANSI committees issued over a dozen final and draft QA standards pertaining to virtually every phase of plant design and construction involving safety equipment. These include designing, pruchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, and testing (see Table III-1). Most of the final and some of the draft standards have been endorsed in AEC/NRC regulatory guides.

The majority of the implementing ANSI standards for planning, managing, and performing overall QA programs were issued and available throughout the nuclear industry in final or draft form prior to 1974. In early 1974, the AEC issued for guidance a series of QA documents which included draft or final versions[9/] of the ANSI standards set forth in Table III-1 (except N45.2.16 and N45.2.23) as follows:

9/ WASH 1283, 1284, and 1309 contained a number of draft QA standards. As these draft standards were issued as approved American National Standards, they were endorsed by Regulatory Guides. The applicability of the Regulatory Guide versus the draft standard was addressed in the implementation section of the Guide.

(a) WASH 1284, October 26, 1974, Guidance on Quality Assurance Requirements During the Operations Phase of Nuclear Power Plants.

(b) WASH 1309, May 10, 1974, Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants, designated the "Green" book.

(c) WASH 1283, May 24, 1974, Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants - Revision 1, designated the "Gray" book.[10/]

For Shoreham, the Staff's SER is totally devoid of any analysis or conclusions concerning the compliance of the design, construction, and installation of QA/QC activities, as implemented, with the regulatory requirements and the PSAR commitments. This is a significant omission since it means that the LILCO QA program has undergone no review by the Staff, which is contrary to the important role of QA/QC in nuclear power plant safety, as discussed in the next section. Rather, the Shoreham SER only addresses the compliance of the proposed operating QA/QC program.[11/] As will be demonstrated hereafter in Section VI, there are serious deficiencies in the operating QA/QC program.

10/ Revision 0 of the "Gray" book was issued on June 7, 1973.

11/ NUREG-0420, Shoreham SER and Supplements, Section 17.

TABLE III-1

ANSI QA/QC STANDARDS

Standard Number	Year First Adopted	Subject	Reg. Guide Number	Reg. Guide Year Issued	Reg. Guide Current Revision & Date Issued
N45.2	1971	General QA requirements	1.28	1972	R2(79/02)
N18.7	1972	QA Program Requirements-Operation*	1.33	1972	R2(78/02)
N45.2.1	1973	Cleaning of fluid systems	1.37	1973	- -
N45.2.2	1972	Packaging, shipping, receiving, storage and handling of equipment	1.38	1973	R2(77/05)
N45.2.3	1973	Housekeeping during construction	1.39	1973	R2(77/09)
N45.2.4	1972	Installation, inspection and testing of instrumentation and electric equipment	1.30	1972	- -
N45.2.5	1974	Installation, inspection and testing of structural concrete and steel	1.94	1975	R1(76/04)
N45.2.6	1973	Qualifications of inspection, examination and testing personnel	1.58	1973	- -
N45.2.8	1975	Installation, inspection and testing of mechanical equipment and systems	1.116	1976	0-R(77/05)
N45.2.9	1974	Collection, storage and maintenance of QA records	1.88	1974	R2(76/10)
N45.2.10	1973	QA terms and definitions	1.74	1974	- -
N45.2.11	1974	QA requirements in plant design	1.64	1973	R2(76/06)
N45.2.12	1977	Requirements for auditing QA programs	1.144	1979	- -
N45.2.13	1976	Procurement of items and services	1.123	1976	R1(77/07)
N45.2.16	1975	Calibration and control of measuring and test	None	N/A	- -
N45.2.23	1978	Qualification of audit personnel	1.146	1980	- -

* Other relevant ANSI Stds. for an operating QA program, as referenced in Reg. Guide 1.33 and ANSI N18.7, include ANSI N18.1 (RG 1.8), ANSI N18.17 (RG 1.17), and ANSI N101.4 (RG 1.54).

III.B. Importance of QA/QC

The importance of the results of such a QA/QC review cannot be denied. The linchpin in the NRC's "defense-in-depth" approach to nuclear safety is its emphasis upon QA and QC in the design, construction, and operation of a nuclear power plant. The NRC in its 1980 Annual Report emphasized that:[12/]

> "The application of disciplined engineering practices and thorough management and programmatic controls to the design, fabrication, construction, and operation of nuclear power plants is essential to the protection of public health and safety and of the environment. QA provides this necessary discipline and control. Through a QA program that meets NRC requirements, all organizations performing work that is ultimately related to the safety of plant operation are required to conduct that work in a preplanned and documented manner; to independently verify the adequacy of completed work; to provide records that will confirm the acceptability of work and manufactured items; and to assure that all individuals involved with the work are properly trained and qualified to carry out their responsibilities." (Emphasis added)

Likewise, the NRC Licensing Boards have repeatedly observed that compliance with applicable quality assurance standards is an issue of critical importance to nuclear plant safety. As the Appeal Board noted in In the Matter of Consumers Power Company (Midland Plant, Units 1 and 2), ALAB-106, 6 AEC 182 183 (1972), "|o|ne of the most significant elements of the Commission's 'defense-in-depth' approach

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12/ NUREG-0774, NRC 1980 Annual Report, pp. 69-70.

to nuclear safety is its emphasis upon quality assurance and quality control in the construction of nuclear power plants." Another Appeal Board in In the Matter of Duke Power Company (William B. McGuire Nuclear Stations, Units 1 and 2), ALAB-128, 6 ACE 399, 410 (1973), observed that:[13/]

> In an area as significant as quality assurance, the record should leave no doubt as to whether the applicant is in full compliance with applicable criteria and, if not, the basis upon which the regulatory staff authorizes any departure from such criteria. (emphasis supplied).

The preceding is not meant to infer that QA/QC will ensure that errors will be totally eliminated. Rather, in designing and constructing any complex facility, errors are inevitable because people are not infallible. QA and QC recognize human imperfections and thus impose a control system designed to detect those inevitable errors and, therefore, to ensure that the facility is, in fact, designed and constructed to the required standards. As will be set forth hereafter in Section IV, there is substantial evidence that the Shoreham QA/QC program has, in fact, not identified and corrected all the non-compliances with the FSAR commitments and the NRC regulatory requirements. Indeed, in my opinion, the facts clearly document a serious failure of LILCO to implement the required QA/QC program.

13/ See also, In the Matter of Consumers Power Company (Midland Plant, Units 1 and 2), LBP 74-1, 8 AEC 584, 597-600 (1974); In the Matter of Commonwealth Edison Company (Zion Nuclear Power Plant, Units 1 and 2), LBP-73-35, 6 ACE 861, 896 (1973).

IV. DEFICIENCIES IN THE IMPLEMENTATION OF THE SHOREHAM QA/QC PROGRAM FOR DESIGN, CONSTRUCTION, AND INSTALLATION DOCUMENT A FAILURE OF LILCO TO COMPLY WITH QA/QC REQUIREMENTS

Throughout the construction of Shoreham, there have been repeated violations of Appendix B requirements by LILCO. These violations have been documented in I&E reports and are summarized in Attachments 2 and 5 to this testimony. These violations, by themselves, disclose a pattern of QA/QC violations which show that LILCO did not comply with Appendix B requirements during the design and construction of Shoreham.

In considering LILCO's QA/QC breakdowns, it is important to keep the overall QA/QC objectives in mind. LILCO has previously presented, on paper, a description of its QA program for design and construction of Shoreham. However, the adequacy of that program cannot be judged by the words therein but only by examination of the program's actual implementation. This point was emphasized by Harold Denton at the Staff's March 15, 1982 meeting with LILCO (attended by this author) where he challenged LILCO not to demonstrate by objective evidence that the requisite QA/QC measures were, in fact, effectively implemented at Shoreham.

The actual performance of the LILCO program is one of failure to comply with Appendix B. In addition to those breakdowns described in Attachments 2 and 5, there now is additional evidence of further breakdowns by LILCO which document that LILCO has not complied with Appendix B. Indeed, a recent special first of a kind NRC inspection conducted by the Construction Assessment Team (CAT) from NRC Region I documents that critical design and construction QA/QC controls at Shoreham are lacking and that a number of activities at Shoreham were not conducted in accordance with FSAR commitments.[14/]

IV.A. NRC Region I Team Identifies Failure To Comply with QA/QC Regulations and FSAR Commitments at Shoreham

A special inspection of Shoreham was conducted by personnel from Region I of the NRC on February 8 to 26, 1982. This was 12 years after Appendix B was adopted and almost 10 years after Shoreham had received its construction permit. The purpose of the recent inspection was a comparison of the completed construction and physical installation at the Shoreham Nuclear Power Station with regulatory commitments and engineering and design documents. No attempt was made during the CAT inspection

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14/ IE Report 50-322/82-04 dated May 12, 1982. The report is included herein as Attachment 4.

to visit GE or Stone & Webster to systematically verify that the design documents were consistent with the design criteria set forth in the PSAR and FSAR. Limited resources were devoted to the CAT inspection. The inspection involved only 373 hours on-site and 73 hours in-office by 3 region-based inspectors, a supervisor, and the Shoreham senior resident inspector. Nevertheless, as will be shown in the following discussion, a wide number and variety of QA/QC deficiencies were discovered during the CAT inspection.

Team members inspected the physical installation of the Residual Heat Removal (RHR) System and compared the as-built installation to flow diagrams, logic diagrams, construction drawings, and other design and engineering information. The RHR system, of course, plays an important safety role in the long term cooling mode and has been classified by LILCO as safety-related. Thus, all applicable Appendix B criteria must be fully satisfied for this system. The NRC inspections also examined selected portions of other plant systems which are required to support the RHR system in normal and emergency operation. In the course of the inspection, random sampling was done of construction and management control activities such as purchase documentation, material control, quality control inspections, repair and rework, engineering and design changes, and maintenance of completed installation.[15/]

15/ Ibid 14, p. 3.

During the limited CAT inspection the following QA/QC breakdowns were discovered:

IV.A.1: QA/QC Breakdowns 1 to 7 - Failure to Control Design

Criterion 3 of Appendix B requires in part that "measures shall be established to assure that applicable regulatory requirements and the design basis,.....as specified in the license application,.....are correctly translated into specifications, drawings, procedures, and instructions." Contrary to this requirement, the NRC inspection team identified seven discrepancies between the as-built configuration and LILCO's commitments in the FSAR as follows:[16/]

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

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

16/ Ibid 14, pp. 6-7.

c) FSAR Fig. 7.3.1-6 and Table 7.3.2-4 show LPCI loop selection logic and instruments. Contrary to this, the logic has been deleted.

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

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

- Loop fill on B loop should be between valves F015 and F017.
- Relief valves F030A-D go to floor drains, not CRW.
- Relief Valve F025 is not a thermal relief, contrary to Note 12.
- Location of line to Radwaste thru valves MO-F040 and F049 shown incorrectly.
- Cooling water to RHR pumps is RBCLCW, not the emergency equipment cooling water.
- Drains from RHR pump suction and discharge do not tie together.

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

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

IV.A.2: QA/QC Breakdowns 8 to 15 - Failure to Verify Conformance With Drawings and Control Nonconformances

Criteria 10 and 11 require in part that inspections and tests be conducted to verify conformance with drawings. Contrary to this requirement, there have been numerous instances of differences between the "as built" and "as designed" configuration of the plant. For example, the following eight discrepancies between flow diagrams and existing piping and hardware were identified by the NRC team:[17/]

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

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

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

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

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

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

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

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

17/ Ibid 14, pp. 7-8

In addition, in the preceding examples, contrary to the requirements of Criterion 15 of Appendix B, LILCO also failed to establish measures to control materials, parts, or components which do not conform to requirements to prevent their inadvertent use or installation during construction activities at Shoreham.

IV.A.3: QA/QC Breakdown 16 - Failure To Preserve Equipment

Criterion 13 requires that measures be established to preserve materials and equipment to prevent damage or deterioration. Contrary to this requirement, the inspection team found that: 18/

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

IV.A.4: QA/QC Breakdown 17 - Failure To Indicate Status

The inspection team found that: 19/

18/ Ibid 14, p. 9.

19/ Ibid 14, p. 10.

> .....several electrical jumpers used in Control Room panel 612 and two wires from cable 1B21BBX198 in Control Room panel 601 which had been removed from the terminal block were not tagged. The free terminals were not protected against possible shorting of adjacent terminations.

The preceding situation is contrary to the requirement of Criterion 14 that measures be established to indicate by the use of markings such as tags the status of inspections and tests.

IV.A.5: QA/QC Breakdowns 18 to 22 - Failure To Document Activities Affecting Quality and Failure To Prevent Installations Which Do Not Conform To Requirements

Criterion 5 requires that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings and shall be accomplished in accordance with these documents. Contrary to this requirement, the following was found concerning separation of electrical cables:[20/]

> a) .....separation of cables in transition from tray to tray and tray to conduit was not addressed specifically in the FSAR nor was transition separation addressed in (Stone and Webster) electrical installation Specification SH1-159.

20/ Ibid 14, pp. 11-12.

b) ....the required one-inch horizontal and one-foot vertical separation criteria were not maintained for non-class 1E/class 1E cables in transition from one raceway to another for the following raceway groupings:

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

-- 1TC400R, 1TC404R and 1TC411N.

-- 1TK794B, 1TK785N.

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

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

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

The NRC team noted that a walkdown to verify cable separation was planned by LILCO for all safety-related cables as part of the electrical "as-built" program. However, in February, 1982, LILCO was not able to provide the NRC with written instructions or procedures for this planned verification.[21]

In addition the NRC once again found that:[22]

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

The preceding installation of cables in a non-conforming manner is contrary to the requirement of Criterion 15 that measures be established to control items which do not conform to requirements in order to prevent their inadvertent use or installation.

IV.A.6: QA/QC Breakdown 23 - Failure to Prevent Deterioration and Failure to Document Activities Affecting Quality

Criterion 13 requires in part that measures be established to control the preservation of items to prevent

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21/ Ibid 14, p. 12.

22/ Ibid 14, p. 12.

deterioration. Criterion 5 requires that activities affecting quality shall be prescribed by instructions, procedures, or drawings and shall be accomplished in accordance with these documents. Contrary to the requirements of these criteria, the CAT inspectors found that: 23/

> ....electrical components were not completely sealed to prevent moisture entry. The final run of cable to a component was often via a metal conduit. The conduit was not sealed where the electrical cable entered. This opening provides a moisture entry path to the component. A review of the associated documentation revealed that conduit sealing was required. Two Valcor Engineering Corp. solenoid valves in the RHR system (E11*SOV-166A and 167A) had the following note in the manufacturer's technical manual: "Owner is responsible for sealing the conduit connection and preventing the entrance of moisture thru the conduit to maintain the validity of the IEEE-323 qualification". Also, E&DCRs F-5750 and F-575DA, dated December 7, 1976 and January 6, 1977 respectively, stated that the Reactor Building and other areas are considered wet locations and required that conduits in these areas be sealed. Despite the above requirements, the licensee was unable to identify any existing program or procedure which would seal the subject conduits or inspect the adequacy of the seals, once installed.

IV.A.7: QA/QC Breakdown 24 - Failure to Control Design To Assure Regulatory Requirements And Licensee Commitments Are Achieved

The CAT inspectors reviewed the RHR system controls

23/ Ibid 14, p. 13.

against applicable regulatory requirements, licensee commitments, good human factors practices, and inspector judgment. The following deficiencies were found: 24/

a) Annunciator 1122 has a seemingly contradictory label.

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

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

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

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

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

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

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

i) Local instruments are not clearly labeled as to function.

24/ Ibid 14, p. 14.

These items had not been specifically addressed by LILCO or the Staff in earlier control room human factors reviews. Thus, contrary to the requirements of Criterion 3, LILCO failed to establish measures to assure that applicable regulatory requirements and the design basis, as specified in the license application, are correctly translated into drawings.

IV.A.8: QA/QC Breakdowns 25 and 26 - Failure To Control Design For Manual System Initiation

Criterion 3 states, in part, that measures shall be established to assure that applicable regulatory requirements as specified in the license application are correctly translated into specifications, drawings, procedures, and instructions. In addition, 10 CFR 50.55a(h) states that protection systems shall meet requirements of Institute of Electrical and Electronics Engineers Standard 279 (IEEE-279). Paragraph 4.17 of IEEE-279-1971 requires that protection systems include means for manual initiation of each protective action at the system level. The Shoreham FSAR in paragraphs 7.3.2.1.2.19 and 7.6.2.5.2.12 states that the Emergency Core Cooling System and the Reactor Building Closed Loop Cooling Water System meet Regulatory Guide 1.62, which describes an acceptable method of complying with IEEE-279 Paragraph 4.17. Paragraph C.2 of

RG 1.62 states that manual initiation of a protective action at the system level should perform all actions performed by automatic initiation, such as starting auxiliary or supporting systems and sending signals to appropriate valve-actuating mechanisms to assure correct valve position. Contrary to the preceding, the NRC team found the following QA/QC deficiencies:[25]

a) .....the manual initiation circuitry for the Low Pressure Coolant Injection (LPCI) System (a portion of ECCS) does not provide signals to start and assure correct valve position for the following LPCI auxiliary systems: Reactor Building Closed Loop Cooling Water (RBCLCW) for LPCI pumps seal coolers, area coolers for air-cooled LPCI pump motors, or chilled water to these area coolers. Additionally, eight LPCI valves are not sent signals to assure correct valve position upon manual initiation.

b) .....there is no system level manual initiation for the RBCLCW system.

IV.A.9: QA/QC Breakdowns 27 and 28 - Failure To Control Design For Override or Bypass Indication

IE Bulletin 79-08 sets forth controls for valve positioning. LILCO's response to Bulletin 79-08 states that if a motor operated valve (MOV) has a given safety position, and it is moved from that position with consequent loss of ability to return automatically, then its respective system "inop" alarm is sounded. Two areas of the RHR system were found by the CAT inspectors not to satisfy this commitment.[26]

25/ Ibid 14, Notice of Violation, pp. 1-2.

26/ Ibid 14, p. 16.

a) closure of a single RHR pump suction valve, E11*MOV-031; and

b) the case where E11*MOV-037 A and B are blocked closed by a shutdown cooling isolation signal.

As before, the preceding are contrary to the design control requirements of Criterion 3.

IV.A.10: QA/QC Breakdown 29 - Failure To Meet General Design Criterion and To Initiate Corrective Action

During the review of the containment isolation valve designs for the RHR system, the CAT inspectors identified one line whose valves:[27/]

> .....did not meet the requirements of 10 CFR 50, Appendix A, Criterion 56. This criterion describes the CIV's required for lines which penetrate the primary containment and connect directly to the containment atmosphere. The RHR system line connected to penetrations X43 and XS-5 is such a line. For this line Criterion 56 requires two CIV's, which must be either automatic or locked closed and which must not be check valves. A HPCI steam drain line ties into this RHR system line and has only two check valves (numbers 3144 and 3145) as containment isolation valves. The arrangement is depicted in FSAR Fig. 6.2.4-2. This violation of GDC-56 was not identified nor justified prior to the inspection. 10 CFR 50, Appendix B, Criterion III requires correct translation of applicable regulatory requirements.

Further the CAT inspectors also found that LILCO had not yet resolved a previous violation (322-81-02-01) which involved

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27/ Ibid 14, pp. 17-18.

a similar situation where CIV's were not located as close as practical to containment. 28/ The preceding is contrary to the requirement of Criterion 16 that measures be established to assure that conditions adverse to quality are promptly identified and corrected.

IV.A.11: QA/QC Breakdown 30 - Failure To Document Procurement Requirements

Criterion 4 requires in part that measures be established to assure that applicable requirements which are necessary to assure adequate quality are included or referenced in procurement documents. Contrary to this requirement, the CAT inspectors found for globe valves that: 29/

> .....the licensee had stated in a submittal that the testing would be conservative since the valves were normally seated with a force at least three times greater than the test pressure seating force..... The "three times" criterion was met for all valves except two, E41*MOV-049 and E51*MOV-049, as purchased. These two valves required additional demonstration of meeting the "three times" criteria; this was provided by the vendor. The inspector questioned controls existing to ensure that closing force for these two valves would be maintained throughout plant life, including maintenance or replacement. For maintenance, the proper valve actuator torque switch settings are documented. For replacement, the licensee acknowledged that existing

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28/ Ibid 14, p. 17.

29/ Ibid 14, p. 18.

> controls might not be sufficient. Therefore, prior to completion of the inspection, the licensee issued an E&DCR to the valve purchase and (sic) specification, which noted the "three times" requirement for these two valves to ensure proper controls if replacement is required. (emphasis added)

IV.A.12: QA/QC Breakdown 31 - Failure To Document and Control Tests

The CAT inspector determined that: 30/

> .....not all valve tests ensured that the post accident differential pressure (Pa) would be applied across the valve under test, since the tests did not always provide an atmospheric vent path downstream of the valve under test. If the valve under test leaked significantly, the volume downstream of that valve could pressurize, thus reducing the differential pressure across the valve. This situation would give artificially low test results. Some valves for which downstream venting was not specified are:
>
> - penetration X10A: valves G11*MOV-639 and the G11 check valves.
> - penetrations X-42/XS-5: valves E11*01V-3144, MOV-55A & B, and MOV-56A&B.
> - penetrations X-8A&B: valves E11*MOV-042A&B.

The preceding is contrary to the QA/QC measures for documentation and testing required by Criterion 5 and Criterion 11.

30/ Ibid 14, p. 19.

IV.A.13: QA/QC Breakdown 32 - Failure To Establish Corrective Action Measures

During inspection of the Service Water system, the CAT inspector found that carbon steel bolts and nuts which hold together the copper-nickel (Cu-Ni) flanges of the service water piping had corroded. Salt water and two dissimilar metals in contact caused corrosion of the bolts and nuts by electrolysis and galvanic corrosion. The inspector reviewed licensee actions to replace corroded bolts and to prevent recurrence and concluded that: 31/

> .....there was not an adequate program to identify and replace all corroded carbon steel bolts and nuts on all Cu-Ni flanges of the service water system, the corrective action taken to date has not involved appropriate levels of management, and that the problem may not have been thoroughly reviewed for reportability to NRC.

The preceding is contrary to the corrective action measures required by Criterion 16.

IV.A.14: QA/QC Breakdown 33 - Failure To Provide Controls Over Activities Affecting Quality

Criterion 2 states, in part, that the quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components,

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31/ Ibid 14, pp. 20 and 21.

to an extent consistent with their importance to safety.

Contrary to the above, the CAT inspectors determined that: 32/

> .....pipe support P42*PSST-056 on the Reactor Building Closed Loop Cooling water system was not properly maintained in its as-built configuration in accordance with drawing requirements after final inspection and acceptance, in that one of the struts was at least 5½ degrees out of the vertical, in excess of the design tolerance of 4 degrees.

IV.A.15: QA/QC Breakdowns 34 and 35 - Additional Examples of Failure to Control Design

As discussed previously in Section IV.A.1, the CAT inspectors reviewed pertinent documents and drawings and performed system walk-downs to verify that the piping system was constructed in accordance with P&IDs and the FSAR. Two additional discrepancies for the RBCLCW system were found: 33/

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

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

32/ Ibid 14, Notice of Violation, p.2.

33/ Ibid 14, page 22.

The preceding drawing discrepancies are contrary to the design control and inspection requirements of Criteria 3 and 10.

IV.A.16: QA/QC Breakdowns 36 to 38 - Failure to Maintain Adequate Housekeeping and Fire Protection

Criterion 5 states, in part, that activities affecting quality shall be prescribed by documented instructions and procedures. Likewise, Stone & Webster Engineering Corporation Construction Site Instruction 13.1, states, in part, that "Work areas shall be kept sufficiently clean and orderly so that construction activity can proceed in an efficient manner... excess material shall not be allowed to accumulate and create conditions that will adversely affect quality... Equipment and instructions for the protection from the prevention of damage by fire shall be provided ..." Contrary to the above, three examples of inadequate housekeeping and fire protection were found as follows: 34/

a) On February 12, 1982 and again on February 24, 1982, these fire hazards were identified in Fuel Oil Transfer Rooms: fuel leaking from pumps; fuel oil in drip trays, wells and buckets; combustible fumes in rooms while transfer pumps running and room vents taped closed. On February 25, 1982 welding of fuel oil transfer pump check valves in transfer room "C" was observed with no fire extinguishers present, no fire watch designated and no cleanup of fire hazards identified on February 24, 1982.

34/ Ibid 14, Notice of Violation, pp. 2-3.

b) On February 12, 1982 and again on February 24, 1982, these fire hazards were identified in Emergency Diesel Generator (EDG) Rooms: fuel oil overflowing from plastic hoses on the fuel oil day tank, fuel oil in open buckets and fuel oil on floor and foundations under enginer and generator. On February 25, 1982, welding was observed on EDG's A and C with fuel oil still under engines and generators.

c) On February 24, 1982, both rooms of the Screenwell Pumphouse were observed to have accumulated excess material and were extremely dirty. Material blocked access to electrical panels and hindered work.

The preceding breakdowns in QA/QC are also contrary to cleaning and preservation measures required by Criterion 13.

IV.A.17: QA/QC Breakdown 39 - Failure To Develop Program to Compile "As-Built" Information

The CAT inspectors reviewed documentation and held discussions with LILCO and S&W personnel to determine the adequacy of the program for revising and up-grading the drawings and other engineering design information to reflect the as-built/ as-installed condition of Shoreham plant piping. The inspectors determined that the status of the "as-built" program is as follows: 35/

> .....the program to compile "as-built" information, to incorporate this information into the design drawings and isometrics, and to resolve any deviation from original design was still incomplete.

35/ Ibid 14, p. 30.

> This area remains unresolved pending further definition of this program, and formalization of control procedures to reconcile "as-built" and "as-installed" drawings and place the corrected, approved final drawing into the plant permanent record.

The preceding is contrary to the requirements of Criterion 2 that a QA program be established at the earliest practicable time, and to the requirements of Criteria 3, 5 and 6 concerning design control measures, documentation, and document control. The CAT inspectors also found the similar incomplete status of the electrical structural aspects of the "as-built" QA/QC program as follows: 36/

> The "as-built" program for electrical systems is comprised of three parts. One part is to verify and sketch the supports for cables and raceways and identify these on drawings for stress reconciliation; according to licensee representatives, this part is about 30% complete. The second part is to sketch the supports for conduits and relate these to drawings; this part is about 10% complete. The third part is the as-built sketching of conduits; this part is not yet started.

IV.A.18: QA/QC Breakdown 40 - Failure To Control Design Changes

Criteria 3 and 6 require that design changes be controlled and that design changes be distributed to and used at the location where the prescribed activity is performed. Contrary

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36/ Ibid 14, p. 30.

to this requirement the CAT inspectors concluded that the Shoreham system of Engineering and Design Coordination Reports (E&DCR's) is deficient as follows: 37/

> Because of the number and frequent revision of these documents, the design information and requirements were fragmented into numerous E&DCRs, drawings, and specification. This fragmentation makes it difficult to use drawings and specifications unless one is quite familiar with them and their pertinent E&DCR's. A clear, concise, and timely dissemination of technical and design information is fundamental to effective and error-free execution of engineering construction. The E&DCR system as implemented at Shoreham, with the lack of timely drawing revision, does not provide such dissemination.

The CAT inspectors also found the following additional deficiencies in the current E&DCR system at Shoreham: 38/

> The inspector observed that the E&DCR system was also being used for documenting interpretations of design requirements, and site-project technical communications. This has led to a large number of E&DCR's. The inspector also noted that many E&DCRs themselves had been revised, modified, and/or augmented with additional information over a period of time without incorporation into the affected design documents. Furthermore, some E&DCRs, classified as "generic", were comprehensive primary design documents used over and over for a long period of time; these E&DCR's underwent many revisions themselves without incorporation in any drawing and/or specification. Such practices and uses of the E&DCR system, primarily intended for change control, has created a somewhat unwieldy and cumbersome system. The revision of primary design documents, i.e. drawings and specifications,

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37/ Ibid 14, pp. 31-32.

38/ Ibid 14, p. 31.

> has not kept pace with generation of E&DCRs. The drawings and specifications were posted with listings of E&DCR's affecting them. The inspector observed that, in the case of drawings FM-20A and FM-20B used for this inspection, the referenced E&DCRs numbered 26 and 21, respectively. These E&DCRs date back to June, 1978, for FM-20A and April, 1978, for FM-20B (E&DCR Nos. F-14071 and F-11993A, respectively).

Given the large number of E&DCR's at Shoreham, the foregoing deficiencies are particularly significant, indicating a lack of controls at crucial design interfaces. Such controls are essential if the goals of the QA/QC regulations are to be met. LILCO plainly has not met these goals.

IV.A.19: <u>QA/QC Breakdowns 41 and 42 - Failure To Develop Adequate Technical Specifications</u>

<u>Criterion 3</u> requires that the design be controlled in accordance with the licensing commitments in the FSAR. Likewise, the NRC requires that Technical Specifications (TS) be submitted which are derived from the analyses and evaluations in the FSAR. 39/ Contrary to these requirements, the proposed TS for the RHR and related systems which were submitted to the NRC were deficient in the following respects: 40/

> a) ....important, safety-related, plant unique features described in the FSAR were not included in the proposed TS. The TS contained neither Limiting Conditions for Operation nor Surveillance Requirements for these plant unique systems: the RBCLCW System, RHR area coolers, LPCI Motor-Generator Sets, Drywell Floor Seal, Drywell Floor Seal Pressurization System, and the Leakage Return System.

39/ See 10 CFR Part 50.36(b).

40/ Ibid 14, pp. 32-33.

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

- Not all RHR System snubbers were included, e.g. E11-PSSP-806, 831 and 902.

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

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

IV.A.20: <u>QA/QC Breakdown 43 - Failure To Conduct Activities In Accordance with FSAR Commitments For Control Room Cabinet Seismic Mounting</u>

The 18 criteria of Appendix B, and particularly <u>Criterion 3</u>, are intended to assure that the Shoreham plant is actually designed and constructed in accordance with the FSAR commitments. Additional examples where Shoreham is deviating from the preceding requirements, and thus is not in compliance with the regulations, include the following: <u>41</u>/

<u>41</u>/ Ibid 14, Notice of Deviation, p. 1.

FSAR Section 3.10.2.1.1B and Table 3.10.2.3-1 establishes approved criteria for installation of Standard Cabinets using a specified number of 5/8-inch mounting bolts. Contrary to the above, Standard Cabinet H11*PNL-608 was installed with twenty 5/8-inch bolts instead of forty bolts and Standard Cabinets H11*PNL-635 and H11*PNL-636 were each installed with eight 5/8-inch bolts instead of twelve 5/8-inch bolts.

IV.B. QA/QC Breakdowns Indicate Widespread Failure To Comply With Appendix B Criteria

The preceding 43 examples of recently discovered QA/QC breakdowns at Shoreham clearly document that LILCO and its major subcontractors did not develop and implement a QA/QC program in compliance with Part 50, Appendix B in a timely manner. Rather, the evidence is clear that there have been major QA/QC breakdowns, and the QA/QC breakdowns have continued to the present time. These breakdowns are not isolated or minor; rather, as indicated by the CAT inspection, the breakdowns include violations of at least eleven of the eighteen Appendix B criteria, including:

a) Criterion 1, concerning the responsibility for the establishment and execution of the QA program.

b) Criterion 2, concerning establishing a QA program at the earliest practicable time.

c) Criterion 3, concerning design control measures.

d) Criterion 4, concerning procurement document control.

e) Criterion 5, concerning documented instructions, procedures, and drawings.

f) Criterion 6, concerning control of documents.

g) Criterion 10, concerning control of inspection activities.

h) Criterion 11, concerning control of testing activities.

i) Criterion 13, concerning handling, storage, shipping, cleaning, and preservation of material and equipment.

j) Criterion 15, concerning control of nonconformances to prevent their inadvertent use or installation.

k) Criterion 16, concerning the identification and correction of conditions adverse to quality.

The previously described QA/QC breakdowns involve only a single basic system, the RHR, plus auxiliary and supporting systems. I conclude from these breakdowns, and from the patterns of breakdowns in Attachments 2 and 5, that there has been a serious and widespread breakdown in the Shoreham QA/QC program implementation for design and construction. Based on my experience in QA/QC matters, I also conclude that, given the pattern of QA/QC deficiencies already disclosed and LILCO's failure to detect these problems on its own, that further investigation of other safety systems will almost certainly reveal more errors and QA breakdowns. Thus, despite the fact that the errors found by the CAT group and other I&E inspectors have been identified, the QA/QC breakdowns at Shoreham lead me further to conclude that there can be no present assurance that the Shoreham plant has been designed and constructed in accordance with regulatory requirements and FSAR commitments. Rather, the breakdowns at Shoreham cast substantial doubt on the safe design and construction of Shoreham. There can be no such assurance -- and thus no basis for licensing --

unless a comprehensive program such as described in Section VII hereof is carried out, including the implementation of changes necessitated by that review.

The breakdowns which have been documented already lead me also to conclude that other QA criteria may have been violated. For example, the large number of QA/QC deficiencies which LILCO failed to detect but which were detected by the NRC in its limited inspection program lead me to conclude that the LILCO audit program, required by Criterion 18, was not effectively implemented. The LILCO audit activities required by Criterion 18 were controlled by the same LILCO and S&W Q/A manuals which have been shown to have been inadequately planned and implemented with respect to other criteria.

Finally, the Shoreham QA/QC deficiencies discussed herein are particularly significant because they went undetected by LILCO and NRC Staff inspectors for years. The NRC's regulations, particularly the eighteen QA requirements of Appendix B to 10 CFR Part 50, are specifically designed to detect such defects and thus ensure that nuclear plants are designed and constructed in accordance with all requisite requirements. The plain fact is that these defects were not discovered by LILCO or the Staff until years after most of the work had been performed and then they were discovered only because a special audit was conducted. If a proper audit/QA program had been in place, these errors would have been detected years earlier. The fact that these errors have now been found does not in any way "cure" the failure of LILCO to comply with Appendix B. The vast majority of the design and construction of Shoreham has had no special audit but rather has had only the plainly inadequate QA which allowed these errors to exist in the first place and to go undetected for years.[42/] I thus conclude:

-- LILCO did not and has not implemented a QA/QC program in compliance with Appendix B.

-- LILCO's failure to comply with Appendix B was widespread, reaching all the QA criteria.

-- There is no assurance that the facility has been designed or constructed in accordance with regulatory and FSAR commitments.

-- The errors which have been discovered to date do not "cure" LILCO's failure to implement a program in compliance with Appendix B. These errors, instead, demonstrate that there is substantial doubt concerning the actual quality achieved in Shoreham design and construction.

42/ The CAT audit of February 1982 was not a comprehensive review. It was limited to on-site inspection and did not reach crucial offsite work, such as the GE and S&W design processes. To determine whether Shoreham has been safely designed and constructed in view of LILCO's QA breakdowns, a far more comprehensive audit program must be conducted.

V. THE NRC STAFF HAS FAILED TO VERIFY THAT LILCO'S QA/QC PROGRAM FOR DESIGN AND CONSTRUCTION OF SHOREHAM HAS BEEN EFFECTIVELY IMPLEMENTED

The QA/QC program mandated by the NRC for Shoreham by 10 C.F.R. 50, Appendix B and Criterion 1 of Appendix A is intended to assure that the safety and reliability features specified by the regulations and committed to by the system designers are in fact implemented by the many different organizations involved. Successful implementation of the QA program is essential to achieving expected levels of safety. However, as documented in Section IV and in Attachments 2, 3 and 5, the facts demonstrate that adequate implementation of the Shoreham QA/QC program has not been attained.

The NRC's Inspection and Enforcement (I&E) program should provide verification that the Shoreham facilities' structures, systems, and components are designed, manufactured, installed, and operated in strict accordance with the applicable QA/QC requirements. However, the I&E program has not fulfilled this function for either Shoreham specifically or for nuclear stations generically. As a result, I am of the opinion that the Staff's apparent conclusion that LILCO has generally complied with Appendix B must be rejected and, instead, one must conclude from LILCO's QA/QC breakdowns that LILCO has not complied with Appendix B.

The limitations of the NRC Staff review of the design and construction QA/QC program for Shoreham, and for nuclear plants generically, are presented in this section of the testimony.

V.A: The NRC Staff Review of the Design and Construction QA/QC Programs for Shoreham Has Been Severely Limited

At the prehearing conference in March 1982, representatives of the County provided the Board and other parties in the Shoreham operating license proceeding with a list and tabulation of QA/QC breakdowns which the NRC I&E program had identified at Shoreham. A revised tabulation is included herein as Table V-1 and the I&E reports on which the tabulation is based are listed in Table V-2 of the testimony. The NRC Staff inspectors reported about 70 violations of or deviations from QA/QC criteria. I disagree that this is the proper number of violations and deviations which has been found. Based on my judgment and experience and a review of the referenced I&E reports, the list of QA/QC criteria of Appendix B which have been violated should be adjusted from the 71 violations of single criteria cited by the NRC to encompass an expanded list of 180 items which more properly, in my opinion, indicate the extent to which the criteria of Appendix B were not complied with by LILCO.

TABLE V-1

SHOREHAM SITE QA/QC BREAKDOWNS*

APPENDIX B CRITERIA	1974	1975	1976	1977	1978	1979	1980	1981	CRITERIA TOTALS
1									0
2					1	5			6
3			3	1	1	2	3	6	16
4									0
5	5	3	4	5	7	5	2	15	46
6	1		3		1	1		6	12
7	4		1	1			1		7
8				1					1
9	1	1		2	7	3			14
10	2	2		2	4	5			15
11			1					1	2
12		1	1					1	3
13	3	1	5	2	1	4			16
14						1	1	6	8
15		1	2	4	5	5	2	1	20
16	3		3	2		2		1	11
17					1				1
18	1		1						2
Yearly Totals	20	9	24	20	28	33	9	37	180

*Source: NRC I&E Reports, Shoreham Nuclear Generating Station. (See Table V-2)

TABLE V-2

I&E REPORTS UTILIZED IN TABLE V-1

74-03	78-02
74-05	78-03
74-08	78-05
75-01	78-06
75-03	78-12
75-04	78-15
75-05	78-16
76-01	79-02
76-02	79-04
76-06	79-05
76-07	79-07
76-08	79-12
76-09	79-16
76-12	79-24
77-01	80-10
77-05	80-14
77-12	80-15
77-17	81-01
77-23	81-02
	81-13
	81-14
	81-16
	81-22

In addition, presented herein as Attachment 5 is a summary prepared by MHB Technical Associates which lists all violations and deviations issued at Shoreham by the NRC's I&E inspectors from 1974 through February, 1982.[43/] This Attachment is organized so that each of the 71 QA/QC violations or deviations is summarized in terms of both the problem discovered and its resolution. The details and pattern of these QA/QC deficiencies raise a number of serious questions about the soundness of any Staff conclusions concerning the adequacy of implementation of the Shoreham QA/QC program.

Since 1974, the NRC has issued approximately 69 violations and 2 deviations to LILCO for Shoreham. Thus, over 70 commitments have been broken, including violations of 10 C.F.R. Part 50 Appendices A and B criteria, and 10 C.F.R. Part 50.55(e) criteria. In no year has Shoreham been free of violations or deviations. While most of the specific deficiencies have been corrected, as of April 15, 1982, 9 items still carried the status of "open" and one of these has been open for almost three years.

A major concern from the pattern of LILCO QA/QC breakdowns is the fact that the majority of these non-compliances fall into a few concentrated areas. These repeated violations of the same criteria indicate that the Staff's program has been ineffective in compelling LILCO to implement a program in compliance with NRC requirements once deficiencies have been found.

43/ The listing does not include the recent CAT inspection results which are discussed in Section IV.

controls might not be sufficient. Therefore, prior to completion of the inspection, the licensee issued an E&DCR to the valve purchase and (sic) specification, which noted the "three times" requirement for these two valves to ensure proper controls if replacement is required. (emphasis added)

IV.A.12: QA/QC Breakdown 31 - Failure To Document and Control Tests

The CAT inspector determined that: 30/

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

- penetration X10A: valves G11*MOV-639 and the G11 check valves.
- penetrations X-42/XS-5: valves E11*01V-3144, MOV-55A & B, and MOV-56A&B.
- penetrations X-8A&B: valves E11*MOV-042A&B.

The preceding is contrary to the QA/QC measures for documentation and testing required by Criterion 5 and Criterion 11.

30/ Ibid 14, p. 19.

Fifty-four of the 69 violations issued to Shoreham between 1974 and February, 1982 involved 3 specified QA/QC criteria from 10 C.F.R. Part 50, Appendix B:

(a) <u>Criterion 5</u>, "Instructions, Procedures, & Drawings"

(b) <u>Criterion 9</u>, "Control of Special Processes"

(c) <u>Criterion 16</u>, "Corrective Action"

Violations of <u>Criterion 5</u> amounted to a cumulative total of 40. While the numbers have fluctuated from year to year, there has been no year where LILCO has received less than 2 violation notices for non-compliance with the <u>Criterion 5</u> requirements. Examples of these deficiencies include: (a) incorrect records and documents; (b) instructions not followed; (c) equipment found unprotected, dirty and/or not properly maintained; (d) actions taken without approval; (e) missing parts; (f) incorrect part sizes in use; (g) wrong equipment in use; and (h) required actions not having been taken.

In addition to the high concentration of violations of <u>Criterion 5</u>, there have been an additional 7 violations to <u>Criterion 9</u>, "Control of Special Processes," issued in the same time period. Furthermore, violations to <u>Criterion 16</u>, "Corrective Action," also amount to 7 during this period. These violations, which began to occur in 1974, include 3 issued for ineffective corrective action taken and 4 issued for no corrective action taken at all. This pattern of repeated violation of the same regulatory requirements leads me to conclude that the I&E program has been ineffective in getting LILCO to implement a QA/QC program which complies with Appendix B.

The Staff's I&E program is deficient for a further reason. The Staff has not evaluated the Shoreham QA deficiencies against any objective baseline criteria to measure or quantitatively compare the effectiveness of the Shoreham quality program. Instead, the Staff appears dependent upon subjective qualitative judgment to measure program effectiveness. In addition, the I&E program has no means of determining improvements in, or the effectiveness of, corrections since no comparative measures are used.

Further, from the perspective of time invested, the I&E program in its present state simply cannot, with any degree of confidence, fulfill its intended responsibility to ensure that all licensing commitments made at the Shoreham station are, in fact, kept. For example, I&E's current audit procedure is to evaluate plant commitments by inspecting only a sampling of a variety of general areas during each I&E inspection. Thus, the fact that a violation notice has not been issued concerning a particular criterion, area, component or procedure does not mean that violations do not, in fact, exist. On the contrary, given the actual time spent and area covered by the NRC during these inspections, in comparison to the magnitude of each plant and its QA/QC program, it seems nearly impossible for the NRC to reasonably assure that a plant licensee is upholding its commitments. In the case of Shoreham, for instance, in contrast to the 7200 manhours applied by the NRC on the I&E program,[44/] LILCO and its site subcontractors have spent over 2.4 million manhours on QA/QC.[45/] Thus, the NRC has only applied 0.3% of the amount of time that LILCO has expended in the Shoreham QA/QC program. Such limited effort by the Staff inspectors makes it impossible for the Staff to conclude with any degree of assurance that LILCO has successfully implemented its QA/QC program.

While the Staff's limited effort precludes it from reaching supported conclusions about overall successful implementation of QA/QC at Shoreham, the Staff results do provide telling data about the inadequacies of the LILCO program. The NRC Staff, despite having invested so little time in review of Shoreham QA in comparison to LILCO and its subcontractors, has identified over 70 non-compliances at Shoreham. If the NRC had not performed the I&E audits, would LILCO's QA program have identified the deficiencies at Shoreham? I would think not because many of these deficiencies were discovered by the Staff only after countless opportunities for LILCO to have discovered the problems. Furthermore, would these problems have been assessed and corrected in the same manner and to the same extent if the NRC audits were not

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44/ LILCO letter to NRC, April 19, 1982. SNRC #689.

45/ LILCO/NRC Meeting on Shoreham QA Program, March 15, 1982, Bethesda, MD.

involved? Finally, since the I&E audits are such a small fraction of the total QA/QC program at Shoreham, what degree of assurance is there that the non-audited parts of Shoreham's QA/QC program do not contain serious non-conformances and/or deficiencies? More specifically, what are the implications for safety of the fact that the NRC Staff identified over 70 non-compliances at Shoreham in only 0.3% of the time that LILCO had invested in the program? I believe the meaning is clear: the LILCO audit program was ineffective and not in compliance with Appendix B, and, if proper and comprehensive audit efforts (as required by Criterion 18) were undertaken, far more deficiencies would likely be found. The current failures at Shoreham to comply with Appendix B lead to substantial doubt regarding the effectiveness of implementation of the Shoreham QA program. Such doubt, in my opinion, is intolerable in an area as important to safe operation as QA.

Finally, as discussed in Section VI, a further inadequacy of the Staff review program is that it addresses only the QA/QC applied to items classified as safety-related. Although Appendix A, GDC 1, clearly requires that QA be applied to all items important to safety -- at least to a degree commensurate with the importance of an item -- the Staff does not specifically review that aspect of the QA program and has developed no QA requirements for GDC 1 QA compliance.[46/] I believe that the Staff's failure in this regard precludes the Staff from reaching

46/ NRC Staff Contention 7B testimony, pp. 8-9.

any conclusion that LILCO's overall QA program meets regulatory requirements.

V.B: Generic Limitations and Gaps in the NRC Staff Review of Design and Construction QA/QC Programs Which Are Also Applicable to the Shoreham Review

The "after the fact" discovery by the NRC of quality deficiencies at the North Anna,[47/] Brown's Ferry,[48/] and TMI-2[49/] plants raises serious questions about the adequacy of the whole I&E program. The recently disclosed QA/QC breakdowns at Diablo Canyon, Zimmer, South Texas, Midland, and Marble Hill, as set forth briefly herein in Attachment 3, confirm that the I&E program deficiencies are pervasive and systematic, rather than random occurrences which have now been addressed by the Staff's I&E efforts, including the resident inspector program. Set forth below are

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47/ EMD-77-30, Allegations of Poor Construction Practices on the North Anna Nuclear Power Plants, U.S. General Accounting Office, Washington, D.C., June 2, 1977.

48/ Brown's Ferry Nuclear Plant Fire, Hearings before the Joint Committee on Atomic Energy, September 16, 1975.

49/ Kemeny, John, et al, The Need for Change: The Legacy of TMI, Report of the President's Commission on the Accident at Three Mile Island, Vol. IV by the Technical Assessment Task Force entitled, "Quality Assurance," Washington, D.C., October, 1979. (Also see Attachment 6 hereto.)

specific problems which have been identified with respect to the I&E program.

V.B.1: Sandia's 1976 Study

The deficiencies in the I&E program are not new concerns. In partial response to the numerous criticisms of the NRC I&E practices, in May, 1976, the NRC provided Sandia Laboratories of Albuquerque, New Mexico, with over a quarter of a million dollars of funding to conduct a comprehensive, independent assessment of the NRC activities related to the review, approval, and inspection of quality assurance programs at commercial nuclear power plants. Specifically, the study assessed the effectiveness of the overall philosophy of the NRC program and the relative strengths and weaknesses of the practices employed to assure a high standard of quality assurance for nuclear reactors.[50/]

The Sandia study's final report was released as a NUREG series document in September, 1977.[51/] While the 16 recommendations of the study group were carefully worded in a positive manner so as to not imply that the existing NRC I&E program is inadequate, the message is still clear. The report states in the summary that "based on the results of our survey and the stringent demands for reactor safety, we conclude that further improvements are warranted

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50/ NRC Press Release No. 76-122, entitled "Independent Assessment of NRC Quality Assurance Activities Planned," May 25, 1976.

51/ NUREG-0321, A Study of the Nuclear Regulatory Commission Quality Assurance Program, U.S. Nuclear Regulatory Commission, Washington, D.C., August, 1977.

in both industry quality assurance programs and NRC regulation of these programs." Specifically, the study concludes that "routine direct NRC inspection and testing of hardware [should] be increased, and that data pertinent to quality decisions made in the construction and operation of a plant [should] be evaluated by the NRC on a routine basis. This includes the evaluation, for example, of radiographic and ultrasonic test data." The Sandia recommendations have still not been fully implemented by the NRC. Thus, at Shoreham, as noted previously, the I&E effort has been too limited to permit I&E from reaching an informed conclusion that LILCO has complied with Appendix B. Rather, most I&E efforts have been confined to review of LILCO paper work. The few times that the program has examined how LILCO actually implemented QA -- such as the CAT inspection -- the conclusion has been inescapable that LILCO design and construction activities are not in compliance with Appendix B.

V.B.2: GAO Study Shows I&E Program Weaknesses Generically and At Shoreham

In 1978, a study conducted by the General Accounting Office (GAO) described the following weaknesses in the NRC's I&E program during nuclear power plant construction which may result in QA deficiencies going undetected:[52/]

52/ EMD-78-80, The Nuclear Regulatory Commission Needs to Aggressively Monitor and Independently Evaluate Nuclear Power Plant Construction, U.S. General Accounting Office, Washington, D.C., September 7, 1978.

> "Although the Nuclear Regulatory Commission is responsible for assuring that nuclear power plants are constructed safely, it has not been independently testing the quality of construction work. The Commission should do this, plus
>
> -- improve its inspection and reporting practices,
>
> -- use the inspectors' time and talents more efficiently, and
>
> -- better document its inspection findings.
>
> The Commission is aware of the need for improvements and has made some changes, one of which is the assignment of resident inspectors to selected reactors under construction."

The Salem and Shoreham nuclear stations from NRC Region I were two of the six construction sites selected by GAO for review. The following chart summarizes the results of GAO's detailed review of selected NRC report items for Region I:[53/]

NRC Region	Powerplant	No. Items Reviewed	Items Deficient	
			Number	Percent
I	Salem	5	5	100
	Shoreham	10	8	80
	Total:	15	13	87

The nature and number of the thirteen deficiencies are shown below:[54/]

53/ Ibid 52, p. 13.

54/ Ibid 52, p. 13.

	Nature of Deficiency	Number
a.	Inadequate Reporting	7
b.	Inadequate Attention to Detail	1
c.	Acceptance of Inadequate Licensee Action	3
d.	Inadequate Investigation	2
	Total:	13

Thus, as early as 1978, the NRC was on notice that the I&E inspections at Shoreham may have significant weaknesses.

V.B.3: President's Commission Review of QA Including Its Assessment of The Region I Program

A recent investigation of the adequacy of QA program implementation was conducted by the President's Commission on the Accident at Three Mile Island.[55/] Some of its key insights related to quality assurance implementation deficiencies are summarized in the "Findings and Conclusions" section of the QA Task Force report which is appended as Attachment 6 to this testimony.

The President's Commission review of the independent assessment program for nuclear power plants, as defined by NRC quality assurance regulations and requirements, was accomplished by examining the major elements of the NRC and one of its five regional

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55/ Also see, Inskeep, G. W., "The Cause and Effect at Three Mile Island," Quality, June, 1980, pp. 42-45.

offices (Region I) and one utility company (Met Ed). This somewhat limited review resulted in two general conclusions by the President's Commission Task Force on QA as follows: 56/

(a) . . . the overview and independent assessment performed by NRC were limited only to those items which were identified as safety-related, including intensive analyses of recovery from postulated accidents which resulted in a narrow overview of the utility. This narrow view was further confined by the application of the forerunner of the Standard Review Plan which programmed the review effort by NRC to carefully defined areas. Further, this narrow and confined review was bothered by a focusing problem brought about by doubts about the interpretation and application of the term "safety-related" to equipment; this further affected related procedures, inspection, maintenance and problem resolution. Combining this narrow view with a weak NRC-to-utility management interrelationship, left voids that prevented the NRC from knowing the "health" of the utility. More important, the NRC did not have an independent assessment activity to "tell them that they didn't know." (emphasis added)

b) .....(the) utility joined the NRC's narrow and confined view on the safety items and virtually ignored other vital parts of plant operation..... These other parts were those whose performance not only supported the safety-related items, but were those that were also vital to assuring that the plant would reliably perform. This illustrated that the utility management had not exhibited the desire or capacity to go beyond the NRC requirements to provide a well-designed, maintained, and staffed plant capable of realiable performance that would not jeopardize the health and safety of the public and its own workers. Like the NRC, the utility management had no independent assessment system to tell them that their plant was "sick". (emphasis added)

These findings are directly applicable to Shoreham. The Staff, as noted previously, has narrowly confined its QA review to safety-related items. It has thus ignored a wide range of systems to which GDC 1 applies and thus leaves a void regarding the actual quality of site activities achieved at Shoreham.

56/ Ibid 49, pp. 105-106.

V.B.4: Staff Review of Design Activities Has Been Inadequate

In addition to the inadequacies in the Staff's site and construction inspection program, the NRC Staff also has conducted virtually no indepth review of the implementation of the design process at GE and Stone & Webster, the designers of Shoreham safety features. I believe this is a significant omission in the Staff's NRR and I&E program reviews. 57/

57/ The weaknesses in the Staff's I&E program and FSAR design reviews were acknowledged by Harold Denton, Director of NRR, Richard Volmer, Director of NRR Division of Engineering, and Richard DeYoung, Director of Office of Inspection and Enforcement, at a Commission briefing on QA/QC effectiveness on January 29, 1982 as follows:

(a) One of the areas that we never inspected very heavily was at the architect/engineer's design office. We did a lot of inspection, I think, over the years of the quality construction assuming that the blue prints were correct, and the main gap we have been focusing on in these first two cases (San Onofre and LaSalle) is that gap of whether the drawings themselves properly reflected the design commitment in the application. (Denton at Tr. 31).

(b) Vic Stello has again and again, tried to point out the problem we have with QA is the gap in the implementation of design. (DeYoung at Tr. 32).

(c) We have . . . traditionally accepted the design drawing and said "if it is built that way, it is okay." We spent very little time looking at implementation of the design criteria and bases to get to that drawing and that is where the gap is. (DeYoung at Tr. 33).

(Cont'd on next page)

In response to the serious quality assurance problems that have arisen during the past few years at a number of plants, William Dircks, Executive Director for Operations for the NRC, in a January 21, 1982 letter to NRC Chairman Palladino, outlined various corrective actions being considered by the Staff to strengthen quality assurance. In the area of design management, the Staff is now considering a number of initiatives to verify the proper management and implementation of the design for nuclear power plants. Alternatives under consideration include:58/

57/ (Continued)

(d) Very seldom do we see detailed drawings in areas of electrical . . . some detailed schematics are looked at on an audit basis. In mechanical engineering, for example, we will take sample typing problems and run verification with our consultant codes to see that we get about the same answers as the licensees calculations for stress points and hanger locations. The structural branch goes out and does an audit and takes selected areas and looks in detail, but that is a small sample . . . I think we are trying to weave QA into our method of [FSAR] review more than we have before, but it is still a small sample and many of the plants designed have been designed many years ago when we didn't do that. (Volmer at Tr. 34).

(e) I think in years past it was kind of assumed that the architect/engineers were professional and that they had their own internal processes and checkers, and that was not one we needed to spend as much time in as other areas so historically, the focus has been on the criteria and the code rather than on the results of some engineering. (Denton at Tr. 35).

58/ Letter, Dircks to Palladino, January 21, 1982, p. 5.

(a) The development of criteria to employ contractor design audits at selected sites in the construction phase. The audits would provide additional confidence in the proper implementation of the design in questionable areas.

(b) Revising vendor and construction inspection programs to include inspection elements to be used to verify the proper management and implementation of design.

(c) Assignment of resident inspectors at selected major vendors and architect engineers.

(d) Some type of accreditation of design organization, including those of nuclear steam supply, system suppliers, and architect engineers.

In addition, NRC is revising its vendor and construction inspection program to be more effective. In the revised program:59/

> Considerations are being given to increase NRC emphasis on inspection of design and design interfaces or reducing the review of documentation associated with QA programs.

While the preceding initiatives should provide valuable insight in the future, it is clear that none of these initiatives is planned to be applied to Shoreham. The result is that the Staff's review process actually applied to Shoreham fails to

59/ Letter, Dircks to Palladino, January 21, 1982, p. 6.

provide the needed assurance that LILCO's FSAR commitments are, in fact, properly reflected in Shoreham design documents. Thus, the Staff can provide no assurance that the design QA/QC requirements have been complied with as required by the regulations.

V.C: <u>Conclusion: The Staff Review of Shoreham QA/QC Does Not Provide Assurance that the Facility Has Been Designed and Constructed in Accordance with Regulatory Requirements</u>

I conclude that the Staff I&E program has been inadequate to provide assurance that the design and construction of Shoreham, despite the severe pattern of QA/QC breakdowns, are adequate and in compliance with regulatory requirements. The Staff program has revealed a pattern of repeated breakdowns in QA/QC at Shoreham. These breakdowns, in my opinion, document a failure by LILCO to implement a design and construction QA program in accordance with Appendix B and Appendix A, GDC 1.

The Staff program, however, has failed to ensure corrective action to eliminate future violations. The repeated violations of Criteria 5, 9 and 16, plus the host of new violations discovered in the CAT inspection, document that no effective remedial actions have been taken by LILCO.

I emphasize my technical conclusion that the QA/QC deficiencies disclosed at Shoreham should not be characterized as minor failures in the dotting of "i's" and crossing of "t's". Rather, <u>the documented Shoreham QA/QC deficiencies were widespread and per-</u>

vasive, as contrasted with deficiencies requiring narrow fine tuning of a fully functioning and implemented QA/QC program.

The foregoing situation, in my technical opinion, leaves the Shoreham facility in a position where I cannot support operation of the plant absent means to determine whether, notwithstanding the QA/QC breakdowns, the plant is designed and constructed in accordance with regulatory requirements. QA/QC should provide such assurance but where, as at Shoreham, widespread breakdowns have occurred, a void necessarily is present. Thus, I believe the plant must not operate until LILCO demonstrates that, notwithstanding the breakdowns, there is a sound basis to permit the facility to operate. In my opinion, the only means by which LILCO could provide such a demonstration would be to carry out a complete design and construction review as described in Section VII of this testimony. This review is essential to:

-- Determine the extent of QA/QC breakdowns;

-- Clearly identify root causes of the breakdowns;

-- Demonstrate necessary remedial actions in terms of alterations to the facility design and construction to eliminate the errors which have been allowed to exist; and

-- Provide the only factual evidence of whether the plant is built so that it can be operated safely.

Until the foregoing actions are taken, I am of the opinion that LILCO has not demonstrated and cannot demonstrate that the plant has been designed and constructed in accordance with regulatory requirements.

VI. LILCO'S OPERATIONS QA/QC PROGRAM FAILS TO COMPLY WITH REGULATORY CRITERIA

In the preceding Sections of this testimony, evidence is presented concerning LILCO's non-compliance with QA/QC program requirements for design and construction activities and the NRC's review of LILCO's QA/QC activities. The testimony was, by its nature, retrospective. In contrast, this portion of the testimony addresses LILCO's operations QA/QC program. Thus, the following testimony will address activities that have not yet occurred, but for which a QA/QC program will be required.

VI.A: The Importance of Operating QA/QC Program Implementation

In addition to assuring the quality of design and construction, it is equally important and necessary to ensure that LILCO's operating quality assurance program, including operating procedures, quality verification activities, and maintenance procedures are in place and are adequate to assure safe operation. Such a QA program is essential if quality design, construction, maintenance and operation is to be maintained over the entire life of the plant. LILCO's operating QA/QC program implementation is particularly important since, unlike during construction where LILCO essentially provided an audit function,[60/] during operation

60/ For design and construction QA/QC activities, LILCO essentially delegated the QA/QC functions to GE, S&W, and their subcontractors, with LILCO providing an audit function.

LILCO personnel will be directly responsible for conducting a majority of the QA program activities.

In short, it may do little, or absolutely no good, to spend millions of dollars to design, inspect and test Shoreham's safety systems if they can be defeated by a forgotten or improper maintenance procedure, bypassed by improperly qualified test personnel, or degraded by improper selection or installation of replacement parts and materials.

My review of selected aspects of Shoreham's operating QA program is presented in succeeding sections. This review concludes that the LILCO operating QA program is not in compliance with regulatory requirements and, accordingly, it provides no assurance that the facility can be safely operated.

VI.B: Deficiencies in Shoreham Operations QA/QC Program

As with design and construction QA, the critical measure of the adequacy of LILCO's operations QA program is whether it is properly implemented. More than a paper commitment to a paper program is required.[61/] Thus, in my opinion, LILCO must provide objective evidence of effective implementation of the operations QA/QC program. As will be set forth in the following discussion, LILCO has neither developed nor implemented a QA/QC

61/ As discussed later, even on paper the operations QA program for Shoreham has severe deficiencies.

program for operations in compliance with the regulatory requirements and in accordance with the license application (FSAR) commitments.

VI.B.1: The Shoreham Operations QA/QC Program Is Not Defined

LILCO and the NRC Staff have not demonstrated that the QA/QC program for operations at Shoreham, as described by LILCO in Section 17.2 of the FSAR, is in compliance with regulatory requirements.[62/] Nor has LILCO demonstrated that the Shoreham QA/QC program description in Section 17.2 of the FSAR is documented in accordance with the methods described in Regulatory Guide 1.70, Chapter 17, dated October 1975, entitled, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition."

The LILCO QA Program description in Section 17.2 of the FSAR lacks specificity and, in general, fails to describe how the QA/QC program elements will be accomplished. For example, the LILCO program to control special processes, as described in Section 17.2.9 of the FSAR, is cursory and lacks the specificity required to demonstrate that the Shoreham program complies with the measures required by Criterion 9 of Appendix B. First,

62 / Appendix B to 10 C.F.R. Part 50, and GDC 1 of Appendix A to 10 C.F.R. Part 50.

the FSAR does not even attempt to identify the criteria which will be applied to define a special process. In my experience, a special process can be defined as follows:

(a) The process affects or measures safety functional aspects of an item;

(b) The results of the process are highly dependent upon the control of the process or the skill of the operator or both;

(c) The specified quality cannot be readily determined by inspection or tests (particularly the long-term operation).

Second, the FSAR provides little insight into what processes beyond those covered by the ASME will be controlled. Examples of special processes which should be encompassed by the Shoreham QA/QC program for operation include soldering of circuit boards and crimping of electrical terminations. Other examples include certain chemical (plating, etching, anodizing, cleaning, milling), metallurgical (heat treating, surface hardening, welding, brazing, soldering), organic (surface treating and special finishing or coating), bonding, sealing, and filling processes. Third, the program for the certification, inspection, authorization, and period verification of such processes to the special degree necessary for these work functions should be described in detail in the FSAR.

The QA program should define who is responsible for the control of special processes, what these processes are, when they will be initiated, and how the activity is to be accomplished. Because of the absence of such detail in Section 17.2.9 of the FSAR, it is not possible to conclude that the LILCO QA/QC program for defining, selecting, and controlling special processes and special process operators for non-ASME code processes which is to be applied during the operation of Shoreham to safety features will "assure that special processes . . . are controlled and accomplished by qualified personnel using qualified procedures" in accordance with the requirements of 10 C.F.R. 50, Appendix B, Criterion 9.

LILCO's lack of specificity in the FSAR is in direct conflict with regulations concerning the content of applications[63/] which require that the Applicant "shall include a description of <u>how</u> the applicable requirements of Appendix B will be satisfied." (Emphasis added). In addition, Regulatory Guide 1.70, Chapter 17, requires that "the SAR addresses, at a minimum, each of the criteria in Appendix B in sufficient detail to enable the reviewer to determine <u>whether and how</u> all the requirements of the Appendix will be satisfied." (Emphasis added). LILCO's cursory recitation of the 18 criteria of Appendix B fails to comply with either the intent of the regulations or with the normal NRC Staff practices.

63/ <u>See</u> Paragraph 6(ii) of 10 C.F.R. 50.34(b) for FSAR requirements.

In many SARs docketed since 1975, the SAR description of the QA Program is supplemented by a more detailed topical report which is written in sufficient detail to enable an independent reviewer to assess how the QA/QC program is being implemented. Clearly, if a detailed topical report is not submitted by LILCO, the FSAR description of the operations QA/QC program must be expanded to meet both the letter and the intent of the regulations. Until such effective action is taken to explain how the operations QA program will be implemented, there can be no finding that LILCO complies with regulatory requirements.

In addition to the foregoing FSAR deficiencies, I conclude that the current status of the LILCO operations QA/QC program is very confusing. This confusion is additional reason for finding that LILCO does not comply with regulatory requirements. First, during discovery in April, 1982, it was determined that there are two LILCO operating QA/QC manuals -- a LILCO operational QA Manual, Revision 5, dated November 15, 1978 and a draft undated LILCO QA Manual for operations with a corporate statement dated December 1, 1981. The existence of two QA/QC program descriptions has been known to LILCO management for some time. For example, the audit report for "The Combined Utility Assessment of the Adequacy of the LILCO QA Program For Nuclear Application," dated February 27, 1981, and included herein as Attachment 7 observed that:[64/]

64/ See Attachment 7, p. 6-8.

(a) "LILCO should act to establish one QA Program. The QA Manual presently in Draft form, subsequent to resolution of comments is to be used as a basis for a FSAR Amendment and will then allow procedure issues in a timely manner prior to licensing hearings.

(b) The "Draft" QA Manual has been written to replace the present EQA and OQA manuals. The review cycle for the Draft Manual is not complete. The EQA and OQA manuals infer that two QA Programs exist at LILCO. In reality, LILCO is working toward an operational status with one QA Program."

The outside auditors recommended that LILCO should: [65/]

(a) "Utilize the QA Manual as the QA portion of the FSAR.

(b) Submit the QA Manual as a Topical Report to the NRC for approval. An approved Topical may then be utilized as the QA portion of the FSAR."

I agree with these recommendations. However, neither of these recommendations has been implemented by LILCO to date. Further, it is not clear that the other numerous and detailed recommendations of the audit team have been implemented since the last change to the Operating QA Manual occurred in 1978. Due to the multiplicity of Operating QA Manuals, and their uncertain status, neither of the manuals has been reviewed in detail for this testimony.

However, in one respect the LILCO operating QA program appears clearly deficient in scope and it is impossible for an independent reviewer to assess that it is adequate. Criterion 2 of Appendix B requires, in part, that LILCO should identify the structures, systems, and components to be covered by the quality assurance program. GDC 1 of Appendix A requires, in part, that structures, systems and

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65/ Ibid 64.

components important to safety should be designed, fabricated, erected, and tested to quality standards commensurate with the safety functions to be performed. Contrary to this requirement, the LILCO operating QA/QC program addresses only the so-called "safety-related" items.[66/] The two LILCO operating QA program manuals appear to address only the safety-related items presented in FSAR Table 3.2.1-1, rather than all items important to safety, as required by the regulations.

For example, the new "draft" QA manual states that LILCO's operating QA/QC is applicable as follows:[67/]

> FSAR Table 3.2.1-1, Equipment Classification, identifies safety-related structures, systems and components as QA Category I. These, and associated consumables such as welding materials, nuclear fuel, diesel fuel oil, etc., are subject to the requirements of the QA Program. (emphasis added).

The 1978 Operational QA Manual applicability was defined by LILCO as follows:[68/]

> The Operational Quality Assurance Program is applicable to station operations such as operating, maintaining, repairing, refueling, modifying, inspecting and testing of safety-related structures, systems, and components. Safety related structures, systems, and

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66/ See definitions of "safety-related" and "important to safety" included as Attachment 1 to the County's direct testimony on County Contention 7B. The definitions will not be repeated herein for the sake of brevity.

67/ Paragraph 2.1.4 of "Draft" QA Manual.

68/ Paragraph 2.2 of Operational QA Manual.

> components covered by this program are identified in Appendix B of this manual. (emphasis added). 69/

The list in Appendix B of the LILCO QA manual also appeared to be directly extracted from the FSAR Table 3.2.1-1.

Thus, LILCO's operating QA/QC program, and the associated NRC review to date, as set forth in Section 17 of the Shoreham SER, appear to have ignored items important to safety, but not safety-related. Further, the list of safety-related items in the FSAR lacks sufficient detail to assure uniform identification of the safety-related items. For example, some FSAR categories are too broad (i.e., electrical modules) and in other cases the FSAR list is incomplete (i.e., remote shutdown panel). The result is that there is no assurance that the QA/QC requirements for operations will be applied by LILCO personnel on a uniform and consistent basis in the disciplined manner required by Appendix B and GDC 1 of Appendix A.

The NRC Staff review of the operating QA/QC program is also severely flawed. First, the NRC review of the Shoreham operational QA/QC program for items important to safety, as well as the design and construction review for these important items, is non-existent. The preceding deficiency in the Staff review was acknowledged by the Staff witnesses in testimony in this proceeding on Contention 7B, as follows:

69/ In LILCO's Further Response to Suffolk County Interrogatories in April 1982, LILCO acknowledged the limitations of the operating QA/QC program in Response 22 as follows:

> "The QA Program is applied to the safety-related structures, systems and components listed in FSAR Table 3.2.1-1. Activities affecting these safety-related structures are considered quality-related."

> The staff's present review process does not require that this subset [items important to safety but not safety-related] be specifically identified in a listing, nor has the staff developed quality assurance requirements, analogous to Appendix B, for these items. The staff simply requires an applicant to commit to meeting the provisions of GDC 1 and has permitted applicants to determine the appropriate quality assurance requirements for these items consistent with their importance to safety. 70/

Thus, the Staff can provide no assurance that the QA/QC requirements of GDC 1 of Appendix A have, in fact, been implemented. This is a significant deficiency in the Staff's review of Shoreham's safety features.

Second, NRR personnel in Bethesda have limited their review to the cursory QA/QC program description provided in the FSAR. 71/ No review of LILCO's implementing QA/QC manual (or manuals) has been conducted by NRR, 72/ and thus, in my opinion, the NRR review can best be categorized as a "desk-top" assessment. For example, the fact of two LILCO operating QA manuals is not even mentioned in the Shoreham SER.

70/ Staff 7B testimony, p. 9.

71/ Personal discussions with John Gilroy of NRR during informal discovery at NRC Offices in Bethesda on March 16, 1982.

72/ Ibid 71.

The Region I personnel plan eventually to review the status and adequacy of LILCO's operating QA/QC program procedures. However, as of late March, this Region I review had not yet taken place.[73/] Thus, there can be no assurance based on Staff review that the LILCO operating QA/QC program is, in fact, in compliance with the regulations and is capable of being implemented.

VI.B.2: Operations QA/QC Staffing Must Be Augmented

The Operational QA organization "now consists of eight permanent LILCO personnel, supplemented by seven contract personnel."[74/] The value of replacing the contract personnel was recognized in 1981:[75/]

> Consideration should be given to staffing the OQA function with LILCO employees rather than S&W. Permanent LILCO employees continuing thru operations would eliminate future retraining.

The difficulty in accomplishing this transition was also recognized:[76/]

73/ Personal discussions with Region I personnel, including the Shoreham resident inspector, during informal discovery at NRC Offices in King of Prussia on March 23, 1982.

74/ Attachment C to LILCO letter SNRC 689 to NRC, dated April 19, 1982, p. C-3.

75/ Ibid 64, p. 6-4.

76/ Ibid 64.

> Continue the effort to add required QA personnel to ensure all aspects of the quality program requirements are covered. The QA Department is actively pursuing recruitment of additional quality assurance people, and are experiencing difficulty in the selection of candidates which is somewhat due to inconsistencies in present salary structures in relation to industry scales, the local cost of living and housing costs and a highly competitive labor market for qualified QA people.

The limited staffing and qualifications of LILCO's operating QA/QC staff were acknowledged in the March response to interrogatories as follows:77/

> LILCO Operational Quality Assurance will be staffed with an Operating QA Engineer, QA Engineer, QC Engineer and 5 QC Inspectors during the regular work day. As workload requires (i.e., during scheduled, major maintenance overhauls or scheduled refueling outages) OQA personnel will work scheduled overtime as necessary. During any emergency work at the station, OQA personnel will be on-call to provide QA/QC coverage. It is expected that during station refueling outages, OQA may require outside QA/QC contractor assistance. Outside QA/QC contractor personnel will be qualified and certified to the LILCO QA program.
> . . .

77/ LILCO's March 26, 1982 "Response" to Suffolk County Interrogatory 11.

Further, LILCO also acknowledged that no QA/QC personnel are normally scheduled for shifts other than the "regular work day."

The three key LILCO QA/QC personnel have "QA/QC or Nuclear" experience of 7 years, 2 years, and 5 years respectively.[78/] The QA/QC experience, and its relevance to an operating QA/QC program, were not discernable from LILCO's interrogatory responses, but even if all 14 years of experience are relevant, the staffing proposed by LILCO still appears to be inadequate in both number and qualifications. Finally, the fact that [79/] "the Operating Quality Assurance Engineer reports administratively to the Plant Manager but takes quality direction from the Quality Assurance Manager during construction and startup testing" suggests that the LILCO organization fails to comply with the requirement of Criterion 1 of Appendix B that "persons and organizations performing quality assurance functions shall report to a management level such that this required authority and freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided."

VI.B.3: Operating QA/QC Feedback System Just Being Implemented

In response to the lessons learned following the TMI-2 accident, and the resulting requirements from the Staff set

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78/ Ibid 77, Attachments to Response to Interrogatory 10.

79/ Ibid 74.

forth in NUREG-0737, LILCO has committed to organize an Independent Safety Engineering Group (ISEG) to control the feedback of operating experience. However, during informal discovery with NRR and Region I personnel of the NRC in March, 1982, it was confirmed that LILCO's ISEG was not yet in place. Further, neither NRR nor Region I had yet reviewed any of the procedures which will be utilized by ISEG regarding operational feedback.

Therefore, contrary to the requirements of Criterion 16 of Appendix B that the cause of significant conditions adverse to quality be determined and corrective action taken to preclude repetition, LILCO has not yet demonstrated and the NRC has not yet evaluated the Shoreham operating feedback system. Thus, it cannot be assumed that LILCO has established appropriate QA/QC measures for the identification, reporting, and analysis of all equipment failures discovered during operation and maintenance at Shoreham and at other operating BWR stations with similar equipment.[80/]

VI.B.4: Control of Replacement Materials and Parts Not Defined

LILCO's QA/QC program for operations is particularly inadequate as it applies to replacement parts and materials

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80/ I am informed that during the week June 21-25, 1982, certain LILCO draft ISEG materials were distributed by LILCO. I have not had an opportunity to review these data.

which are included within items "important-to-safety" or "safety-related."[81/]. As used in this testimony, the terms "replacement" and "spare" parts are the same and hereafter shall be referred to as parts. Parts as described herein include the following:

1. Original part from original vendor (replacement in kind).
2. Updated part from original vendor.
3. Off-the-shelf part (commercial grade).
4. Repaired part (Owner/others).
5. Manufactured part (Owner/others).

My concern with this aspect of the LILCO program focuses on which of the 18 criteria of Appendix B apply to the design, fabrication, installation, and testing of parts of safety equipment to assure that the replacement will be performed in a manner to achieve an equivalent level of safety as that provided by the original equipment. The LILCO QA/QC program manuals for operation of Shoreham provide an inadequate (indeed almost no) description of the measures to be implemented to assure that items selected and utilized as replacement parts for important safety equipment will not degrade the safety and reliability of the plant. This is a significant omission. The purpose of maintenace and replacement under the QA program is to ensure that quality consistent with Appendix B and Appendix A, GDC 1, is maintained. LILCO's program does not provide that assurance.

81/ The two LILCO QA Manuals for Operation apply only to safety-related items, and, thus, do not address at all the requirements for the replacement of parts of non-safety-related items.

In my opinion, LILCO must institute a systematic concept of classifying characteristics of parts. Key areas in such a system are outlined in Table VI-1. There are many systems for classifying characteristics - such as Class 1, 2, 3, or Critical, Major, Minor and Incidental - but only one will be illustrated here. This is the traditional system used in the military and in many industries. One advantage of this system is that many people are familiar with the concept and are experienced in using it. The following definitions are taken directly from Military Standard 105:

a) <u>CRITICAL DEFECT</u>: A critical defect is a defect that judgment and experience indicate is likely to result in hazardous or unsafe conditions for individuals using, maintaining, or depending upon the product; or a defect that judgment and experience indicate is likely to prevent performance of the tactical function of a major end item such as a ship, aircraft, tank, missile, or space vehicle.

b) <u>MAJOR DEFECT</u>: A major defect is a defect, other than critical, that is likely to result in failure, or to reduce materially the usability of the unit of product for its intended purpose.

c) <u>MINOR DEFECT</u>: A minor defect is a defect that is not likely to reduce materially the usability of the unit of product for its intended purpose, or is a departure from established standards having little bearing on the effective use or operation of the unit.

TABLE VI-1

SYSTEM FOR CLASSIFYING CHARACTERISTICS

How Product Used:	How Product Works:	How Product Manufactured:
IDENTIFY END PRODUCT CHARACTERISTICS		
Function Features Aesthetics Safety Maintenance Service Shipping Installation Fit to Environment	LOGIC TREE STRUCTURE OF PART CHARACTER-ISTICS THAT CONTRI-BUTE TO HOW PRODUCT WORKS	LOGIC TREE STRUCTURE OF PROCESS PARAMETERS INFLUENCING PART CHARACTERISTICS

Thus, parts with characteristics that might cause "critical defects" require special emphasis in the QA/QC program, such as imposition of the 18 criteria of Appendix B. Correspondingly, the system provides a rational basis for applying a lesser degree of QA/QC to parts which might only cause a minor defect.

Based on the foregoing, I conclude that LILCO has failed to demonstrate that QA/QC measures for replacement materials and parts of items important to safety or safety-related will be equivalent to the original equipment, that replacements will be installed in accordance with adequate process procedures, and that the repaired or reworked item will be adequately inspected, tested, and documented in as-built drawings.

VII. NEED FOR A COMPLETE INDEPENDENT DESIGN REVIEW AND PHYSICAL INSPECTION OF SHOREHAM SAFETY FEATURES

In the foregoing sections I have described the basis for my conclusion that LILCO has not complied with QA/QC regulatory requirements and FSAR commitments. In my opinion, this situation leads to the conclusion that either the plant should not be operated or a comprehensive program needs to be undertaken to verify that the design and construction have been implemented in accordance with FSAR commitments and regulatory requirements.

This portion of the testimony addresses the outlines of the kind of comprehensive review program which is necessary if LILCO desires to attempt to overcome the QA/QC breakdowns which preclude any present confidence in the quality of Shoreham design and construction. Thus, the program must document the actual quality of all installed structures, systems, and components at Shoreham. In my opinion the only effective means to do so would be through a comprehensive design verification and physical reinspection of electrical and mechanical systems. Guidelines for the initial phase of such an independent physical reinspection and design verification audit program are presented in this section of the testimony.

The Chairman of the Board of LILCO in an April 16, 1982 letter to the Suffolk County Executive, which responded to an earlier letter from the County Executive, announced that LILCO would undertake a full independent physical inspection of all safety features of Shoreham. An outline of LILCO's plan for the proposed independent inspection was presented to representatives of the County in a meeting on May 12. The LILCO press release and the viewgraphs used in the May 12 presentation are included herein as Attachment 8. Separately, in an attachment to an April 19, 1982 letter to the NRC, LILCO proposed to conduct an independent design review of one piping system to demonstrate that all aspects of Shoreham's safety-related piping design for the Core Spray System have been properly performed from both a technical and QA standpoint. LILCO's description of the independent design review of the Core Spray System is included herein as Attachment 9. Set forth in the following is a brief summary of my major comments and criticisms on the proposed scope, content, and administrative procedures for such a design verification and construction inspection program for Shoreham. As documented in these comments, my conclusion is that the present scope of the LILCO program is too limited to document the actual quality of the Shoreham design and construction.

VII.A: <u>Purpose and Scope of Independent Audit Program</u>

The overall purpose of the audit program should be to conduct an independent review of the safety features of Shoreham from the NRC-approved design basis to implementation by the constructor or installer. Thus, the scope of safety features included in the program should be sufficient to enable an independent consulting firm to assess the adequacy of the Shoreham plant design, construction and installation processes. The scope of the safety features to be included in the program should be based on the thirty-two [32] Shoreham safety systems as proposed by LILCO.[82/] In addition, other related Category I safety features, as set forth by LILCO in the FSAR Table 3.2.1-1, should also be appropriately evaluated. The foregoing safety features should be evaluated for compliance with the design and construction commitments provided by LILCO in the Operating License Application for Shoreham as presented in the FSAR and, when appropriate, the commitments provided in the construction permit as set forth in the PSAR.

The study elements for the independent inspection program should be grouped in three major tasks to provide a complete review of all critical design and construction activities. Task A should cover the area of design control and design verification. The major goal of Task A should be to assure that the design process results in a design in conformance with the Shoreham License Application. Task B should cover the site inspection of the as-built plant to assure compliance with design and quality requirements. The major focus of Task C should be to assess whether and how the Shoreham QA/QC

82/ A list of Shoreham's 32 safety systems, as presented by LILCO at the May 12 meeting, is included herein as part of Attachment 8.

programs assure that design, construction and installation activities are conducted in accordance with the applicable safety requirements. In addition, the underlying or root cause of any deficiencies disclosed in the first two Tasks should, if feasible, be determined in Task C. The content of the LILCO proposal, as suggested by the County following the May 12 meeting, should be restructured as set forth in the following testimony.

VII.A.1: Design Control and Design Verification Program Outline

The Design Control and Design Verification Program should address the procedures, controls, and practices concerning the development, accuracy, transmittal and use of design information, both within LILCO and within each major design contractor's organization,[83/] as well as the transmittal of design information between LILCO and each design contractor. The Program for Task A should include a review of all procedures used in the design process to determine that the basic process is adequate, a review of the points where the procedures should

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83/ In describing this task, it is assumed that the design work for safety features was performed by LILCO, Stone & Webster, or General Electric. If this is not the case, and other organizations also performed significant safety features design work, then those organizations should be identified and their design control and design verification process evaluated.

have been implemented to assure that they were, in fact, practiced, and a technical review of the final design documents which were the products of the design process. Further, the review should include performance of a suitable number of calculations related to each contract and within LILCO to verify the adequacy and accuracy of the design process for the selected structures, systems, and components.

LILCO has proposed such a design review for one mechanical system, the S&W designed Core Spray System, in an April 19, 1982 letter to the NRC.[84] LILCO should supplement the Core Spray System review with a similar independent evaluation of the design of at least one additional mechanical system (the GE designed Reactor Recirculation System is suggested) and two electrical systems (the GE designed Reactor Protection System and the S&W designed Emergency Diesel Generators System or Heating, Ventilating, and Air Conditioning System are suggested.)

The scope of systems included in the design review should, of course, be expanded to include additional verifications if deficiencies are detected in the initial sample of four systems. Sufficient additional independent verifications should be conducted to ensure that no significant errors are propogated

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84/ LILCO's April 19, 1982 proposal to the NRC for an independent design review of the Core Spray System is included herein as Attachment 9.

through any part of the Shoreham design and construction. Quantitative techniques should be used to assure that the audit results can be correlated statistically to provide verification of the adequacy of the design QA system to detect design errors or to distinguish between random errors and systematic failures.

VII.A.2: Program Outline for Inspection Of As-Built Plant

The inspection of the as-built plant should include a physical walk-down, re-inspection, and tests, where applicable, of the selected Shoreham structures, systems, and components to assure their compliance with the applicable design criteria. Thus, the objective of Task B should be to verify, to the maximum extent possible with nondestructive visual examinations, the proper construction and installation of safety features. The re-inspections, tests, and reverification walk-downs should encompass work performed by all major site contractors. In addition, an evaluation should be provided for all areas of quality workmanship and materials publicly alleged as not meeting the required standards by site employees.

The proposed program, as outlined by LILCO on May 12, appears to satisfactorily address this task with the clarification that electrical conduits and raceway systems should receive an equivalent review to that proposed for piping systems.

In addition, while sample sizes may not be picked on a statistical basis, statistical methods should be combined with engineering judgment to the maximum extent possible in selecting samples, and in evaluating the significance of the findings. Further, any samples should be representative of the total population, including appropriate consideration of such factors as coverage of each safety system, each building, each contractor, each process, and other key attributes. Where sampling is utilized, the bases for criteria proposed to be used for the selection of a suitable number of samples to be performed under this program, as well as the criteria to be used for expanding the sample size based upon the results of the initial samples, should be documented and reviewed. Finally, as stated previously for the design review, the physical audit should be sufficiently detailed to provide statistically valid data to permit projection of the audit results to safety features beyond those structures, components, and QA records reviewed.[85/]

VII.A.3: Program Outline for QA/QC Review

In addition to the programs planned or proposed by LILCO, a separate QA/QC audit for Shoreham should be conducted which complements the programs described in Sections VII.A.1 and

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85/ For example, at Marble Hill the NRC required that sampling should be sufficient to achieve at least 95 percent assurance of 95 percent reliability (less than 5% errors).

VII.A.2. The assessment of the effectiveness of the Shoreham QA/QC program should include a thorough review of the QA/QC program used by LILCO and its major subcontractors in the design, construction and installation of systems at Shoreham. Suggested guidelines for such a review are set forth in Attachment 10.

The Task C review should determine whether and how the QA/QC programs comply with each of the 18 criteria of 10 CFR 50 Appendix B and the guidance provided in the ANSI Standards and Regulatory Guides applicable to Shoreham design and construction activities. In addition, an investigation should be conducted to the depth and extent necessary to determine the root causes of any QA/QC program breakdowns which contributed to any deficiencies identified in Tasks A and B. As discussed at the May 12 meeting, Task C should be conducted in parallel with Tasks A and B. Further, a separate independent consulting firm should be retained for Task C. Finally, LILCO's operational QA/QC program, implementing procedures, and staffing should also be compared to the QA/QC measures prescribed in the applicable Regulatory Guides cited in the FSAR.

VII.B: Need for Independence and an Acceptable Protocol

The credibility of the reports prepared by the consultant depend substantially on the consultant's independence from

LILCO's control. If there is to be public confidence in those reports, there must be no compromise -- or even appearance of compromise -- of the consultant's independence from LILCO. All interested persons must be assured that the consultants have no stake, real or apparent, in the outcome of the audit.

Furthermore, the parties in the ongoing license proceeding (LILCO, NRC, Suffolk County, and SOC), as well as the Board, must be provided a meaningful role in the design review and physical inspection and in the QA/QC audit. This will not only enhance the credibility of the review and inspection and the QA audit, but will also provide a basis for the Board's reliance on the reports prepared by the independent consultants.

A guiding principle in developing an acceptable protocol for the audit is that LILCO and the other parties should have equivalent relationships with the independent auditors. Thus, both LILCO and the other parties should have the opportunity to review and comment on the work scope, acceptance criteria, schedule, and resource allocation developed by the independent contractors. Further, during the conduct of the independent inspection, all parties should have the right to attend all meetings, observe all audits in progress, and review all documents presented to the auditors. To facilitate the fulfillment of these responsibilities, the independent consultants should provide weekly "look-ahead" reports to the Board and parties which set forth the work plan for the coming week in sufficient detail to enable one to select appropriate witness points. This kind of program will provide objective evidence to all parties and the Board on which to assess LILCO's compliance with regulatory requirements. The results of such review should then be utilized to determine design and plant modifications which are required by NRC regulations and licensing commitments.

VIII. CONCLUSION

Based on the foregoing, my conclusions are as follows:

a) The NRC established comprehensive QA/QC regulations in 10 CFR Part 50, Appendix B in 1970. These regulations, through 18 detailed criteria, and as supplemented by ANSI Standards and Regulatory Guides, specify mandatory actions which a licensee must take during design, construction, and operation to ensure that the facility, as designed and built, complies with the licensee's application and with substantive regulatory requirements. The overall requirements for quality standards and records were further emphasized in 1971 by the NRC in General Design Criterion 1 of Appendix A to 10 CFR Part 50.

b) The Part 50, Appendix B, regulations were effective immediately upon issuance in 1970. Allowing a reasonable length of time for a licensee to develop and implement a program in compliance with these regulations, LILCO and its major subcontractors should have had a QA/QC program which fully satisfied Appendix B by no later than April, 1973, when the Shoreham construction permit was issued.

c) Commencing in 1974 and continuing to the present time, there have been major QA/QC breakdowns at Shoreham. Such breakdowns are not isolated or minor but, as indicated by the NRC Staff's February 1982 CAT inspection, include violations of at least eleven of the eighteen Appendix B criteria. The cited QA/QC breakdowns involve a large number of systems, components, and equipment important to the safe operation of Shoreham, leading to the conclusion that there has been a serious and widespread pattern of breakdowns in the QA/QC program implementation for design and construction.

d) Based on my experience in QA/QC matters, it is my opinion that, given the succession of QA/QC deficiencies already disclosed, further investigation of other safety systems will almost certainly reveal more errors and the violation of other QA criteria. Indeed, this is also suggested by the fact that repeated QA/QC program breakdowns have been identified in all areas subject to the NRC's narrow CAT reinspection program. The evidence of LILCO's pattern of QA/QC breakdowns demonstrates that LILCO has failed to satisfy the QA/QC requirements of 10 C.F.R. Part 50, Appendix B, and Appendix A, GDC 1 which constitute the linchpin of the NRC's defense-in-depth approach to nuclear plant safety.

e) The Shoreham QA/QC deficiencies are significant also because they went undetected by LILCO and NRS Staff inspectors for years. The NRC's regulations, particularly the eighteen QA requirements of Appendix B to 10 C.F.R.

Part 50, are specifically designed to detect such defects on a timely basis and thus to ensure that nuclear plants are designed and constructed under disciplined procedures that systematically assure compliance with regulatory requirements. The QA/QC breakdowns of the pattern discovered at Shoreham should have been detected by LILCO, and on a timely basis, if the Appendix B requirements had been properly implemented. The fact that many QA/QC breakdowns were not discovered until recently -- and only then by a brief NRC Staff inspection rather than by LILCO QA personnel -- is further evidence that the LILCO QA/QC program has not been established and thus not implemented in accordance with regulatory requirements. In my technical judgment, where such a pattern of QA/QC breakdowns occurs and the breakdowns are not discovered by the licensee, the QA/QC program is deficient and the plant cannot be shown to be in compliance with regulatory requirements.

f) The deficiencies in the QA/QC program at Shoreham cannot be minimized by reference to Staff's QA/QC inspection program. While that program has discovered QA/QC breakdowns at Shoreham, the NRC program is far too limited to provide any assurance that all QA/QC deficiencies have in fact been found. Indeed, the opposite conclusion must be reached: the large number of QA/QC deficiencies found by the Staff's limited I&E program is strong additional evidence that LILCO has not satisfied 10 C.F.R. Part 50, Appendix B and Appendix A, GDC 1.

g) The NRC's I&E program should provide verification that LILCO has acted systematically to assure that Shoreham structures, systems, and components important to safety are designed, manufactured, installed, and operated in accordance with NRC requirements and LILCO's own commitments. The I&E program has not fulfilled this intended function, and the result at Shoreham is that no judgment can be made that Shoreham complies with NRC regulations. First, the I&E program is limited in resources and scope, with I&E auditing only a small portion of the LILCO QA/QC program. Nevertheless, despite these limitations, I&E has found widespread QA/QC breakdowns at Shoreham. These preclude any finding that LILCO has established and implemented an adequate QA/QC program. Surely, if LILCO had such a program, the limited NRC audits would not find a pattern of QA/QC breakdowns such as has occurred at Shoreham.

Second, the I&E program focuses <u>only</u> on those systems, structures and components classified as safety-related. However, the NRC's QA/QC requirements clearly apply to items which are important to safety as well. Thus, there is an entire area of QA review which the Staff has never even addressed. This deficiency, as well as evidence that the

LILCO operational QA program also only focuses on safety-related items, leads me to conclude that (1) there has been a failure to satisfy Appendix B requirements and (2) the evidence does not support a finding that the facility has been designed and constructed in accordance with LILCO's FSAR commitments.

h) A linchpin in the NRC's defense-in-depth approach to nuclear safety is its emphasis upon QA and QC. The evidence of QA/QC breakdowns and the limited I&E program regarding QA/QC have established substantial uncertainty in the actual quality level achieved in design, construction, and installation of structures, systems, and components at Shoreham. In my opinion, this uncertainty prevents a finding that LILCO has met regulatory requirements and FSAR commitments.

i) The serious pattern of breakdowns at Shoreham and the accompanying doubt concerning Shoreham's compliance with applicable criteria results in the facility being unlicenseable at the present time. If LILCO, nevertheless, seeks to overcome this lack of demonstration of requisite quality assurance, in my opinion it can do this only by performing a design verification and physical inspection of all Shoreham structures, systems, components, and other features important to safety by an independent consulting firm. The results of such a complete design review and physical inspection, as well any necessary changes to plant design and construction, should be subject to the scrutiny of the Board and the meaningful involvement of all parties in the ongoing Shoreham licensing proceeding. Only by such steps is it possible to ascertain the actual quality achieved at Shoreham and thus attempt to compensate for LILCO's failure to comply with Part 50, Appendix B and Appendix A, GDC 1.

j) Finally, LILCO has failed to demonstrate that its QA/QC program for operation of Shoreham will be implemented in compliance with NRC requirements. LILCO's operating QA program is only summarily described in the FSAR and, from review of the LILCO QA manuals, its scope is far too limited, particularly since it excludes items which are important to safe operation of Shoreham. Thus, I conclude that the operating QA program does not satisfy Appendix B and GDC 1 of Appendix A. This deficiency is all the more significant because, unlike during construction where LILCO essentially provided an audit function, during operation LILCO personnel will be directly responsible for conducting a majority of the QA/QC program activities. LILCO QA/QC personnel have little direct experience in implementing a nuclear grade quality program at an operating nuclear station. Thus, the lack of necessary detail or scope of the operating QA program precludes a finding that LILCO will meet regulatory requirements.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
LONG ISLAND LIGHTING COMPANY	)	Docket No. 50-322 (O.L.)
(Shoreham Nuclear Power Station, Unit 1)	)	

ATTACHMENTS TO
PREPARED DIRECT TESTIMONY OF
RICHARD B. HUBBARD
ON BEHALF OF SUFFOLK COUNTY

REGARDING

SUFFOLK COUNTY CONTENTIONS 12, 13, 14 AND 15

QUALITY ASSURANCE/QUALITY CONTROL

June 29, 1982

LIST OF ATTACHMENTS

ATTACHMENT	DESCRIPTION
1	Supplement: Experience and Qualifications of Richard B. Hubbard
2	Appendix I to SC Contention 12
3	Testimony of NRC Chairman Palladino and NRC Operations Director Dircks before the U.S. House Subcommittee on Energy and the Environment, November 19, 1981
4	IE Report 50-322/82-04, As-Built Inspection of the Residual Heat Removal System, May 12, 1982
5	Summary of Shoreham "Violations" and "Deviations" Cited by NRC IE Program (1974 to February, 1982)
6	Staff Report on Quality Assurance by the President's Commission on the Accident at Three Mile Island, October, 1979
7	Report for the Combined Utility Assessment of the Adequacy of the LILCO QA Program for Nuclear Application, February 27, 1981
8	LILCO's News Release and Viewgraphs Describing Its Shoreham Physical Inspection Program
9	LILCO's Independent Design Review of the Core Spray System
10	Guidelines for Independent Inspection, Task C, QA/QC Programs

ATTACHMENT 1

SUPPLEMENT: EXPERIENCE AND QUALIFICATIONS OF RICHARD B. HUBBARD

ATTACHMENT A

EXPERIENCE AND QUALIFICATIONS OF RICHARD B. HUBBARD

Richard B. Hubbard is a Professional Quality Engineer licensed by the State of California (license number QU 805), a technical consultant, and a founder (in 1976) and vice president of MHB Technical Associates, a corporation engaged in the business of technical consulting on energy and environmental issues, having its principal office at 1723 Hamilton Avenue, San Jose, California, 95125. Hubbard holds a B.S. in Electrical Engineering from the University of Arizona (1960) and an M.B.A. from the University of Santa Clara (1969). He has seventeen years experience in the design, manufacture, construction, and operation of nuclear power generation facilities including eleven years experience in responsible managerial positions in the Nuclear Instrumentation Department (1965-1971), Atomic Power Equipment Department (1971-1975), and Nuclear Energy Control and Instrumentation Department (1975-1976) of the General Electric Company (GE). For the past six years, Hubbard, along with his co-founders of MHB Technical Associates, have conducted numerous studies pertaining to the safety, quality, reliability, and economic aspects of nuclear power facilities.

From November, 1971 to February, 1976, Hubbard was Manager of Quality Assurance for the manufacturing operations at the San Jose, California, headquarters of GE's Nuclear Energy Division. He was responsible for the development and implementation of quality plans, programs, methods, and equipment to assure that equipment for nuclear plants manufactured and procured by General Electric met quality requirements as defined in NRC regulation 10 C.F.R. 50, Appendix B; ASME Boiler and Pressure Vessel Code; customer contracts; and GE Corporate policies and procedures. While employed by General Electric, Hubbard was responsible for developing a quality system which received NRC certification in 1975. The system was also successfully surveyed for ASME "N" and "NPT" symbol authorizations in 1975. He was also responsible for the quality assurance program and its implementation at GE's spare and renewal parts warehouse.

Hubbard is a member of the IEEE Nuclear Power Engineering standards subcommittee responsible for the preparation and revision of Quality Assurance standards for safety-related aspects of nuclear power facilities. He is currently a member of the IEEE Committee which is preparing a standard which addresses the requirements, including the quality assurance program, for the selection and utilization of replacement parts for Class IE equipment during the construction and operation phase.

Hubbard has testified on safety-related aspects of nuclear power facilities' quality assurance programs as an expert witness before the Nuclear Regulatory Commission Atomic Safety and Licensing Boards; before (and at the request of) the NRC's Advisory Committee on Reactor Safeguards; before the Joint Committee on Atomic Energy of the United States Congress; and before various federal and state legislative and administrative bodies.

ATTACHMENT 2

LETTER, LANPHER TO BRENNER DATED MARCH 15, 1982

RESTATING S.C. CONTENTIONS 12 AND 13

APPENDIX I

This Appendix constitutes a particularization of Suffolk County Contention 12 (Revised), to which SOC is also a party. Suffolk County contends that LILCO has failed to comply with 10 C.F.R. Part 50, Appendix B, as illustrated by the following:

(a) Quality Assurance Program [particularization of former S.C.12(d)]

Contrary to Criterion 2, activities affecting quality were not accomplished under suitably controlled conditions in that:

(i) As of October 31, 1979 periodic inspections by personnel qualified in accordance with ANSI N45.2.6 were not performed to ensure the control of items in storage as required by ANSI N45.2.2. (I&E Report 79-16)

(ii) No mechanism exists to update the Equipment Storage History Cards at the time when equipment changes location either in the warehouse or from the warehouse to a permanent inplant location. (I&E Report 79-16)

(iii) Periodic cleanness checks are not specified for many of the components stored in the plant (e.g., Standby Liquid Control Pumps and Motors, Core Spray Motors, and Residual Heat Removal Pumps and Motors). Additionally, as noted on inspections conducted between October 3 and October 11, 1979, many components were not maintained with adequate cleanness. (I&E Report 79-16)

(iv) Caps, covers or plugs were noted to have been removed and not immediately replaced on several Category I components during inspections conducted between October 3 and October 26, 1979. (I&E Report 79-16)

(v) The space heaters in panels 1H21*PNL 10 and 1H21*PNL 26 were found to be de-energized on October 16, 1979. (I&E Report 79-16)

(vi) On April 5, 1978, the quality assurance manual governing Reactor Controls Incorporated installation activities for the control rod drive system did not identify those responsible for inspection of the system for conformance to drawings (except welding), nor define acceptance criteria for such inspection, nor provide for verification of completion and evaluation of such inspection, nor provide for documentation of the results. One result of the absence of such provisions was the presence of reverse slope and low points on 1 1/2 inch exhaust water headers and charging water headers contrary to the slope specified on drawing FP-12C-5A, without identification and documentation by site staff for RCI engineering resolution and consideration management. (I&E Report 78-05)

(b) <u>Design Control [particularization of S.C.12(a),(b),(n) and (p)]</u>

Contrary to Criterion 3, design control measures were inadequate in that:

(i) On December 14, 1981 there was no Yellow-Lined Master in the Startup Resource Center Yellow-Lined Master File for drawings ESK-11R4204 or ESK-6T2301. (I&E Report 81-22)

(ii) As of December 14, 1981, although the affected Preoperational Tests had been completed, there was no stamping or other documentation on the following drawings in the Startup Resource Center Yellow-Lined Master File to indicate that the latest revisions had been reviewed by the Test Engineer: ESK-5R2303, ESK-5R2304, ESK-6P2108, and ESK-6P2111. (I&E Report 81-22)

(iii) As of December 14, 1981 numerous superseded drawings were not retained in the Startup Resources Center Yellow-Lined Master File, including:

--ESK-11R4204, original and Rev. 1
--ESK-6T2301, Rev. 2
--ESK-6G1133, Rev. 5
--ESK-6P2111, Rev. 3

(I&E Report 81-22)

(iv) As of December 14, 1981, numerous superseded drawings retained in the Startup Resource Center Yellow-Lined Master File were not marked "VOID", including:

--ESK-11R4201, Rev. 3
--ESK-11R4202, Rev. 2
--ESK-5R301, Rev. 11, 11A, 12, and 13
--ESK-6R4308, Rev. 3
--ESK-5R2304, Rev. 6, 6A, 6B, and 6C
--ESK-6G1104, Rev. 3
--ESK-6G1114, Rev. 2

(I&E Report 81-22)

v. The requirement that containment isolation valves be located as close to containment as practical was not prescribed by documented instructions, procedures, or drawings for the small bore piping containment isolation valves. As a result, the following outside containment isolation valves were not located as close to containment as practical:

--Valves 1C11*01V-1028 A and B were installed 10 to 15 feet from the containment penetration; and

--Valves 1P50*MOV-103 A and B were installed approximately 40 feet from the containment penetration.

(I&E Report 81-02)

(vi) On June 18, 1980, the inspector observed that the process monitor sample point at elevation 75'-0", azimuth 210° and sampling lines extending from elevation 78'-7" to elevation 96'-0" did not incorporate principles for airborne sampling identified in ANSI N13.1-1969. (I&E Report 80-10)

(vii) Vent lines and vent valves not specified on the S&W drawing (FM-25A) were installed in three locations of the High Pressure Coolant Injection System without an authorizing E&DCR to modify the drawing. (I&E Report 80-14)

(viii) On June 19, 1980 the inspector observed that the redundant safety-related conduit installation for system 1G33 did not meet the separation criteria of specification SH1-159. (I&E Report 80-10)

(ix) On August 13, 1979, battery room ventilation control panels PNL-VC16, VC17 and VC18 were installed without approved engineering drawings. (I&E Report 79-12)

(x) As of May 25, 1979 S&W specification SH-1-159 and associated change EDCR-F19039 permit installation of raceways which do not conform to the minimum separation criteria, and permit subsequent installation of cables in the nonconforming raceways. (I&E Report 79-07)

(xi) On April 10, 1978, the quality assurance manual governing Reactor Controls incorporated installation activities for the control rod drive system did not identify those responsible for inspection of the system for conformance to drawings (except welding), nor define acceptance criteria for such inspection, nor provide for verification of completion and evaluation of such inspection, nor provide for documentation of the results. One result of the absence of such provisions was the presence of reverse slope and low points on 1 1/2 inch exhaust water headers and charging water headers contrary to the slope specified on drawing FP-12C-5A, without identification and documentation by site staff for RCI engineering resolution and consideration by management. (I&E Report 78-05)

(xii) The LILCO Engineering Field Extension Office reviewed, approved, and issued a repair welding procedure RP-38, titled "Repair of Defects in Weld of Stainless Steel Forging to Carbon Steel Pipe in CRD Penetrations" which was not qualified, as issued, to the applicable code and specification. (I&E Report 77-01)

(xiii) Obsolete drawings were not removed from the work area and destroyed as required by QA/QC procedures. (I&E Report 76-02)

(xiv) Not all of the safety related Engineering and Design Change Reports which were originated during March at the site were reviewed by LILCO as required by Paragraph W3.2.2 of Procedure P-305. The following are examples of E&DCR's which were not reviewed by the quality assurance organization:

--F-2847
--F-2848
--F-2855
--F-2862

(I&E Report 76-06)

(xv) An audit performed by LILCO and documented in Audit Report FA-322 dated November 6, 1975, identified that the constructor was not filing Engineering and Design Change Reports with the applicable specifications and procedures as required by site procedures. A followup audit was performed by the licensee and documented in Audit Report FA-399, dated April 27, 1976, where it was again identified that the same conditions existed. (I&E Report 76-06)

(c) <u>Instructions, Procedures, and Drawings</u>

Contrary to Criterion 5, the measures for the documentation of instructions, procedures and drawings were inadequate in that:

(i) On December 14, 1981 there was no Yellow-Lined Master in the Startup Resource Center Yellow-Lined Master File for drawings ESK-11R4204 or ESK-6T2301. (I&E Report 81-22)

(ii) As of December 14, 1981, although the affected Preoperational Tests had been completed, there was no stamping or other documentation on the following drawings in the Startup Resource Center Yellow-Lined Master File to indicate that the latest revisions had been reviewed by the Test Engineer: ESK-5R2303, ESK-5R2304, ESK-6P2108, and ESK-6P2111.

(iii) As of December 14, 1981 numerous superseded drawings were not retained in the Startup Resources Center Yellow-Lined Master File, including:

--ESK-11R4204, original and Rev. 1
--ESK-6T2301, Rev. 2
--ESK-6G1133, Rev. 5
--ESK-6P2111, Rev. 3

(I&E Report 81-22)

(iv) As of December 14, 1981, numerous superseded drawings retained in the Startup Resource Center Yellow-Lined Master File were not marked "VOID", including:

--ESK-11R4201, Rev. 3
--ESK-11R4202, Rev. 2
--ESK-5R301, Rev. 11, 11A, 12, and 13
--ESK-6R4308, Rev. 3
--ESK-5R2304, Rev. 6, 6A, 6B, and 6C
--ESK-6G1104, Rev. 3
--ESK-6G1114, Rev. 2

(I&E Report 81-22)

(v) PT.315.001B and C, "125V DC Power Distribution Preop. Test" for the B and C Systems, were being performed in January 1981, while the DC Bus Current and Voltage meters and the Battery Charger DC output current and voltage meters, which are required for conducting the PT, had not been recalibrated within one year. (I&E Report 81-01)

(vi) On August 5 & 6, 1981, the inspector identified the following leads in Panel 601 lifted with no documentation in the Log or Tags hung:

-- Leads to pressure indicator E51-P1 001;

-- Two leads from cable E11 BBC 640;

-- Two leads from cable E11 BBC 641; and

-- Lead CC 75 to Terminal Board HH.

(I&E Report 81-14)

(vii) On August 5 & 6, 1981, the authorization block was not signed on tags #1528, 1529 and 1838. (I&E Report 81-14)

(viii) On August 5 & 6, 1981, two fuses were found installed in the Remote Shutdown Panel even though RED tags

#30224 and 30225 specified that the fuses be pulled. (I&E Report 81-14)

(ix) On August 5 & 6, 1981, BLUE startup jurisdictional tags were hung concurrently with YELLOW construction jurisdictional tags on a number of components of the Reactor Building Closed Loop Cooling Water System (P42). (I&E Report 81-14)

(x) On August 5 & 6, 1981, none of the four active jumper/lifted lead permits had the expected duration recorded. (I&E Report 81-14)

(xi) On August 5 & 6, 1981, the jumpers for Permit #81-611 had been removed but the permit and jumper log had not been updated. (I&E Report 81-14)

(xii) On August 5 & 6, 1981, the Main Control Room set of controlled station procedures was not maintained current in that contained Rev. 1 vice Rev. 2 of SP 12.035.01, "Control of Lifted Leads and Jumpers". (I&E Report 81-14)

(xiii) On September 11, 1981, the inspector observed that Project Procedure P304 does not require report of a possible reportable deficiency. (I&E Report 81-16)

(xiv) The requirement that containment isolation valves be located as close to containment as practical was not prescribed by documented instructions, procedures, or drawings for the small bore piping containment isolation valves. As a result, the following outside containment isolation valves were not located as close to containment as practical:

-- Valves 1C11*01V-1028 A and B were installed 10 to 15 feet from the containment penetration; and,

-- Valves 1P50*MOV-103 A and B were installed approximately 40 feet from the containment penetration.

(I&E Report 81-02)

(xv) On June 19, 1980 the inspector observed that the redundant safety-related conduit installation for system 1G33 did not meet the separation criteria of specification SH1-159. (I&E Report 80-10)

(xvi) On December 7, 1979, snubbers No. 1G33*PSSP-228, 229, 230 and 231 had been released to construction for installation and were subsequently installed without evidence of having been stroked to assure that there was no binding. (I&E Report 80-15)

(xvii) As of March 30, 1979, the following Engineering Quality Assurance Procedures (EQAP's) had not been updated though their respective Change Notices were in effect in excess of a calendar year.

-- EQAP 2.3, Revision 2, Change Notice (CN) No. 1, dated April 15, 1977.

-- EQAP 2.8, Revision 0, CN No.1, dated April 15, 1977.

-- EQAP 3.3, Revision 2, CN No.1, dated March 10, 1978.

-- EQAP 4.1, Revision 3, CN No.1, dated April 18, 1977.

-- EQAP 15.2, Revision 3, CN No.1, dated July 18, 1977.

-- EQAP 16.1, Revision 3, CN No.1, dated January 3, 1977.

(I&E Report 79-05)

(xviii) Beam support welds numbered 17 and 17A were accepted by RCI quality control on November 15, 1978 although fitup gaps of 3/16 inch existed in the completed welds. Similar fitup gaps exist in three or more other similar beam supports. (I&E Report 79-07)

(xix) On May 24, 1979, installed and inspected RCIC system instrument tubing was separated by less than one foot and was not provided with physical barriers at the connection to pipe spool 1"-SLP-9-151-2-1. (I&E Report 79-07)

(xx) On August 13, 1979, battery room ventilation control panels PNL-VC16, VC17 and VC18 were installed without approved engineering drawings. (I&E Report 79-12)

(xxi) On May 19, 1979 Weld Joint No. 1G-33-WD9-3-1-FW "D" was welded using ER-308 filler metal. (I&E Report 79-24)

(xxii) On February 15, 1978, twenty unused and unreturned low-hydrogen type E7018 weld electrodes of various diameters were found in scattered locations inside the containment drywell where safety related welding work activities were in progress. This was in addition to significantly greater quantities of partly used electrodes also lying loose throughout these work areas. (I&E Report 78-02)

(xxiii) On March 9, 1978, installed and inspected hangers had: 7 jam nuts missing from three hangers, 2 loose nuts and bolts on two hangers, 3 lock wires or cotter pins missing on one hanger, 1 spring preset piece missing from one hanger, and 8 1/4" anchor bolt spacing not meeting the 6" specified for one hanger. (I&E Report 78-03)

(xxiv) On April 5, 1978, the quality assurance manual governing Reactor Controls Incorporated installation activities for the control rod drive system did not identify those responsible for inspection of the system for conformance to drawings (except welding), nor define acceptance criteria for such inspection, nor provide for verification of completion and evaluation of such inspection, nor provide for documentation of the results. One result of the absence of such provisions was the presence of reverse slope and low points on 1 1/2 inch exhaust water headers and charging water headers contrary to the slope specified on drawing FP-12C-5A, without identification and documentation by site staff for RCI engineering resolution and consideration by management. (I&E Report 78-05)

(xxv) On May 3, 1978, the pipe to elbow, nuclear class 1, field weld 1E11-IC020-FW 3 was observed in the final accepted condition with areas ground 1/16" below the pipe surface and the weld reinforcement surface was not ground parallel to the adjacent pipe surface in accordance with drawing note of Std-MP-1056-3-3. (I&E Report 78-06)

(xxvi) On August 9, 1978, the front weld (inside diameter) size on pipe spool slip-on flange 1B21-SLP-211-3-1 was less than 1/4 inch. The back weld (outside diameter) size on pipe spool slip-on flanges 1B21-SLP-211-3-1 and 1B21-SLP-203-3-1 was less than 3/8 inch. (I&E Report 78-12)

(xxvii) On September 29, 1978, S&W field forces performed welding on a Skewed Tee Joint of angle 54 degrees, on the rectangular tubular pipe support number 1E11-PSA-072 of the residual heat removal system, using non-applicable procedure W70G which was not prequalified for the weld joint configuration. (I&E Report 78-15)

(xxviii) On October 19, 1978, the energized safety related 4160 volt switchgear 1R22*SWG103 in the 103 emergency switchgear room, elevation 25', control room building, did not comply with the requirements of procedure C.S.1 13.I. There was an accumulation of dirt inside of the energized switchgear equipment which could create conditions that would adversely affect the quality of the component and the equipment operation. (I&E Report 78-16)

(xxix) On June 13, 1977 and July 7, 1977, structural steel identified on shipping tickets No. 3663 and 3787, respectively were accepted by Field QC and released for construction without receipt of Shipping Release Tags or Stone & Webster Certificates of Compliance. (I&E Report 77-12)

(xxx) On December 6, 1977, two safety related, non-manual valves, numbers T46-TCV024A and T46-TCV025A, were uncovered and exposed to the rain in the short term storage area adjacent to the reactor building. (I&E Report 77-23)

(xxxi) On December 7, 1977, LILCO's level storage A did not conform with the requirements of the Quality Control Procedure 17.1 in that an atmosphere free from dust was not maintained as evidenced by significant dust on the shelves and safety related items in storage. (I&E Report 77-23)

(xxxii) During the period of August 16, 1976, to March 4, 1977, the disposition of N&D 922, dated August 16, 1977, and titled "Control Rod Drive Penetrations", was revised from a grind out and reweld repair of the safety related control rod drive penetrations to the welding of sleeves over the dissimilar metal weld joints without a "revised" disposition being issued, reviewed, and approved as required by EAP-15.1, revision 3. (I&E Report 77-17)

(xxxiii) On February 9, 1977, Field QC had performed a final visual inspection and accepted field weld IE 41-1C179 FW06, although the transition between this weld and pump P-016 did not conform to the required profile for approximately 200° of its circumference. (I&E Report 77-17)

(xxxiv) Obsolete drawings were not removed from the work area and destroyed as required by QA/QC procedures. (I&E Report 76-02)

(xxxv) Not all of the safety related Engineering and Design Change Reports which were originated during March at the site were reviewed by the licensee as required by Paragraph W3.2.2 of Procedure P-305. The following are examples of E&DCR's which were not reviewed by the quality assurance organization:

--F-2847
--F-2848
--F-2855
--F-2862

(I&E Report 76-06)

(xxxvi) On November 9, 1976, it was observed that the refueling cavity liner surface was contaminated with iron which resulted from thermal cutting operations of carbon steel structural components adjacent to the liner. (I&E Report 76-12)

(xxxvii) Vendor documentation for pipe spool 1E11-WR232-2-01 was not reviewed in accordance with Field QC (FQC) Procedure 9.1; and pipe hangers in storage were not protected against deterioration as required by FQC 17.1. (I&E Report 76-01)

(xxxviii) It was found that Air Meter No. 8/12/9 had not been returned for calibration by the due date shown on the "Recall List" and that a Nonconformance and Disposition Report had not been issued as required by FQC 16.1. (I&E Report 75-01)

(xxxxix) QC documentation of Class I concrete preplacement inspections did not provide or reference specified quantitative acceptance criteria for construction joint preparation and the inspection sheets were incorrectly filled out with respect to certain construction joint, keyway and water-stop inspection items. (I&E Report 75-03)

(xxxx) The dew point of the nitrogen blanket within the RPV was not maintained within the limits established by the storage procedure and the records showing dew point vs. outside air temperatures were not maintained as required by the Storage Procedures. (I&E Report 75-05)

(xxxxi) LILCO audits of certain Stone & Webster site quality control activities were p·rformed at a greater interval than prescribed by LILCO's Quality Assurance Procedure (QAP) 18.2. (I&E Report 74-03)

(xxxxii) Receipt inspections of KSM studs was being performed by Stone & Webster without an approved procedure. (I&E Report 74-03)

(xxxxiii) KSM studs were accepted by Stone & Webster receipt inspection although the required certification was not to the standard specified in the Purchase Order. (I&E Report 74-03) (xxxxiv) Access to the RPV and the head was uncontrolled. (I&E Report 74-05)

(xxxxv) The nitrogen pressure, as indicated by a manometer, was less than half the amount specified. (I&E Report 74-05)

(d) <u>Document Control [particularization of S.C.12(b)]</u>

Contrary to Criterion 6, the measures to control documents were inadequate in that:

(i) On December 14, 1981 there was no Yellow-Lined Master in the Startup Resource Center Yellow-Lined Master File for drawings ESK-11R4204 or ESK-6T2301. (I&E Report 81-22)

(ii) As of December 14, 1981, although the affected Pre-operational Tests had been completed, there was no stamping or other documentation on the following drawings in the Startup Resource Center Yellow-Lined Master File to indicate that the latest revisions had been reviewed by the Test Engineer: ESK-5R2303, ESK-5R2304, ESK-6P2108, and ESK-6P2111. (I&E Report 81-22)

(iii) As of December 14, 1981 numerous superseded drawings were not retained in the Startup Resources Center Yellow-Lined Master File, including:

--ESK-11R4204, original and Rev. 1
--ESK-6T2301, Rev. 2
--ESK-6G1133, Rev. 5
--ESK-6P2111, Rev. 3

(I&E Report 81-22)

(iv) As of December 14, 1981 numerous superseded drawings retained in the Startup Resources Center Yellow-Lined Master File were not marked "VOID," including:

--ESK-11R4201, Rev. 3
--ESK-11R4202, Rev. 2
--ESK-5R301, Rev. 11, 11A, 12, and 13
--ESK-6R4308, Rev. 3
--ESK-5R2304, Rev. 6, 6A, 6B, and 6C
--ESK-6G1104, Rev. 3
--ESK-6G1114, Rev. 2

(I&E Report 81-22)

(v) On August 5 & 6, 1981, the Main Control Room set of controlled station procedures was not maintained current in that it contained Rev. 1 vice Rev. 2 of SP 12.035.01, "Control of Lifted Leads and Jumpers". (I&E Report 81-14)

(vi) On August 13, 1981, Startup Manual No. 43, located in the control room and used by persons in the control room, was not adequately controlled in that it was not updated to include:

-- Manual Revision No. 12, dated February 18, 1981;

-- Startup Instruction No. 8, Revision 0, dated February 3, 1981;

-- Startup Instruction No. 1, Revision 5, dated May 27, 1981;

-- Startup Instruction No. 6, Revision 1, dated March 3, 1981;

-- Startup Instruction No. 7, Revision 1, dated June 29, 1981.

In addition, the "Controlled" Manuals Distribution List posted in revision 12, dated February 18, 1981 incorrectly assigned Startup Manual Copy No. 43 to a different recipient. (I&E Report 81-13).

(vii) As of March 30, 1979, the following Engineering Quality Assurance Procedures (EQAP's) had not been updated though their respective Change Notices were in effect in excess of a calendar year.

-- EQAP 2.3, Revision 2, Change Notice (CN) No. 1 dated April 15, 1977.

-- EQAP 2.8, Revision 0, CN No.1, dated April 15, 1977.

-- EQAP 3.3, Revision 2, CN No.1, dated March 10, 1978.

-- EQAP 4.1, Revision 3, CN No.1, dated April 18, 1977.

-- EQAP 15.2, Revision 3, CN No.1, dated July 18, 1977.

-- EQAP 16.1, Revision 3, CN No.1, dated January 3, 1977.

(I&E Report 79-05)

(viii) On April 5, 1978, the quality assurance manual governing Reactor Controls Incorporated installation activities for the control rod drive system did not identify those responsible for inspection of the system for conformance to drawings (except welding), nor define acceptance criteria for such inspection, nor provide for verification of completion and evaluation of such inspection, nor provide for documentation of the results. One result of the absence of such provisions was the presence of reverse slope and low points on 1 1/2 inch exhaust water headers and charging water headers contrary to the slope specified on drawing FP-12C-5A, without identification and documentation by site staff for RCI engineering resolution and consideration by management. (I&E Report 78-05)

(ix) Obsolete drawings were not removed from the work area and destroyed as required by QA/QC procedures. (I&E Report 76-02)

(x) Not all of the safety related Engineering and Design Change Reports which were originated during March at the site were reviewed by LILCO as required by Paragraph W3.2.2 of Procedure P-305. The following are examples of E&DCR's which were not reviewed by the quality assurance organization:

--F-2847
--F-2848
--F-2855
--F-2862

(I&E Report 76-06)

(xi) An audit performed by LILCO and documented in Audit Report FA-322 dated November 6, 1975, identified that the constructor was not filing Engineering and Design Change Reports with the applicable specifications and procedures as required by site procedures. A followup audit was performed by LILCO and documented in Audit Report FA-399, dated April 27, 1976, where it was again identified that the same conditions existed. (I&E Report 76-06)

(xii) It was observed that although the Architect-Engineer's specification SH1-75 had been revised to include Addendum 3, and that this revised specification had been issued to the containment construction contractor and that this contractor was performing his work in accordance with this Addendum, the contractors QA manual which ostensibly controlled his quality related activities still identified specification SH1-75, Addendum 2 as the appropriate specification. (I&E Report 74-08)

(e) <u>Control of Purchased Material, Equipment, and Services [particularization of S.C.12(k) and (o)]</u>

Contrary to Criterion 7, the measures to assure that purchased material, equipment, and services conform to the procurement documents are inadequate in that:

(i) On December 7, 1979, snubbers No. 1G33*PSSP-228, 229, 230 and 231 had been released to construction for installation and were subsequently installed without evidence of having been stroked to assure that there was no binding. (I&E Report 80-15)

(ii) On June 13, 1977 and July 7, 1977, structural steel identified on shipping tickets No. 3663 and 3787, respectively were accepted by Field QC and released for construction without receipt of Shipping Release Tags or Stone & Webster Certificates of Compliance. (I&E Report 77-12)

(iii) Vendor documentation for pipe spool 1E11-WR232-2-01 was not reviewed in accordance with Field QC (FQC) Procedure 9.1; and pipe hangers in storage were not protected against deterioration as required by FQC 17.1. (I&E Report 76-01)

(iv) Receipt inspections of KSM studs was being performed by Stone & Webster without an approved procedure. (I&E Report 74-03)

(v) KSM studs were accepted by Stone & Webster receipt inspection although the required certification was not to the standard specified in the Purchase Order. (I&E Report 74-03)

(f) Identification and Control of Materials, Parts, and Components [particularization of S.C.12(k)]

Contrary to Criterion 8, the identification and control of materials, parts, and components were inadequate in that:

On June 13, 1977 and July 7, 1977, structural steel identified on shipping tickets No. 3663 and 3787, respectively were accepted by Field QC and released for construction without receipt of Shipping Release Tags or Stone & Webster Certificates of Compliance. (I&E Report 77-12)

(g) Control of Special Processes [particularization of S.C.12(e),(f) and (j)]

Contrary to Criterion 9, the measures to assure that special processes are properly controlled are inadequate in that:

(i) On or about November 11, 1978, weld joint 1B21-IC175-FW6 was heated, for post weld heat treatment, at a rate exceeding the ASME III Code allowable. Specifically, the joint was heated between 640°F and 930°F at a rate of 290°F/hr. while the maximum allowable rate was 225°F/hr. (I&E Report 79-02)

(ii) In October 1976, the Courter Company crafts, under direction of Stone and Webster, performed thermal cutting of attachment welds to remove pressure caps from nozzles N3 and N4 of residual heat removal heat exchangers No. 034A and No. 034B, without qualified and approved procedures and apparently without performing preheat required by the applicable specifications. (I&E Report 79-04]

(iii) On May 19, 1979 Weld Joint No. 1G-33-WD9-3-1-FW "D" was welded using ER-308 filler metal. (I&E Report 79-24)

(iv) On February 15, 1978, twenty unused and unreturned low-hydrogen type E7018 weld electrodes of various diameters were found in scattered locations inside the containment drywell where safety related welding work activities were in progress. This was in addition to significantly greater quantities of partly used electrodes also lying loose throughout these work areas. (I&E Report 78-02)

(v) On March 8, 1978, the temperature, as measured using a 200°F temperature indicating crayon, of the RHR pipe field weld E11-IC017-FW-03, schedule 80, 1.031" wall thickness, was less than 200°F during the in process welding operations. (I&E Report 78-03)

(vi) On May 3, 1978, the pipe to elbow, nuclear class 1, field weld 1E11-IC020-FW 3 was observed in the final accepted condition with areas ground 1/16" below the pipe surface and the weld reinforcement surface was not ground parallel to the adjacent pipe surface in accordance with drawing note of Std-MP-1056-3-3. (I&E Report 78-06)

(vii) On August 9, 1978, the single-bevel-groove weld joint angles for pipe break restraints FWR 1, 2, and 15 were 30°. (I&E Report 78-12)

(viii) On August 9, 1978, the front weld (inside diameter) size on pipe spool slip-on flange 1B21-SLP-211-3-1 was less than 1/4 inch. The back weld (outside diameter) size on pipe spool slip-on flanges 1B21-SLP-211-3-1 and 1B21-SLP-203-3-1 was less than 3/8 inch. (I&E Report 78-12)

(ix) On September 29, 1978, S&W field forces performed welding on a Skewed Tee Joint of angle 54 degrees, on the rectangular tubular pipe support number 1E11-PSA-072 of the residual heat removal system, using non-applicable procedure W70G which was not prequalified for the weld joint configuration. (I&E Report 78-15)

(x) On October 25, 1978, general areas of undercut in excess of 1/32" deep were observed on the trucks, bridge, and trolley welds of the reactor building polar crane. (I&E Report 78-16)

(xi) The Engineering Field Extension Office reviewed, approved, and issued a repair welding procedure RP-38, titled "Repair of Defects in Weld of Stainless Steel Forging to Carbon Steel Pipe in CRD Penetrations" which was not qualified, as issued, to the applicable code and specification. (I&E Report 77-01)

(xii) During the period of August 16, 1976, to March 4, 1977, the disposition of N&D 922, dated August 16, 1977, and titled "Control Rod Drive Penetrations", was revised from a grind out and reweld repair of the safety related control rod drive penetrations to the welding of sleeves over the dissimilar metal weld joints without a "revised" disposition being issued, reviewed, and approved as required by EAP-15.1, revision 3. (I&E Report 77-17)

(h) <u>Inspection [particularization of S.C.12(h)]</u>

Contrary to Criterion 10, the inspection activities were inadequate in that:

(i) As of October 31, 1979 periodic inspections by personnel qualified in accordance with ANSI N45.2.6 were not performed to ensure the control of items in storage as required by ANSI N45.2.2. (I&E Report 79-16)

(ii) Caps, covers or plugs were noted to have been removed and not immediately replaced on several Category I components during inspections conducted between October 3 and October 26, 1979. (I&E Report 79-16)

(iii) Beam support welds numbered 17 and 17A were accepted by RCI quality control on November 15, 1978 although fitup gaps of 3/16 inch existed in the completed welds. Similar fitup gaps exist in three or more other similar beam supports. (I&E Report 79-07)

(iv) On May 24, 1979, installed and inspected RCIC system instrument tubing was separated by less than one foot and was not provided with physical barriers at the connection to pipe spool 1"-SLP-9-151-2-1. (I&E Report 79-07)

(v) On August 13, 1979, battery room ventilation control panels PNL-VC16, VC17 and VC 18 were installed without approved engineering drawings. (I&E Report 79-12)

(vi) On March 9, 1978, installed and inspected hangers had: 7 jam nuts missing from three hangers, 2 loose nuts and bolts on two hangers, 3 lock wires or cotter pins missing on one hanger, 1 spring preset piece missing from one hanger, and 8 1/4" anchor bolt spacing not meeting the 6" specified for one hanger. (I&E Report 78-03)

(vii) On May 3, 1978, the pipe to elbow, nuclear class 1, field weld 1E11-IC020-FW 3 was observed in the final accepted condition with areas ground 1/16" below the pipe surface and the weld reinforcement surface was not ground parallel to the adjacent pipe surface in accordance with drawing note of Std-MP-1056-3-3. (I&E Report 78-06)

(viii) On April 5, 1978, the quality assurance manual governing Reactor Controls Incorporated installation activities for the control rod drive system did not identify those responsible for inspection of the system for conformance to drawings (except welding), nor define acceptance criteria for such inspection, nor provide for verification of completion and evaluation of such inspection, nor provide for documentation of the results. One result of the absence of such provisions was the presence of reverse slope and low points on 1 1/2 inch exhaust water headers and charging water headers contrary to the slope specified on drawing FP-12C-5A, without identification and documentation by site staff for RCI engineering resolution and consideration by management. (I&E Report 78-05)

(ix) On October 19, 1978, the energized safety related 4160 volt switchgear 1R22*SWG103 in the 103 emergency switchgear room, elevation 25', control room building, did not comply with the requirements of procedure C.S.1 13.I. There was an accumulation of dirt inside of the energized switchgear equipment which could create conditions that would adversely affect the quality of the component and the equipment operation. (I&E Report 78-16)

(x) On February 9, 1977, Field QC had performed a final visual inspection and accepted field weld IE 41-1C179 FW06, although the transition between this weld and pump P-016 did not conform to the required profile for approximately 200° of its circumference. (I&E Report 77-17)

(xi) Stone & Webster (S&W) Deficiency Correction Order Nos. 10182E, 10185E and 10187E were issued in November, 1976, to correct identified instances where field routed safety related cable installation within switchgear enclosures did not meet the separation criteria of S&W Specification No. SH1-159. The corrective actions specified and taken by S&W personnel did not include corrective actions to preclude repetition of the nonconformances. As a result, on March 2, 1977, the NRC inspector identified safety related cable installations in switchgear enclosure 1R22*SWG-101-1 and 101-2, 1R22*SWG-103-2 and 1H11*MCB-01-29 and 01-31, which did not meet the specified separations criteria and these nonconformances were repetitious of those previously identified by S&W personnel in November, 1976. (I&E Report 77-05)

(xii) QC documentation of Class I concrete preplacement inspections did not provide or reference specified quantitative acceptance criteria for construction joint preparation and the inspection sheets were incorrectly filled out with respect to certain construction joint, keyway and water-stop inspection items. (I&E Report 75-03)

(xiii) The dew point of the nitrogen blanket within the RPV was not maintained within the limits established by the storage procedure and the records showing dew point vs. outside air temperatures were not maintained as required by the Storage Procedures. (I&E Report 75-05)

(xiv) Receipt inspections of KSM studs was being performed by Stone & Webster without an approved procedure. (I&E Report 74-03)

(xv) KSM studs were accepted by Stone & Webster receipt inspection although the required certification was not to the standard specified in the Purchase Order. (I&E Report 74-03)

(i) Test Control [particularization of S.C.12(h)]

Contrary to Criterion 11, the test control measures are inadequate in that:

(i) On August 5 & 6, 1981, the inspector identified the following leads in Panel 601 lifted with no documentation in the Log or Tags hung:

-- Leads to pressure indicator E51-P1 001;

-- Two leads from cable E11 BBC 640;

-- Two leads from cable E11 BBC 641; and

-- Lead CC 75 to Terminal Board HH.

(I&E Report 81-14)

(ii) During the inspection of the placement of safety related concrete for the radwaste building on April 27 and 28, 1976, the constructor's field quality control inspector observed that the batch plant moisture detecting equipment was not operable and he so noted this condition on the checksheet of Surveillance Inspection Plan No. 65 used for his inspection. However, the inspector failed to initiate corrective action to assure the equipment was returned to operable status as required by Paragraph 4.4 of Quality Control Procedure 10.3 and the equipment remained in an inoperable status. (I&E Report 76-08)

(j) Control of Measuring and Test Equipment [particularization of S.C.12(i)]

Contrary to Criterion 12, the measures to control measuring and test equipment are inadequate in that:

(i) PT.315.001B and C, "125V DC Power Distribution Preop. Test" for the B and C Systems, were being performed in January 1981, while the DC Bus Current and Voltage meters and the Battery Charger DC output current and voltage meters, which are required for conducting the PT, had not been recalibrated within one year. (I&E Report 81-01)

(ii) During the inspection of the placement of safety related concrete for the radwaste building on April 27 and 28, 1976, the constructor's field quality control inspector observed that the batch plant moisture detecting equipment was not operable and he so noted this condition on the checksheet of Surveillance Inspection Plan No. 65 used for his inspection. However, the inspector failed to initiate corrective action to assure the equipment was returned to operable status as required by Paragraph 4.4 of Quality Control Procedure 10.3 and the equipment remained in an inoperable status. (I&E Report 76-08)

(iii) It was found that Air Meter No. 8/12/9 had not been returned for calibration by the due date shown on the "Recall List" and that a Nonconformance and Disposition Report had not been issued as required by FQC 16.1. (I&E Report 75-01)

(k) Handling, Storage and Shipping

Contrary to Criterion 13, the measures to control handling, storage, and shipping are inadequate in that:

(i) As of October 31, 1979 periodic inspections by personnel qualified in accordance with ANSI N45.2.6 were not performed to ensure the control of items in storage as required by ANSI N45.2.2. (I&E Report 79-16)

(ii) No mechanism exists to update the Equipment Storage History Cards at the time when equipment changes location either in the warehouse or from the warehouse to a permanent inplant location. (I&E Report 79-16)

(iii) Periodic cleanness checks are not specified for many of the components stored in the plant (e.g., Standby Liquid Control Pumps and Motors, Core Spray Motors, and Residual Heat Removal Pumps and Motors). Additionally, as noted on inspections conducted between October 3 and October 11, 1979, many components were not maintained with adequate cleanness. (I&E Report 79-16)

(iv) The space heaters in panels 1H21*PNL 10 and 1H21*PNL 26 were found to be de-energized on October 16, 1979 (I&E Report 79-16)

(v) On October 19, 1978, the energized safety related 4160 volt switchgear 1R22*SWG103 in the 103 emergency switchgear room, elevation 25', control room building, did not comply with the requirements of procedure C.S.1 13.I. There was an accumulation of dirt inside of the energized switchgear equipment which could create conditions that would adversely affect the quality of the component and the equipment operation. (I&E Report 78-16)

(vi) On December 6, 1977, two safety related, non-manual valves, numbers T46-TCV024A and T46-TCV025A, were uncovered and exposed to the rain in the short term storage area adjacent to the reactor building. (I&E Report 77-23).

(vii) On December 7, 1977, LILCO's level storage A did not conform with the requirements of the Quality Control Procedure 17.1 in that an atmosphere free from dust was not maintained as evidenced by significant dust on the shelves and safety related items in storage. (I&E Report 77-23)

(viii) On November 9, 1976, it was observed that the refueling cavity liner surface was contaminated with iron which resulted from thermal cutting operations of carbon steel structural components adjacent to the liner. (I&E Report 76-12)

(ix) Vendor documentation for pipe spool 1E11-WR232-2-01 was not reviewed in accordance with Field QC (FQC) Procedure 9.1; and pipe hangers in storage were not protected against deterioration as required by FQC 17.1. (I&E Report 76-01)

(x) The 480 volt switchgear 1F28*SWG101 and its associated transformer 1R23*T101 were not maintained in a clean condition as required by Electrical Installation specification SH1-159. At the time of the inspection, the plastic cover installed for protection of the equipment was torn and was too small to completely cover the equipment resulting in deposits of dust and debris on the relay contacts and transformer windings. (I&E Report 76-07)

(xi) Storage and preservation of equipment did not meet the requirements of Construction Site Instruction 4.6, in that:

-- Protection against corrosion of electrical connectors for the Control Rod Drive Hydraulic Control Units in the Reactor Building was not provided to preclude deterioration; and,

-- Storage and storage surveillance for protection against corrosion of controls for RCIC Pump No. 1E 51-P-015 in the Reactor Building was not conducted as required. (I&E Report 76-09)

(xii) The dew point of the nitrogen blanket within the RPV was not maintained within the limits established by the storage procedure and the records showing dew point vs. outside air temperatures were not maintained as required by the Storage Procedures. (I&E Report 75-05)

(xiii) Access to the RPV and the head was uncontrolled. (I&E Report 74-05)

(xiv) The nitrogen pressure, as indicated by a manometer, was less than half the amount specified. (I&E Report 74-05)

(l) <u>Inspection, Test, and Operating Status</u>

Contrary to Criterion 14, the measures to indicate the status of inspections and tests, and operating status are inadequate in that:

(i) On August 5 & 6, 1981, the inspector identified the following leads in Panel 601 lifted with no documentation in the Log or Tags hung:

-- Leads to pressure indicator E51-P1 001;

-- Two leads from cable E11 BBC 640;

-- Two leads from cable E11 BBC 641; and

-- Lead CC 75 to Terminal Board HH.

(I&E Report 81-14)

(ii) On August 5 & 6, 1981, the authorization block was not signed on tags #1528, 1529 and 1838. (I&E Report 81-14)

(iii) On August 5 & 6, 1981, two fuses were found installed in the Remote Shutdown Panel even though RED tags #30224 and 30225 specified that the fuses be pulled. (I&E Report 81-14)

(iv) On August 5 & 6, 1981, BLUE startup jurisdictional tags were hung concurrently with YELLOW construction jurisdictional tags on a number of components of the Reactor Building Closed Loop Cooling Water System (P42). (I&E Report 81-14)

(v) On August 5 & 6, 1981, none of the four active jumper/lifted lead permits had the expected duration recorded. (I&E Report 81-14)

(vi) On August 5 & 6, 1981, the jumpers for Permit #81-611 had been removed but the permit and jumper log had not been updated. (I&E Report 81-14)

(vii) On June 19, 1980 the inspector observed that the redundant safety-related conduit installation for system 1G33 did not meet the separation criteria of specification SH1-159. (I&E Report 80-10)

(viii) No mechanism exists to update the Equipment Storage History Cards at the time when equipment changes location either in the warehouse or from the warehouse to a permanent inplant location. (I&E Report 79-16)

(m) Nonconforming Materials, Parts, or Components <u>[particularization of S.C.12(1)]</u>

Contrary to Criterion 15, measures to control materials, parts, or components which do not conform to requirements are inadequate, in that:

(i) On September 11, 1981, the inspector observed that Project Procedure P304 does not require report of a possible reportable deficiency. (I&E Report 81-16)

(ii) On June 19, 1980 the inspector observed that the redundant safety-related conduit installation for system 1G33 did not meet the separation criteria of specification SH1-159. (I&E Report 80-10)

(iii) On December 7, 1979, snubbers No. 1G33 * PSSP-228, 229, 230 and 231 had been released to construction for installation and were subsequently installed without evidence of having been stroked to assure that there was no binding. (I&E Report 80-15)

(iv) Beam support welds numbered 17 and 17A were accepted by RCI quality control on November 15, 1978 although fitup gaps of 3/16 inch existed in the completed welds.

Similar fitup gaps exist in three or more other similar beam supports. (I&E Report 79-07)

(v) On May 24, 1979, installed and inspected RCIC system instrument tubing was separated by less than one foot and was not provided with physical barriers at the connection to pipe spool 1"-SLP-9-151-2-1. (I&E Report 79-07)

(vi) As of May 25, 1979 S&W specification SH-1-159 and associated change EDCR-F19039 permit installation of raceways which do not conform to the minimum separation criteria, and permit subsequent installation of cables in the nonconforming raceways. (I&E Report 79-07)

(vii) On August 13, 1979, battery room ventilation control panels PNL-VC16, VC17 and VC18 were installed without approved engineering drawings. (I&E Report 79-12)

(viii) On December 3, 1974, concrete placement No. RS-4-12, which is classified as a moderately massive section, had been exposed to a temperature of 38° F on the second day after placement. This nonconformance had not been identified by Field Quality Control and corrective action had not been taken to determine whether the exposure had adversely affected the concrete nor to prevent repetition of such nonconformance. (I&E Report 79-24)

(ix) On March 9, 1978, installed and inspected hangers had: 7 jam nuts missing from three hangers, 2 loose nuts and bolts on two hangers, 3 lock wires or cotter pins missing on one hanger, 1 spring preset piece missing from one hanger, and 8 1/4" anchor bolt spacing not meeting the 6" specified for one hanger. (I&E Report 78-03)

(x) On May 3, 1978, the pipe to elbow, nuclear class 1, field weld 1E11-IC020-FW 3 was observed in the final accepted condition with areas ground 1/16" below the pipe surface and the weld reinforcement surface was not ground parallel to the adjacent pipe surface in accordance with drawing note of Std-MP-1056-3-3. (I&E Report 78-06)

(xi) On August 9, 1978, the front weld (inside diameter) size on pipe spool slip-on flange 1B21-SLP-211-3-1 was less than 1/4 inch. The back weld (outside diameter) size on pipe spool slip-on flanges 1B21-SLP-211-3-1 and 1B21-SLP-203-3-1 was less than 3/8 inch. (I&E Report 78-12)

(xii) On September 29, 1978, S&W field forces performed welding on a Skewed Tee Joint of angle 54 degrees, or the rectangular tubular pipe support number 1E11-PSA-072 of the residual heat removal system, using non-applicable procedure W70G which was not prequalified for the weld joint configuration. (I&E Report 78-15)

(xiii) On April 5, 1978, the quality assurance manual governing Reactor Controls Incorporated installation activities for the control rod drive system did not identify those responsible for inspection of the system for conformance to drawings (except welding), nor define acceptance criteria for such inspection, nor provide for verification of completion and evaluation of such inspection, nor provide for documentation of the results. One result of the absence of such provisions was the presence of reverse slope and low points on 1 1/2 inch exhaust water headers and charging water headers contrary to the slope specified on drawing FP-12C-5A, without identification and documentation by site staff for RCI engineering resolution and consideration by management. (I&E Report 78-05)

(xiv) The Engineering Field Extension Office reviewed, approved, and issued a repair welding procedure RP-38 titled "Repair of Defects in Weld of Stainless Steel Forging to Carbon Steel Pipe in CRD Penetrations" which was not qualified, as issued, to the applicable code and specification. (I&E Report 77-01)

(xv) During the period of August 16, 1976, to March 4, 1977, the disposition of N&D 922, dated August 16, 1977, and titled "Control Rod Drive Penetrations," was revised from a grind out and reweld repair of the safety related control rod drive penetrations to the welding of sleeves over the dissimilar metal weld joints without a "revised" disposition being issued, reviewed, and approved as required by EAP-15.1, revision 3. (I&E Report 77-17)

(xvi) On February 9, 1977, Field QC had performed a final visual inspection and accepted field weld IE 41-1C179 FW06, although the transition between this weld and pump P-016 did not confirm to the required profile for approximately 200° of its circumference. (I&E Report 77-17)

(xvii) Stone & Webster (S&W) Deficiency Correction Order Nos. 10182E, 10185E and 10187E were issued in November, 1976, to correct identified instances where field routed safety related cable installation within switchgear enclosures did not meet the separation criteria of S&W Specification No. SH1-159. The corrective actions specified and taken by S&W personnel did not include corrective actions to preclude repetition of the nonconformances. As a result, on March 2, 1977, the NRC inspector identified safety related cable installations in switchgear enclosure 1R22*SWG-101-1 and 101-2, 1R22*SWG-103-2 and 1H11*MCB-01-29 and 01-31, which did not meet the specified separations criteria and these nonconformances were repetitious of those previously identified by S&W personnel in November, 1976. (I&E Report 77-05)

(xviii) The constructor's field quality control organization did not initiate followup action for overdue responses to Surveillance Inspection Nonconformance Reports (SIN) as required by Paragraph 4.4.4 of Quality Control Procedure 20.2. The following are instances where the required followup memoranda were not issued to the responsible individuals with copies to the next higher supervisory level.

-- SIN 28-2, Due date March 12, 1976

-- SIN 28-1, Due date April 9, 1976

At the time of this inspection, these reports had not yet been answered. (I&E Report 76-06)

(xix) Vendor documentation for pipe spool 1E11-WR232-2-01 was not reviewed in accordance with Field QC (FQC) Procedure 9.1; and pipe hangers in storage were not protected against deterioration as required by FQC 17.1. (I&E Report 76-01)

(xx) It was found that Air Meter No. 8/12/9 had not been returned for calibration by the due date shown on the "Recall List" and that a Nonconformance and Disposition Report had not been issued as required by FQC 16.1. (I&E Report 75-01)

(n) <u>Corrective Action [particularization of S.C.12(m)]</u>

Contrary to Criterion 16, measures to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformance, are promptly identified and corrected, are inadequate in that:

(i) On September 11, 1981, the inspector observed that Project Procedure P304 does not require report of a possible reportable deficiency. (I&E Report 81-16)

(ii) As of May 25, 1979 S&W specification SH-1-159 and associated change EDCR-F19039 permit installation of raceways which do not conform to the minimum separation criteria, and permit subsequent installation of cables in the nonconforming raceways. (I&E Report 79-07)

(iii) On December 3, 1974, concrete placement No. RS-4-12, which is classified as a moderately massive section, had been exposed to a temperature of 38° F on the second day after placement. This nonconformance had not been identified by Field Quality Control and corrective action had not been taken to determine whether the exposure had adversely affected the concrete nor to prevent repetition of such nonconformance. (I&E Report 79-24)

(iv) The Engineering Field Extension Office reviewed, approved, and issued a repair welding procedure RP-38, titled "Repair of Defects in Weld of Stainless Steel Forging to Carbon Steel Pipe in CRD Penetrations" which was not qualified, as issued, to the applicable code and specification. (I&E Report 77-01)

(v) Stone & Webster (S&W) Deficiency Correction Order Nos. 10182E, 10185E and 10187E were issued in November, 1976, to correct identified instances where field routed safety related cable installation within switchgear enclosures did not meet the separation criteria of S&W Specification No. SH1-159. The corrective actions specified and taken by S&W personnel did not include corrective actions to preclude repetition of the nonconformances. As a result, on March 2, 1977, the NRC inspector identified safety related cable installations in switchgear enclosure 1R22*SWG-101-1 and 101-2, 1R22*SWG-103-2 and 1H11*MCB-01-29 and 01-31, which did not meet the specified separations criteria and these nonconformances were repetitious of those previously identified by S&W personnel in November, 1976. (I&E Report 77-05)

(vi) During the inspection of the placement of safety related concrete for the radwaste building on April 27 and 28, 1976, the constructor's field quality control inspector observed that the batch plant moisture detecting equipment was not operable and he so noted this condition on the checksheet of Surveillance Inspection Plan No. 65 used for his inspection. However, the inspector failed to initiate corrective action to assure the equipment was returned to operable status as required by Paragraph 4.4 of Quality Control Procedure 10.3 and the equipment remained in an inoperable status. (I&E Report 76-08)

(vii) The constructor's field quality control organization did not initiate followup action for overdue responses to Surveillance Inspection Nonconformance Reports (SIN) as required by Paragraph 4.4.4 of Quality Control Procedure 20.2. The following are instances where the required followup memoranda were not issued to the responsible individuals with copies to the next higher supervisory level.

-- SIN 28-2, Due date March 12, 1976

-- SIN 28-1, Due date April 9, 1976

At the time of this inspection, these reports had not yet been answered. (I&E Report 76-06)

(viii) An audit performed by LILCO and documented in Audit Report FA-322 dated November 6, 1975, identified that the constructor was not filing Engineering and Design Change Reports with the applicable specifications and procedures as required by site procedures. A followup audit was performed by LILCO and documented in Audit Report FA-399, dated April 27, 1976, where it was again identified that the same conditions existed. The failure of the constructor to effect corrective action in the above instance is contrary to the requirement that nonconformances be identified and corrective action taken to preclude repetition. (I&E Report 76-06)

(o) Quality Assurance Records [particularization of S.C.12(b), (h) and (l)

Contrary to Criterion 17, the maintenance of records to furnish evidence of activities affecting quality is inadequate in that:

On April 5, 1978, the quality assurance manual governing Reactor Controls Incorporated installation activities for the control rod drive system did not identify those responsible for inspection of the system for conformance to drawings (except welding), nor define acceptance criteria for such inspection, nor provide for verification of completion and evaluation of such inspection, nor provide for documentation of the results. One result of the absence of such provisions was the presence of reverse slope and low points on 1 1/2 inch exhaust water headers and charging water headers contrary to the slope specified on drawing FP-12C-5A, without identification and documentation by site staff for RCI engineering resolution and consideration by management. (I&E Report 78-05)

(p) Audit [particularization of S.C.12(g)]

Contrary to Criteria 18, the system of planned and periodic audits to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program, is inadequate in that:

(i) An audit performed by LILCO and documented in Audit Report FA-32? dated November 6, 1975, identified that the constructor was not filing Engineering and Design Change Reports with the applicable specifications and procedures as required by site procedures. A followup audit was performed by LILCO and documented in Audit Report FA-399, dated April 27, 1976, where it was again identified that the same conditions existed. The failure of the constructor to effect corrective action in the above instance is contrary to the requirement that nonconformances be identified and corrective action taken to preclude repetition. (I&E Report 76-06)

(ii) LILCO audits of certain Stone & Webster site quality control activities were performed at a greater interval than prescribed by LILCO's Quality Assurance Procedure (QAP) 18.2. (I&E Report 74-03)

ATTACHMENT 3

TESTIMONY OF NRC CHAIRMAN PALLADINO AND

NRC OPERATIONS DIRECTOR DIRCKS BEFORE THE

U.S. HOUSE SUBCOMMITTEE ON ENERGY AND THE ENVIRONMENT,

NOVEMBER 19, 1981

TESTIMONY OF

NUNZIO J. PALLADINO

UNITED STATES NUCLEAR REGULATORY COMMISSION

BEFORE THE

SUBCOMMITTEE ON ENERGY AND THE ENVIRONMENT

OF THE

COMMITTEE ON INTERIOR AND INSULAR AFFAIRS

UNITED STATES HOUSE OF REPRESENTATIVES

WASHINGTON, D.C.

NOVEMBER 19, 1981

MR. CHAIRMAN AND MEMBERS OF THE COMMITTEE, I AM PLEASED TO APPEAR BEFORE YOU THIS MORNING TO DISCUSS QUALITY ASSURANCE FOR NUCLEAR POWER PLANTS.

I BELIEVE THAT AN EFFECTIVE QUALITY ASSURANCE (QA) PROGRAM IS A VITAL ELEMENT IN THE MANAGEMENT OF ACTIVITIES THAT MUST BE ACCOMPLISHED DURING THE DESIGN AND CONSTRUCTION OF EACH NUCLEAR POWER PLANT. QUALITY ASSURANCE SHOULD BE USED AS A FORMAL MANAGEMENT TOOL TO ATTAIN THE MUTUALLY COMPLEMENTARY GOALS OF ASSURING THAT THE DESIGN IS CORRECT AND THAT THE PLANT IS CONSTRUCTED IN FULL ACCORD WITH THE DESIGN. TO BE EFFECTIVE, A QA PROGRAM MUST HAVE THE FULL SUPPORT AND ATTENTION OF THE NUCLEAR INDUSTRY MANAGERS RESPONSIBLE FOR DESIGN AND CONSTRUCTION.

THE NRC LICENSING AND INSPECTION AND ENFORCEMENT PROCESSES ARE AIMED AT ASSURING THAT AN EFFECTIVE QA PROGRAM IS ESTABLISHED AND IMPLEMENTED TO PROVIDE THE NECESSARY CONFIDENCE THAT EACH NUCLEAR POWER PLANT FULLY SATISFIES NRC REQUIREMENTS.

AFTER REVIEWING BOTH INDUSTRY AND NRC PAST PERFORMANCE IN QA, I READILY ACKNOWLEDGE THAT NEITHER HAVE BEEN AS EFFECTIVE AS THEY SHOULD HAVE BEEN IN VIEW OF THE RELATIVELY LARGE NUMBER OF CONSTRUCTION-RELATED DEFICIENCIES THAT HAVE COME TO LIGHT. HOWEVER, RECOGNIZING THAT THERE IS A PROBLEM IS THE FIRST STEP TO FIXING IT. I HOPE THAT OUR TESTIMONY TODAY WILL DEMONSTRATE NRC'S RESOLVE TO DEAL FORCEFULLY WITH CONSTRUCTION RELATED DEFICIENCIES AND THE QA PROBLEMS THEY REVEAL.

MR. CHAIRMAN, ACCOMPANYING ME TODAY IS MR. WILLIAM DIRCKS, EXECUTIVE DIRECTOR FOR OPERATIONS, MR. HAROLD R. DENTON, DIRECTOR OF THE OFFICE OF NUCLEAR REACTOR REGULATION, MR. RICHARD DEYOUNG, DIRECTOR OF THE OFFICE OF INSPECTION AND ENFORCEMENT, MR. JAMES G. KEPPLER, REGIONAL ADMINISTRATOR OF NRC REGION III, AND MR. JOHN COLLINS, REGIONAL ADMINISTRATOR OF NRC REGION IV. MR. DIRCKS WILL PRESENT THE REST OF NRC'S WRITTEN TESTIMONY, AFTER WHICH WE WILL BE PREPARED TO ANSWER ANY QUESTIONS YOU MAY HAVE.

TESTIMONY OF WILLIAM J. DIRCKS

BEFORE THE

SUBCOMMITTEE ON ENERGY AND THE ENVIRONMENT

OF THE

COMMITTEE ON INTERIOR AND INSULAR AFFAIRS

UNITED STATES HOUSE OF REPRESENTATIVES

WASHINGTON, D.C.

QUALITY ASSURANCE FOR NUCLEAR PLANTS UNDER CONSTRUCTION

THIS TESTIMONY ADDRESSES THE ADEQUACY OF QUALITY ASSURANCE AS IT APPLIES TO NUCLEAR POWER PLANTS UNDER CONSTRUCTION, WHY IDENTIFIED CONSTRUCTION OR QUALITY ASSURANCE DEFICIENCIES HAVE NOT BEEN DETECTED ON A MORE TIMELY BASIS, AND ACTIONS BEING TAKEN TO SOLVE RECOGNIZED PROBLEMS.

THE NRC LOOKS TO THE POWER PLANT OWNERS, THE UTILITIES THEMSELVES, TO TAKE THE LEADERSHIP ROLE IN ASSURING THE QUALITY OF THEIR PLANTS AND OPERATIONS. THIS REQUIRES HEAVY EMPHASIS AND ACTIVE INVOLVEMENT OF TOP LICENSEE MANAGEMENT IN QA PROGRAMS. CAREFUL ATTENTION IS REQUIRED IN THE SELECTION OF ENGINEERING SPECIFICATIONS AND QA PROCEDURES AND PRACTICES FOR EACH TASK AND THEIR IMPLEMENTATION BY THE WORKERS ON THE JOB. MOST IMPORTANTLY, THERE MUST BE ADEQUATE RESOURCES OF QUALIFIED PERSONNEL AT MANAGEMENT, OPERATING, AND STAFF LEVELS.

NRC ASSESSES THE PERFORMANCE OF THE UTILITIES AND THEIR MAJOR CONTRACTORS DURING THE DESIGN AND CONSTRUCTION PHASES. THE NRC DOES NOT ATTEMPT TO REDO THIS WORK OR INSPECT IT COMPLETELY SINCE THE NRC RESOURCES ON A PARTICULAR PLANT ARE ONLY A SMALL FRACTION OF WHAT WE REQUIRE A UTILITY TO DEVOTE TO INSPECTION, QUALITY CONTROL, AND QUALITY ASSURANCE. THE NRC'S REGIONAL OFFICES CARRY OUT A SAMPLING INSPECTION PROGRAM AIMED AT DETERMINING COMPLIANCE WITH THE PROGRAMMATIC COMMITMENTS. THE REGULATORY REQUIREMENTS PLACE THE MAJOR INSPECTION RESPONSIBILITIES FOR QUALITY ASSURANCE

ON THE LICENSEE'S CONTRACTORS, WHICH ARE IN TURN INSPECTED AND AUDITED BY THE LICENSEE'S STAFF. THE NRC'S EFFORT IS AN AUDIT AND OVERVIEW OF THE LICENSEE'S AND ITS CONTRACTORS' QUALITY ASSURANCE ACTIVITIES. IN CARRYING OUT THESE INSPECTION ACTIVITIES, NRC INSPECTIONS COVER APPROXIMATELY 1-5 PERCENT OF THE INSPECTION ACTIVITIES PERFORMED BY THE LICENSEE AND ITS CONTRACTORS.

THE NRC'S QUALITY ASSURANCE REQUIREMENTS ARE CONTAINED IN APPENDIX B TO PART 50 OF TITLE 10 OF THE CODE OF FEDERAL REGULATIONS, "QUALITY ASSURANCE CRITERIA FOR NUCLEAR POWER PLANTS AND FUEL REPROCESSING PLANTS." THESE CRITERIA PROVIDE A BASIS UPON WHICH THE NRC JUDGES THE ACCEPTABILITY OF QA PROGRAMS. THE CRITERIA OF APPENDIX B APPLY TO ALL ACTIVITIES AFFECTING SAFETY-RELATED FUNCTIONS OF NUCLEAR POWER REACTOR STRUCTURES, SYSTEMS, AND COMPONENTS.

QUALITY ASSURANCE IS DEFINED IN OUR REGULATIONS AS "ALL THOSE PLANNED AND SYSTEMATIC ACTIONS NECESSARY TO PROVIDE ADEQUATE CONFIDENCE THAT A STRUCTURE, SYSTEM, OR COMPONENT WILL PERFORM SATISFACTORILY IN SERVICE." WHAT THIS MEANS IS THAT - FOR ITEMS HAVING SAFETY SIGNIFICANCE IN A NUCLEAR POWER PLANT:

- THE DESIGN IS VERIFIED TO BE CORRECT AND TO INCLUDE APPROPRIATE REGULATORY REQUIREMENTS;

- o PROCUREMENT DOCUMENTS CONTAIN ADEQUATE INFORMATION AND ARE VERIFIED;

- o INSPECTION OF PARTS, MATERIALS, AND PROCESSES ARE TIMELY AND ADEQUATE;

- o DEFICIENCIES IN DESIGN, CONSTRUCTION AND INSTALLATION ARE IDENTIFIED AND APPROPRIATELY REMEDIED;

- o THE QA PROCESS IS AUDITED AND REPORTED TO AN ORGANIZATIONAL LEVEL CAPABLE OF ASSURING EFFECTIVE CORRECTIVE MEASURES;

- o RECORDS ARE KEPT WHICH CLEARLY DEMONSTRATE SUFFICIENCY OF ACTIVITIES AFFECTING QUALITY; AND

- o THE ORGANIZATIONS PERFORMING QA FUNCTIONS HAVE SUFFICIENT INDEPENDENCE AND AUTHORITY TO IMPLEMENT THESE ACTIVITIES.

THIS DISCUSSION WILL FOCUS ON SOME EXPERIENCES THAT HAVE AND CONTINUE TO GENERATE WIDESPREAD PUBLIC INTEREST. SPECIFICALLY, THERE HAVE BEEN SOME SERIOUS QUALITY ASSURANCE BREAKDOWNS WITH BROAD REPERCUSSIONS AT THE MARBLE HILL, MIDLAND, ZIMMER, SOUTH TEXAS, AND DIABLO CANYON CONSTRUCTION SITES.

MARBLE HILL

IN 1979, WEAKNESSES WERE IDENTIFIED IN THE PROGRAM FOR THE PLACEMENT OF CONCRETE AND RELATED QUALITY ASSURANCE MEASURES AT THE MARBLE HILL NUCLEAR PLANT CONSTRUCTION SITE IN SOUTHERN INDIANA.

WE INVESTIGATED THESE PROBLEMS WHEN A CONCRETE WORKER RAISED ALLEGATIONS THAT HONEYCOMBING, VOIDS AND SURFACE DEFECTS WERE BEING IMPROPERLY PATCHED. THESE ALLEGATIONS, WHICH WERE SUBSEQUENTLY SUBSTANTIATED, LED TO A BROADER INVESTIGATION THAT ADDRESSED OTHER AREAS OF WORK AT THE SITE. ABOUT THE SAME TIME, CODE COMPLIANCE PROBLEMS WERE IDENTIFIED BY THE INDIANA BOILER CODE INSPECTOR AND THE NATIONAL BOARD OF BOILER AND PRESSURE VESSEL INSPECTORS.

THESE EVENTS LED TO A HALTING OF ALL SAFETY-RELATED WORK AT THE SITE IN AUGUST 1979 -- A MOVE TAKEN BY THE UTILITY AND CONFIRMED BY AN NRC ORDER. WORK WAS NOT PERMITTED BY THE NRC TO RESUME UNTIL DECEMBER 1980, SOME 16 MONTHS LATER, WHEN THE UTILITY'S QUALITY ASSURANCE PROGRAM --AND THAT OF ITS CONTRACTORS -- HAD BEEN SUBSTANTIALLY UPGRADED AND THE ADEQUACY OF COMPLETED CONSTRUCTION WORK HAD BEEN VERIFIED. DELAYS IN CONSTRUCTION AND EFFORTS TO CORRECT THESE AND OTHER PROBLEMS ARE ESTIMATED TO HAVE COST THE UTILITY HUNDREDS OF MILLIONS OF DOLLARS.

MIDLAND

IN THE CASE OF THE MIDLAND FACILITY IN MICHIGAN, EXCESSIVE SETTLEMENT OF THE DIESEL GENERATOR BUILDING WAS OBSERVED IN 1978. THE UNEXPECTED SETTLING WAS SUBSEQUENTLY ATTRIBUTED TO INADEQUATE AND POORLY COMPACTED SOIL UNDER THE BUILDING. FURTHER INVESTIGATION BY THE LICENSEE REVEALED THAT OTHER SAFETY-RELATED SYSTEMS AND STRUCTURES WERE AFFECTED. ALL OF THESE SYSTEMS AND STRUCTURES WERE NEARING COMPLETION AT THE TIME THE PROBLEM WAS DISCOVERED. THE NRC'S INVESTIGATION DETERMINED THAT DESIGN AND CONSTRUCTION SPECIFICATIONS HAD NOT BEEN FOLLOWED DURING PLACEMENT OF THE SOIL FILL MATERIALS AND THAT THERE WAS A LACK OF CONTROL AND SUPERVISION OF THE SOIL PLACEMENT ACTIVITIES BY THE UTILITY AND ITS CONTRACTORS. THE COSTS ASSOCIATED WITH ASSURING PROPER SOIL COMPACTION AND DEMONSTRATING THE ADEQUACY OF THE PLANT DESIGN ARE SIGNIFICANT. THE MATTER HAS STILL NOT BEEN RESOLVED AND THE ISSUES ARE CURRENTLY BEING LITIGATED BEFORE AN NRC HEARING BOARD.

ZIMMER

AT THE ZIMMER FACILITY IN SOUTHERN OHIO, THE NRC HAS BEEN INVESTIGATING ALLEGED QUALITY ASSURANCE IRREGULARITIES SINCE JANUARY OF THIS YEAR. THIS INVESTIGATION EFFORT, WHICH IS STILL ONGOING, STARTED WITH ALLEGATIONS FROM A COUPLE OF SOURCES, BUT SOON BROADENED TO MANY WORKERS AND EX-WORKERS. TO DATE WE HAVE

INTERVIEWED APPROXIMATELY 100 INDIVIDUALS AND EXPENDED OVER 250 MAN-DAYS ONSITE PURSUING THESE ALLEGATIONS.

THE CURRENT INVESTIGATION HAS IDENTIFIED A NUMBER OF QUALITY ASSURANCE-RELATED PROBLEMS AT THE ZIMMER SITE. THE MAJORITY OF THE PROBLEMS IDENTIFIED TO DATE FOCUS ON THE INEFFECTIVENESS OF CONTROLS IMPLEMENTED BY THE LICENSEE AND ITS CONTRACTORS FOR ASSURING THE QUALITY OF WORK PERFORMED. IN THAT REGARD, NUMEROUS DEFICIENCIES HAVE BEEN FOUND CONCERNING TRACEABILITY OF MATERIALS, HANDLING OF NONCONFORMANCE, INTERFACE BETWEEN CONSTRUCTION AND QUALITY CONTROL, QUALITY RECORDS, AND THE LICENSEE'S OVERVIEW OF ONGOING WORK.

THE IMPACT OF THE IDENTIFIED QUALITY ASSURANCE DEFICIENCIES ON THE ACTUAL CONSTRUCTION HAS YET TO BE DETERMINED. AN EXTENSIVE REVIEW OF THE AS BUILT PLANT IS CURRENTLY BEING PERFORMED. LIMITED INDEPENDENT MEASUREMENTS WERE PERFORMED BY THE NRC IN SELECTED AREAS OF CONCERN IN AN ATTEMPT TO CHARACTERIZE THE ACTUAL SAFETY SIGNIFICANCE OF THESE DEFICIENCIES. ALTHOUGH A FEW PROBLEMS REQUIRING CORRECTIVE ACTION WERE IDENTIFIED, THE MAJORITY OF THE TESTS AND EXAMINATIONS DISCLOSED NO HARDWARE PROBLEMS.

BEFORE THE PLANT CAN BE LICENSED A COMPREHENSIVE QUALITY CONFIRMATION PROGRAM WILL HAVE TO BE CONDUCTED AND IDENTIFIED PROBLEM

AREAS RESOLVED. BY ITSELF, WITHOUT FACTORING IN ANY REWORK, THE QUALITY CONFIRMATION PROGRAM WILL BE BOTH COSTLY AND TIME CONSUMING. THE EFFECT OF THIS ON THE CONSTRUCTION SCHEDULE OF THE PLANT REMAINS TO BE DETERMINED.

SOUTH TEXAS

IN JANUARY 1981, HOUSTON LIGHTING AND POWER COMPANY (HL&P) INITIATED A DESIGN REVIEW OF THOSE PORTIONS OF THE ENGINEERING DESIGN WORK PERFORMED BY BROWN AND ROOT, INC., (B&R) FOR THE SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION (STP). THE PURPOSE OF THIS REVIEW WAS TO ASCERTAIN THE OVERALL ADEQUACY OF THE STP DESIGN. QUADREX CORPORATION WAS ASKED TO ASSIST HL&P IN A REVIEW OF THE FOLLOWING B&R TECHNICAL DISCIPLINES:

- CIVIL/STRUCTURAL
- COMPUTER PROGRAMS AND CODES
- ELECTRICAL/INSTRUMENTATION AND CONTROL
- GEOTECHNIC
- HEATING, VENTILATING AND AIR CONDITIONING
- MECHANICAL
- NUCLEAR ANALYSIS
- PIPING AND SUPPORTS/STRESS AND SPECIAL STRESS
- RADIOLOGICAL CONTROL

THE LICENSEE MET WITH QUADREX CORPORATION FOR THE FIRST TIME ON JANUARY 16, 1981, AND SEVERAL OTHER TIMES IN JANUARY AND FEBRUARY 1981, TO PLAN THE REVIEW. THE REVIEW BY QUADREX INVOLVED 12 ENGINEERING CONSULTANT PERSONNEL WHO SPENT MORE THAN SIX WEEKS IN AUDITING B&R DESIGN ENGINEERING DOCUMENTS AND INTERVIEWING VARIOUS B&R DISCIPLINE ENGINEERS. THE REPORT ON THE QUADREX EFFORT DATED MAY 1981, WAS SUBMITTED BY THE LICENSEE TO THE NRC LICENSING HEARING BOARD ON SEPTEMBER 28, 1981. BREIFLY, THE QUADREX REPORT FOUND THAT BROWN & ROOT APPARENTLY FAILED TO PROPERLY IMPLEMENT THE QA PROGRAM IN THE DESIGN AREA BUT ALSO FAILED TO PROPERLY IMPLEMENT AN OVERALL DESIGN PROCESS CONSISTENT WITH THE NEEDS OF A NUCLEAR POWER PLANT. AS A RESULT VERIFICATION OF DESIGN INFORMATION WAS APPARENTLY NOT PERFORMED IN A TIMELY MANNER, AND REGULATORY COMMITMENTS FOR SAFETY DID NOT APPEAR TO BE FULLY OR PROPERLY IMPLEMENTED TO SATISFY NRC REQUIREMENTS FOR LICENSABILITY.

NRC INSPECTION REPORTS DATING BACK TO 1979 FOUND PROBLEMS AT THE SOUTH TEXAS PLANT SIMILAR TO THOSE IDENTIFIED IN THE QUADREX REPORT. HOWEVER, THE AGENCY'S AUDITS DID NOT SURFACE THE NUMBER OF PROBLEMS SUGGESTED BY THE QUADREX REPORT. THOUGH WE WERE AWARE OF QA PROBLEMS AT SOUTH TEXAS AND HAD CITED THE LICENSEE FOR A BREAKDOWN IN THEIR QA PROGRAM IN APRIL 1980, THE MAGNITUDE OF POTENTIAL PROBLEMS WAS NOT FULLY APPRECIATED UNTIL WE FIRST REVIEWED THE REPORT IN AUGUST OF 1981.

IN LATE SEPTEMBER THE LICENSEE ANNOUNCED THAT BROWN AND ROOT WAS BEING REPLACED BY BECHTEL POWER CORPORATION AS ARCHITECT-ENGINEER. WE INTEND TO CAREFULLY MONITOR HOW BECHTEL INVESTIGATES AND DISPOSES OF THE PROBLEMS SURFACED BY THE QUADREX REPORT.

DIABLO CANYON

AT DIABLO CANYON, THE PACIFIC GAS & ELECTRIC COMPANY (PG&E) PROVIDED INCORRECT INFORMATION TO A EXPERT CONSULTANT, WHO USED THE INFORMATION IN DEVELOPING THE SEISMIC RESPONSE SPECTRA FOR THE DESIGN OF CERTAIN SEISMIC PIPING AND EQUIPMENT RESTRAINTS. OUR INVESTIGATORS HAVE FOUND THAT THERE WAS A LACK OF RIGOR AND FORMALITY IN THE PROCEDURES USED FOR VERIFYING THE ACCURACY OF INFORMATION TRANSFERRED BY PG&E TO ITS CONSULTANTS. THESE PROCEDURES DID NOT COMPLY WITH OUR REQUIREMENTS CALLING FOR VERIFICATION OF DESIGN INFORMATION AT EACH STAGE OF THE PROCESS BY AN INDEPENDENT PERSON QUALIFIED IN THE PERTINENT DISCIPLINES. PROPER QUALITY ASSURANCE CONTROLS WERE NOT EMPLOYED IN TECHNICAL AND PROCUREMENT COMMUNICATIONS WITH SERVICE-TYPE CONTRACTORS. NOR WERE DOCUMENT CONTROLS ADEQUATE TO ASSURE THAT THOSE INVOLVED IN DESIGN HAD READY ACCESS TO THE MOST RECENT INFORMATION AVAILABLE.

BECAUSE OF THE INADEQUACY OF QA CONTROLS OVER DESIGN VERIFICATION, PROCUREMENT AND THE TRANSMITTAL OF DOCUMENTS TO SERVICE CONTRACTORS, THE ACCEPTABILITY OF THE DESIGNS BASED ON THEIR ANALYSES IS NOW IN QUESTION.

AS A RESULT, THE STAFF HAS DECIDED THAT THERE IS SUFFICIENT REASON TO REVIEW THE ENTIRE PROCESS FOR SEISMIC DESIGN; TO REVIEW THE ADEQUACY OF OTHER PLANT DESIGN ASPECTS, PARTICULARLY THOSE THAT WERE BASED ON ENGINEERING INFORMATION DEVELOPED UNDER OTHER SERVICE-TYPE CONTRACTS; AND TO REVIEW THE IMPLEMENTATION OF THE UTILITY QA PROGRAM IN THESE AREAS.

IN LOOKING AT THE MARBLE HILL, MIDLAND, ZIMMER, SOUTH TEXAS, AND DIABLO CANYON PROBLEMS, QUESTIONS HAVE BEEN RAISED AS TO WHY THE LICENSEE'S QUALITY ASSURANCE PROGRAM AND THE NRC INSPECTION PROGRAM HAD NOT IDENTIFIED THE PROBLEMS SOONER. CLEARLY, IN EACH CASE, THERE WAS AN OVERRELIANCE BY THE UTILITY ON ITS CONTRACTORS FOR MAINTAINING A THOROUGH QUALITY ASSURANCE PROGRAM. THE UTILITY'S OWN QA STAFF WAS TOO SMALL TO MAINTAIN SUFFICIENT SURVEILLANCE OVER THE WORK OF CONTRACTORS. IN TWO OF THE CASES WE SAW INSTANCES WHERE THE CONSTRUCTION MANAGEMENT DOMINATED OR CONTROLLED THE QUALITY ASSURANCE PROGRAM AND PERSONNEL. AND, IN EACH OF THE CASES WHERE PROBLEMS HAD BEEN IDENTIFIED, THE CORRECTIVE ACTION TAKEN WAS NOT SUFFICIENTLY BROAD. TOO FREQUENTLY, THE RESPONSE WAS ONE OF TREATING THE SYMPTOM, RATHER THAN FINDING THE BASIC CAUSE AND CORRECTING IT.

IN ANALYZING THE IDENTIFIED PROBLEMS AREAS, ONE CAN COME UP WITH A LIST OF IMMEDIATE CAUSES -- SUCH AS UNQUALIFIED WORKERS OR QC INSPECTORS, FALSIFIED RECORDS, INTIMIDATION OF QUALITY

CONTROL INSPECTORS, LACK OF AUTHORITY, LACK OF COMMUNICATION, INADEQUATE STAFFING LEVELS, INADEQUATE CORRECTIVE ACTION SYSTEMS, LACK OF SUPERVISION, POOR TO NONEXISTENT PROCEDURES, POOR DESIGN AND CHANGE CONTROL, DESIGN ERRORS, INADEQUATE ANALYSES, POOR QUALITY COMPONENTS, AND SO ON. MOST OF THESE CAN BE TRACED TO FAILURE OF QUALITY ASSURANCE DUE TO INEFFECTIVE MANAGEMENT CONTROL OF THE QA PROGRAM. THERE ARE A MYRIAD OF EXCUSES AND REASONS WHY MANAGEMENT FAILS. SOME ARE EXPLICIT FAILURES OF PERFORMANCE OR LACK OF ATTENTION. OTHER FAILURES ARISING FROM POOR ATTITUDES AND PERCEPTIONS ARE DIFFICULT TO IDENTIFY. THE NRC CANNOT TOLERATE THESE DEFECTS BECAUSE OF THEIR POTENTIAL IMPACT IN TERMS OF PUBLIC RISK. IT IS SURPRISING THAT SOME LICENSEES ARE INSUFFICIENTLY CONCERNED ABOUT QUALITY ASSURANCE NOT ONLY BECAUSE OF THE SAFETY IMPLICATIONS BUT ALSO BECAUSE OF THE IMMENSE COST OF MISTAKES AND OF THE RESULTING DELAY IN CONSTRUCTION.

GIVEN THESE INSTANCES OF BREAKDOWNS IN MANAGEMENT CONTROL OF CONSTRUCTION QUALITY AND THE COMMISSION'S DISSATISFACTION, THE ISSUE IS "WHAT ARE WE DOING ABOUT IT?"

WITHOUT DOUBT, THERE HAVE BEEN SHORTCOMINGS IN THE NRC INSPECTION PROGRAM AT CONSTRUCTION SITES. THERE HAVE BEEN CASES WHERE WE HAVE FAILED TO SEE THE BREADTH OR DEPTH OF A PROBLEM. WE IDENTIFIED SPECIFIC VIOLATIONS OF REQUIREMENTS WITHOUT REQUIRING THE CORRECTION OF THE BASIC CAUSE OF THE PROBLEM. ADDITIONALLY,

WE MAY HAVE SPENT TOO LITTLE TIME WITH QUALITY CONTROL INSPECTORS AND CONTRUCTION WORKERS TO GET THEIR VIEWS ON THE IMPLEMENTATION OF QUALITY ASSURANCE ACTIVITIES AT THE SITE. HOWEVER, WE ARE TAKING STEPS TO ASSURE ATTENTION TO CONSTRUCTION QA INCLUDING DESIGNATION OF RESIDENT INSPECTORS AT ALL CONTRUCTION SITES.

THE COMMISSION HAS MADE OR IS CONSIDERING A NUMBER OF CHANGES OF ITS INSPECTION AND ENFORCEMENT PROGRAM TO INCREASE THE EMPHASIS ON IMPLEMENTATION OF QA PROGRAMS. LET ME ADDRESS SIX SPECIFIC ACTIVITIES:

1. AS INDICATED ABOVE, NRC RESIDENT INSPECTORS HAVE BEEN OR WILL BE STATIONED AT ALL CONSTRUCTION SITES WHERE ACTIVE CONSTRUCTION IS PRESENTLY UNDER WAY AND THE PROJECT IS AT LEAST 15 PERCENT COMPLETE. BASED ON OUR EXPERIENCE WITH THE RESIDENT INSPECTION PROGRAM TO DATE, WE BELIEVE RESIDENT INSPECTORS ENHANCE THE NRC'S ABILITY TO MONITOR QUALITY ASSURANCE ACTIVITIES AND IDENTIFY THE SYMPTOMS OF BREAKDOWN IN MANAGEMENT CONTROL.

2. THERE HAS BEEN A TOUGHENING OF THE NRC'S ENFORCEMENT POSTURE OVER THE PAST COUPLE OF YEARS AND THE NRC'S REVISED ENFORCEMENT POLICY HAS PLACED EMPHASIS ON DEALING WITH POOR REGULATORY PERFORMANCE IN THE CONSTRUCTION AREAS.

3. WE HAVE COMPLETED A TRIAL PROGRAM OF TEAM INSPECTIONS WHEREBY SEVERAL NRC INSPECTORS GO TO A CONSTRUCTION SITE FOR TWO TO THREE WEEKS TO DO A BROAD, INTENSIVE INSPECTION OF THE QUALITY ASSURANCE PROGRAM FOR ONGOING WORK. THIS APPROACH ENABLES NRC TO GAIN A TOTAL PROJECT PERSPECTIVE TO A GREATER EXTENT THAN PAST PRACTICE. THE ADVANTAGE OF THIS DETAILED "SNAPSHOT" IS AN ENHANCED ABILITY TO EVALUATE MANAGEMENT EFFECTIVENESS. THE USE OF SUCH INSPECTION TEAMS IS EXTREMELY LIMITED BY THE AVAILABILITY OF INSPECTORS AND FUNDS FOR THIS PURPOSE. WITH ADDITIONAL RESOURCES, WE COULD SEND INSPECTION TEAMS TO EACH CONSTRUCTION SITE TO DO MORE COMPREHENSIVE INSPECTIONS.

4. THE NRC CONSTRUCTION INSPECTION PROGRAM IS UNDER REVISION TO ACCOMPLISH SEVERAL OBJECTIVES. WE ARE RECASTING INSPECTION PROCEDURES TO DELETE INSPECTION ACTIVITIES OF LESSER IMPORTANCE AND TO REDUCE DUPLICATION OF EFFORT BY RESIDENT AND REGIONAL-BASED SPECIALIST INSPECTORS. IN SITUATIONS WHERE INSPECTOR RESOURCES LIMITATIONS PRECLUDE COMPLETING THE ENTIRE INSPECTION PROGRAM, WE ARE ORDERING OUR PRIORITIES SO THAT THE MOST IMPORTANT INSPECTIONS WILL BE COMPLETED.

5. FORMALIZED PERFORMANCE APPRAISALS OF LICENSEE REGULATORY PERFORMANCE ARE BEING CONDUCTED ANNUALLY BY THE NRC (SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE PROGRAM). THE APPRAISALS, WHICH REVIEW THE COLLECTIVE NRC EXPERIENCE WITH EACH POWER REACTOR, BRING THE BROAD ISSUES OF PERFORMANCE EFFECTIVENESS TO THE ATTENTION OF SENIOR LICENSEE OFFICIALS.

6. WE ARE NOW USING OUR OWN MOBILE LABORATORY FOR NONDESTRUCTIVE EXAMINATION (NDE) AT CONSTRUCTION SITES. THIS NDE VAN HAS MULTIPLE CAPABILITIES THAT INCLUDE RADIOGRAPH DEVELOPMENT, METALLURGICAL ANALYSIS, AND HARDNESS, ULTRASONIC, DYE PENETRANT AND MAGNETIC PARTICLE TESTING. THE EXAMINATIONS THAT WE PERFORM ARE INTENDED TO CONFIRM QUALITY BASED ON A SELECTIVE SAMPLING APPROACH.

THE COMMISSION IS CONTINUING TO REVIEW ITS RESPONSIBILITIES IN THE NUCLEAR QA AREA IN ORDER TO DEVELOP IMPROVEMENTS IN DEFINING REQUIREMENTS, REVIEWING LICENSEE QA PROGRAMS, AND INSPECTION PRACTICES WHERE THEY ARE CALLED FOR.

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

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

- Relief Valve F025 is not a thermal relief as stated in Note 12.

- The line to Radwaste through valves MO-F040 and F049 is on the opposite side of valve MO-F010 as that shown.

- Cooling water for RHR pumps is Reactor Building Closed Loop Cooling Water, not emergency equipment cooling water.

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

7. FSAR, p. 5.5-22 states that a relief valve on the RHR pump discharge and another on the RCIC steam supply protect the heat exchanger. Contrary to this one relief valve is on the discharge line into the heat exchanger, with two valves intervening from the RHR pump discharge, and the steam supply is from HPCI, rather than RCIC.

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

Your reply to this Deviation should address your plans to meet the provisions of 10 CFR 50.55(d) for bringing the original application for license up to date in its entirety at or about the time of completion of the construction of the facility.

Dated MAY 12 1982

Stewart D. Ebneter
T. T. Martin, Director, Division of
Engineering and Technical Programs

APPENDIX C

OBSERVATIONS

Long Island Lighting Company — Docket No. 50-322

Shoreham Nuclear Power Station — License No. CPPR-95

Based on the results of an inspection conducted on February 8-26, 1982, the following observations were made regarding various licensee programs. These observations are considered weaknesses in the program.

1. No specific requirements were evident for timely incorporation of approved Engineering and Design Change Reports (E&DCR's) into drawings and specifications. As an example, the two flow diagrams for the Residual Heat Removal System used for this inspection were last revised December 10, 1980. There were 34 E&DCR's outstanding against these two drawings at the time of inspection; some date back to 1978. While no violations were identified as a result of the practice, the number of E&DCR's and affected drawings and specifications lead to a concern for timely incorporation of changes as construction nears completion. The primary concern is that drawings be completed and readily useable by plant staff for plants operations.

2. E&DCR F-27961 established requirements for separation of class 1E and non-class 1E electrical cables in transit between raceways. Four examples were found that did not comply with these requirements. The Final Safety Analysis Report description of cable tray separation did not agree with recommendations of Institute of Electrical and Electronics Engineers Standard 384-1974. The licensee is engaged in a major program to ensure adequate electrical separation throughout the plant. The concern is that the program for ensuring electrical separation adequately address all aspects of separation, including redundancy and fire hazard considerations.

3. Proposed Technical Specifications did not include all Residual Heat Removal pipe restraints (snubbers), did not recognize multiple snubbers and did not appropriately classify "high radiation zone" or "especially difficult to remove" snubbers. The proposed Technical Specifications also omitted important, plant unique, safety-related systems such as Reactor Building Closed Loop Cooling water and Low Pressure Coolant Injection Motor Generator Sets. The concern is that submittals reflect the complete detail of the constructed plant.

4. Carbon steel bolting used on copper-nickel flanged piping, particularly the service water system, was observed to be corroded. The condition had been identified by nonconformance reports and a corrective action plan was verbally outlined by licensee representatives. There is a concern that the corrective action may not be thorough and may not preclude recurrence.

You are requested to inform this office within 30 days of receipt of actions taken or planned to address these observations.

MAY 1 2 1982

Date

Stewart D Ebneter

T. T. Martin, Director, Division of
Engineering and Technical Programs

NOTICE

12 MAY 1982

[illegible] WAS NOT CONTAINED PROPRIETARY
CLEARANCE [illegible] CFR 2790

U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No. 50-322/82-04

Docket No. 50-322

License No. CPPR-95 Priority - Category B

Licensee: Long Island Lighting Company

175 East Old Country Road

Hicksville, New York 11801

Facility Name: Shoreham Nuclear Power Station

Inspection at: Shoreham, New York

Inspection conducted: February 8-26, 1982

Inspectors:	L H Bettenhausen L. H. Bettenhausen, Ph.D., Chief, Test Programs Section	5/5/82 date signed
	S. K. Chaudhary S. K. Chaudhary, Reactor Inspector	5/5/82 date signed
	L H Bettenhausen for J. C. Higgins, Senior Resident Inspector	5/5/82 date signed
	H.H. Nicholas H. H. Nicholas, Reactor Inspector	5/5/82 date signed
	L H Bettenhausen for R. J. Paolino, Reactor Inspector	5/5/82 date signed
Approved by:	S. D. Ebneter S. D. Ebneter Chief, Engineering Inspection Branch	5/10/82 date signed

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

TABLE OF CONTENTS

DETAILS

1. Persons Contacted

Long Island Lighting Company (LILCO)

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

Stone and Webster Engineering Corporation (S&W)

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

General Electric Company

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

Burns and Roe Corporation

* R. Grunseich, Senior Licensing Engineer

U. S. Nuclear Regulatory Commission

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

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

2. Inspection Purpose; Summary of Results

2.1 Purpose and Scope of Inspection

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

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

2.2 Summary of Inspection Results

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

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

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

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

<u>The Residual Heat Removal System - Comparisons with Codes, Standards, Regulations, Specifications and Drawings</u>

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

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

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

3.1 RHR Piping and Pipe Supports

3.1.1 Discussion

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

3.1.2 Pipe Supports

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

3.1.3 Piping

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

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

3.1.4 Inspection Findings

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

-- FM-15A-12: Drains from *P-005A and B drawn as going to CRW, but reference locations on Drawing M-10149 are not correct.

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

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

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

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

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

3.2 Instrumentation, Controls and Electrical Power

3.2.1 Discussion

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

3.2.2 Verification of Panels 018 and 021

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

From Panel E11*PNL-021

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

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

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

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

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

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

From Panel E11*PNL-018

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

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

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

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

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

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

3.2.3 Verification of Instrument Line Routing

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

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

3.2.4 Electrical Logic and Wiring

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

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

3.2.5 Swing Bus Design

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

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

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

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

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

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

3.2.6 Electrical Cable Separation

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

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

-- 1TC400R, 1TC404R and 1TC411N

-- 1TK794B, 1TK785N

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

-- 1TX706N, 1CC706NO, 1CC705RR, and 1CC705BL.

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

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

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

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

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

3.2.7 Conduit Sealing

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

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

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

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

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

3.3 RHR System Controls

3.3.1 Discussion

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

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

3.3.2 Labeling

The inspector noted the following RHR system labeling deficiencies:

-- Annunciator 1122 has a seemingly contradictory label.

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

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

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

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

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

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

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

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

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

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

3.3.3 Manual Initiation

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

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

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

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

3.3.4 Override or Bypass indication

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

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

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

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

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

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

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

3.3.5 Remote Shutdown Panel

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

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

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

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

3.4 Containment Isolation Valves (CIV's)

3.4.1 Discussion

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

-- Proper valve type, location, and arrangement.

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

-- Adequate description in the proposed Technical Specifications.

-- Proper physical condition and protection from damage.

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

3.4.2 General Design Criterion 56

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

3.4.3 Leak Rate Testing

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

3.4.3.1 Reverse Direction Testing

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

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

3.4.3.2 Downstream Venting

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

- penetration X10A: valves G11*MOV-639 and the G11 check valves.
- penetrations X-42/XS-5: valves E11*01V-3144, MOV-55A & B, and MOV-56A&B.
- penetrations X-8A&B: valves E11*MOV-042A&B.

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

3.4.4 CIV Timing

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

4. Support Systems

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

4.1 Service Water System

4.1.1 Description

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

4.1.2 Physical Inspection

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

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

4.1.3 Corrosion of Carbon Steel Bolts

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

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

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

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

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

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

4.2 Reactor Building Closed Loop Cooling Water (RBCLCW)

4.2.1 Description

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

4.2.2 Pipe Supports and Restraints

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

4.2.3 Piping

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

Two discrepancies were found:

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

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

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

4.2.4 RBCLCW System Controls

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

4.2.4.1 Labeling

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

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

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

4.2.4.2 Manual Initiation

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

4.3 Emergency Diesel Generators

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

4.3.1 Emergency Diesel Engine Modifications

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

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

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

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

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

4.3.2 Emergency Diesel Generator Support Systems

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

4.3.2.1 EDG Service Water System

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

4.3.2.2 EDG Air Start System

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

4.3.2.3. Pipe Supports and Restraints in EDG Rooms

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

4.3.2.4 Fuel Oil Storage and Transfer Systems

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

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

4.3.2.5 Physical Structures and Surrounding Areas

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

-- Yard Filling Station

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

-- Fuel Oil Storage and Transfer Rooms

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

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

-- Emergency Diesel Generator Rooms

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

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

-- Yard Area Around FOS&T & EDG Rooms

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

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

4.3.2.6 Cleanliness and Fire Prevention

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

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

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

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

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

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

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

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

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

4.4 ECCS Discharge Line Fill System

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

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

4.5 Leakage Return System

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

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

The inspector had no further questions regarding the LRS.

5. Management Controls

5.1 "As Built" Program

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

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

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

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

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

5.2 Design Change and Noncomformance Control

5.2.1 Engineering and Design Coordination Report (E&DCRs)

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

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

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

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

5.2.2 Non-conformance and Disposition Reports

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

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

5.3 Proposed Technical Specifications

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

5.3.1 Snubber Table

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

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

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

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

5.3.2 Plant Unique Features

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

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

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

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

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

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

5.5 Housekeeping

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

In the inspection of systems in the Reactor Building, the Primary Containment and the Screenwell Pump House, examples of many items of construction such as piping, valves, nuts, bolts, studs, hardware, steel rails, angle iron, I-beams, lumber, rags, paper and the like were observed in unwanted locations such as MOV housings and cable trays. Many of these items showed no evidence of have been moved in a period greater than the three weeks of February 8-26, 1982. Of particular concern were the many observed instances of dirt, grit and debris, in, on and around electrical boxes, cabling and operating shafts and stems of motor operated valves. On February 25, 1982, the inspector observed grinding being conducted next to an RHR pump with grit and dust settling around the unprotected RHR pump shaft. In the Screenwell Pump House, both rooms had excessive hardware and equipment not in use. A table was set up against electrical cabinet R24MCC1110 with hardware stacked on it. A ladder was set against electrical cabinet MCC-1184. Both cabinet faces were labeled with NEC-OSHA signs stating "... area in front of electrical panels must be kept clear for thirty six inches in front of cabinet ...".

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

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

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

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

5.6 Cabinet Seismic Mounting in Control Room

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

-- Area Radiation Monitor, 236400(911)

-- TIP Control, 236X401 (913)

-- Startup Neutron Monitor, 236C402 (936)

-- Power Range Monitor, 236X403 (936)

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

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

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

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

6. Unresolved Items

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

7. Exit Interview

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

8. References

8.1 Drawings

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

FC-28A8-3, Revision 3, Control Room & Diesel Generator Room Equipment FDN Details

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

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

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

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

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

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

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

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

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

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

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

Elementary Diagram - RHR Annunciators - Drawing No. 10AN834

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

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

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

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

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

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

Cable Pull Tickets for Cable Nos.

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

Instrumentation Tubing Drawings Nos.

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

Equipment Location Drawing Nos.

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

8.2 <u>Isometric Drawings</u>

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

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

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

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

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

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

E11RHR-E-2821-1C-69 Approved 1-13-82

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

Large Bore Piping Isometrics

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

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

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

Small Bore Piping Isometrics

P-33L9-1

P-33MO-1

P-33NI-1

P-33N2-1

8.3 Specifications

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

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

Specification No. SH1-089
Diesel Generator Sets

Specification No. SH1-159
Electrical Installation

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

8.4 Bulletins, Circulars and Information Notices

8.4.1 Bulletins

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

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

8.4.2 Circulars

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

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

8.4.3 Information Notices

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

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

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

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

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

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

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

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

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

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

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

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

8.6 Manuals and Codes

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

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

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

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

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

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

8.7 Procedures

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

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

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

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

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

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

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

ATTACHMENT 5

SUMMARY OF SHOREHAM "VIOLATIONS" AND "DEVIATIONS"

CITED BY NRC IE PROGRAM (1974 TO FEBRUARY, 1982)

TABLE 5-1

VIOLATIONS AND DEVIATIONS CITED BY NRC IE PROGRAM

REQUIREMENT	1974	1975	1976	1977	1978	1979	1980	1981	1982	CRITERIA TOTAL
Appendix B:										
1										0
2						1				1
3	1			1			2	1		5
4										0
5	8	3	4	5	6	7	2	5		40
6	1							1		2
7										0
8										0
9			1		3	3				7
10					1					1
11										0
12										0
13			2							2
14										0
15	1									1
16	1		3	1		2				7
17	1								1	2
18										0
Deviations		1							1	2
50.55(e)	3	1								4
GDC 55-57								1		1
Yearly Totals	16	5	10	7	10	13	4	8	2	75

1. I&E INSPECTION 74-02

<u>VIOLATION</u>: Contrary to Criteria 5 and 17 of 10 CFR 50, Appendix B, there were no records documenting periodic inspection of cadwelding activities as required by Site Procedure QC-14.2, Section 4.2.

<u>RESOLUTION</u>: LILCO letter to AEC (5/6/74) reports that field QA/QC personnel were directed to establish and maintain records of in-process inspection of cadwelds. Change 1 to Field Quality Control Procedure QC-14.2 was issued by Stone and Webster on March 6, 1974, requiring documentation of the results of in-process inspection of cadwelding activities on a Cadwelding Inspection Report (Form T-S-12). Documentation of the inspection was commenced on March 6, 1974. Furthermore, LILCO Audit Plans for Cadwelding have been revised to require verification that in-process inspections are documented on Form T-S-12.

I&E Inspection 74-05 notes the new procedures. <u>Resolved</u>.

2. I&E INSPECTION 74-02

VIOLATION: Contrary to Criterion 5 of 10 CFR 50, Appendix B, several series B cadweld sleeves were found without protection and with rust formation. (Specification No. SH1-64 and Cadweld Procedure No. W-300A requirements.) About 60 type B sleeves were found unprotected and without rebar installed. Fifty other type B sleeves, without rebar, were inadequately protected to prevent entry of moisture. Also, two type B sleeves with rebar installed for the reactor pedestal support were observed with advanced rust on the rebar and inside the sleeve.

RESOLUTION: LILCO letter to AEC (5/6/74) says that the sleeves have now been cleaned and adequately protected. Per the General Procedure of Cadwelding, LILCO states that that was required for sleeves without rebar installed to minimize the amount of cleaning required during the setting up in preparation for firing. However, congestion in the area of the B series sleeves in question made continuous protection with plastic covers impractical. Therefore, the General Procedure for Cadwelding has been revised to recognize this condition and to emphasize the requirement for cleaning of the sleeve prior to firing. Furthermore, a training session on cadwelding has been conducted for S&W QC inspectors where the requirement to closely monitor the protection and cleaning of cadweld sleeves and rebar ends has been re-emphasized. In addition, LILCO Audit Plans for Cadwelding have been revised to require verification that protection and cleaning requirements have been implemented.

I&E Inspection 74-05 notes new procedures. Resolved.

3. I&E INSPECTION 74-02

VIOLATION: Contrary to Criteria 15 and 16 of 10 CFR 50, Appendix B, visually rejected B series cadweld splices were not being documented in Nonconformance and Disposition (N&D) reports to identify the cause of the deficiencies, the corrective actions taken, or to report these defects to the appropriate level of management.

RESOLUTION: LILCO letter to ACE (5/6/74) contends that the visually rejected B series cadweld splices do not constitute a significant condition adverse to quality, within the intent of Criterion 16, since all cadweld splices are inspected and every rejected splice is removed and replaced. Hence, there is no requirement that they be formally documented on N&D reports. However, LILCO claims, all LILCO rejected cadweld splices are documented on a Cadweld Control Record showing, in part, the welder symbol, bar size, position, and reasons for rejection. S&W has been directed, LILCO adds, to develop a system for identifying and reporting significant trends adverse to quality in cadwelding activities, which will be implemented by 5/15/74.

LILCO followup letter to AEC (5/24/74) reports that the instructions for completing the FQC Cadweld Control Record did refer to rejected splices as nonconformities. Further, S&W FQC Procedure QC-6.1 does contain the requirement that nonconformities be documented on an N&D Report. Nevertheless, LILCO argues, it had not been their intention that all visually rejected cadweld splices be documented on N&D Reports for the reasons cited in their letter of May 6, 1974. Accordingly, the instructions for completing the Cadweld Control Record were revised on May 21, 1974, to delete the specific reference to rejected cadweld splices as nonconformities. Thus, full compliance has been achieved.

Rejected cadweld splices will continue to be documented on the Cadweld Control Record, and continued effectiveness of the Cadweld Control Program will be verified through the LILCO audit program

I&E Inspection 74-05 confirms the new procedures and reports no further deficiencies upon inspection. Resolved.

4. I&E INSPECTION 74-03

VIOLATION: Contrary to the requirement of 10 CFR 50.55(e), LILCO failed to report thin walls in certain portions of pipe pieces of the Main Steam Line which necessitated return to the manufacturer for rework.

RESOLUTION: LILCO Letter to AEC (5/23/74) claims that the matter was not reportable -- "if the problem had remained uncorrected, it would not have adversely affected safety and, therefore, is not reportable."

AEC letter to LILCO (6/14/74) disagrees and orders LILCO to report the deficiency despite the utility's position.

LILCO letter to AEC (7/11/74) still argues that the matter was not reportable. However, it reports that reworked pipe has been inspected by LILCO and S&W before being returned to the site.

LILCO letter to AEC (8/28/74) discusses the report on RPU repairs and qualifications.

I&E Inspection 74-08 examines documents and has no further questions. Resolved.

5. I&E INSPECTION 74-03

VIOLATION: Contrary to the requirement of 10 CFR 50.55(3), LILCO failed to report the discovery of rusty water in the RCIC vacuum pump which necessitated return for rework.

RESOLUTION: LILCO letter to AEC (5/23/74) claims that the situation was not reportable -- "rusty water in the RCIC vacuum pump, had it gone uncorrected, could not have adversely affected the safe operation of the plant." LILCO further reports, however, that the pump will be returned to the manufacturer for clean up and repair for reasons of vendor equipment warranty.

AEC letter to LILCO (6/14/74) has no further questions on the matter. Resolved.

6. I&E INSPECTION 74-03

VIOLATION: Contrary to Criterion 5 of 10 CFR 50, Appendix B, LILCO audits of certain Stone and Webster site quality control activities were performed at a greater interval than prescribed by the licensee's Quality Assurance Procedure (QAP) 18.2.

RESOLUTION: LILCO letter to AEC (5/23/74) reports that audits in question have been completed and reported. Furthermore, it ensures that all areas of the site audit surveillance program are brought and maintained up-to-date. In addition, LILCO reports periodic review of the quality history of Shoreham site activities to make most effective use of QA personnel. It also states that the utility will assign additional QA personnel to the site.

I&E INSPECTION 74-05 reports no deficiencies were found upon examination of audits in question. Moreover, it notes LILCO is taking applications for QA personnel.

I&E Inspection 74-08 notes hirings for QA personnel. Resolved.

7. I&E INSPECTION 74-03

VIOLATION: Contrary to Criterion 5 of 10 CFR 50, Appendix B, receipt inspection of KSM studs is being performed by S&W without an approved procedure.

RESOLUTION: LILCO letter to AEC (5/23/74) reports that receipt inspection at the site has been performed in accordance with S&W Field QC Procedure QC-9.1, Quality Control Receiving Inspection, which was approved prior to use. To supplement the procedure, Field QC Instruction 9.1-001, Stud Material Receiving Instructions, was approved by S&W on 5/15/74 and issued for use at the site. To prevent recurrence, LILCO assures that Stone and Webster site QC personnel responsible for issuing site instructions have been instructed to follow Quality Assurance Directive QAD-5.13, Preparation and Processing of Quality Control Instructions. Inspection personnel have been instructed to operate only to approved procedures and instructions. Continued compliance will be verified through the LILCO site audit program.

I&E Inspection 74-05 documents inspection of the procedure. Resolved.

8. I&E INSPECTION 74-03

VIOLATION: Contrary to Criterion 5 of 10 CFR 50, Appendix B, KSM studs have been accepted by Stone and Webster receipt inspection although the required certification is not to the standard specified in the Purchase Order.

RESOLUTION: LILCO letter to AEC (5/23/74) states that the procurement requirements of AWS D1.0-66 and AWS D2.0-66 to which the KSM studs were certified are the same as the requirements of AWS D.1-72 referenced in the purchase documentation. E&DCR, No. 581, dated May 16, 1974, has been approved by Stone and Webster documenting this equivalency. However, recertification of all KSM studs to the requirements of AWS D1.1-72 has been requested and received, and all future acceptance will be based upon certification to AWS D1.1-72. In addition, LILCO assures that personnel responsible for receipt inspection have been cautioned to accept material only when all the requirements of the procurement documents have been met and that continued compliance will be verified through the site audit program.

I&E INSPECTION 74-05 examined S&W E&DCR 581 and notes AWS D1.1-72 as equivalent to AWS D1.0-66 and AWS D2.0-66. Resolved.

9. I&E INSPECTION 74-05

VIOLATION: Contrary to Criterion 5 of 10 CFR 50, Appendix B, Access to the RPV and the head was uncontrolled. (It should have been fenced in, but the RPV had a very low fence and the reactor head had no fence at all.)

RESOLUTION: LILCO letter to AEC (3/15/74) reports that RPV storage requirements have been reviewed and S&W's procedure, "Site Storage of Reactor Pressure Vessel" (5/19/74), has been revised to reflect more realistic requirements for preservation of the integrity of the RPV while in storage at the Shoreham site. S&W Revision 2 (8/13/74) specifies that the storage area be protected by a steel and/or wooden barricade. Furthermore, FQC will monitor the implementation to ensure conformance.

I&E Inspection 74-06 notes that the NSSS and the A-E are reviewing the requirements for storage for the RPV.

I&E Inspection 74-08 notes S&W Revision 2 of "Site Storage of Reactor Pressure Vessel" (8/13/74) has been approved with comments by the NSSS and forwarded for interim use on 9/2/74. Revision 3 of the procedure which incorporates the NSSS comments is available at the site, although not approved. No deficiencies have been identified during examination of this procedure and its implementation. Resolved.

10. I&E INSPECTION 74-05

VIOLATION: Contrary to Criterion 5 of 10 CFR 50, Appendix B, leakage of nitrogen from RPV tank was approximately 12 cubic feet per hour, more than 200 cubit feet per day.

RESOLUTION: LILCO letter to AEC (8/14/74) reports that RPV storage requirements have been reviewed and S&W's procedure, "Site Storage of Reactor Pressure Vessel" (4/19/74), has been revised to reflect more realistic requirements for preservation of the integrity of the RPV while in storage at the Shoreham site. S&W Revision 2 (8/13/74) specifies that a nitrogen leakage rate is not required to assure that interior surfaces of the vessel and head are blanketed with nitrogen. FQC will monitor implementation to assure conformance.

I&E Inspection 74-06 notes that the NSSS and the A-E are reviewing the requirements for storage for the RPV.

I&E Inspection 74-08 notes S&W Revision 2 (8/13/74) has been approved with comments by the NSSS and forwarded for interim use on 9/2/74. Revision 3 of the procedure which incorporates the NSSS comments is available at the site although not approved. No deficiencies have been identified during examination of this procedure and its implementation. Resolved.

11. I&E INSPECTION 74-05

VIOLATION: Contrary to Criterion 5 of 10 CFR 50, Appendix B, the nitrogen pressure, as indicated by a manometer, was less than one-half the amount specified (.04 psig.).

RESOLUTION: LILCO letter to AEC (8/15/74) reports that RPV storage requirements have been reviewed and S&W's procedure, "Site Storage of Reactor Pressure Vessel" (4/19/74), has been revised to reflect more realistic requirements for preservation of the integrity of the RPV while in storage at the Shoreham site. S&W Revision 2 (8/13/74) specifies that the internal nitrogen pressure be maintained at a positive value. FQC will monitor implementation to assure conformance.

I&E Inspection 74-06 notes that the NSSS and the A-E are reviewing the requirements for storage for the RPV.

I&E Inspection 74-08 notes S&W Revision 2 (8/13/74) has been approved with comments by the NSSS and forwarded for interim use on 9/2/74. Revision 3 of the procedure which incorporates the NSSS comments is available at the site, although not approved. No deficiencies have been identified during examination of this procedure and its implementation. Resolved.

12. I&E INSPECTION 74-08

VIOLATION: Contrary to the requirement of 10 CFR 50.55(e), LILCO failed to report extensive damage to the removable head of the containment structure due to improper post-weld heat treatment of certain weld seams.

RESOLUTION: LILCO letter to AEC (11/27/74) reports that the initial report of this deficiency was made verbally to the Region I inspectors at its Hicksville offices on 10/11/74. The thirty-day written report was forwarded to Dr. D. F. Kruth, DORO, Washington, D.C. with a copy to the NRC on 11/11/74. In this report, LILCO explains that repairs, to be carefully monitored, will be completed by 12/16/74. In addition, LILCO states in its letter that it will emphasize to its personnel and QA to be more strict and more alert. It will also examine past reports to ensure that no other past nonconformances are reportable.

I&E Inspection 75-01 notes: 1) Inspector reviewed documents; 2) LILCO examined earlier nonconformance reports and found no other that needed reporting, 3) at a meeting of leading LILCO engineers and at a meeting of S&W project engineering staff the matter was discussed; and 4) training sessions were held for S&W engineering staff on reporting of significant problems for preventative action. Resolved.

13. I&E INSPECTION 74-08

VIOLATION: Contrary to Criterion 6 of 10 CFR 50, Appendix B, identification of the wrong addendum to SH1-75 as the latest issued on S&W specifications index was observed (although the Architech-Engineer's specification SH1-75 had been revised to include Addendum 3, and this revised specification had been issued to the containment construction contractor and this contractor was performing his work in accordance with this Addendum; the contractor's QA manual which ostensibly controlled his quality-related activities, still identified specification SH1-75, Addendum 2 as the appropriate specification).

RESOLUTION: During the course of I&E Inspection 74-08 where the violation was noted, a revision to the QA manual which identified Addendum 3 of SH1-75 as the applicable specification, was furnished to the site. The item was subsequently considered resolved and no further action by LILCO was required. Resolved.

14. I&E INSPECTION 74-09

VIOLATION: Contrary to Criterion 3 of 10 CFR 50, Appendix B, the procedures and equipment of the test laboratory at the site were not inspected by a qualified national authority for competence in performance of the required tests because concrete specification SH1-42 wrongly identified ACI-72 as the applicable code to follow. ACI-72 did not require the inspection of the laboratory test site and, therefore, did not conform to the PSAR requirements.

RESOLUTION: LILCO letter to AEC (1/23/75) states in part that LILCO issued revisions to specifications in 1972, including 1972 edition of ACI 301 code; the newer code omitted earlier references in the 1966 edition to the requirement for outside testing and inspection by a national authority, so LILCO left it out of the specifications.

LILCO letter AEC (1/28/75) states in part that it does not believe code specifies outside inspection of test facility, so no violation has occurred.

AEC letter to LILCO (3/19/75) reviews the LILCO 1/23/75 letter, states that AEC believes that the '72 ACI code does require the outside certification.

I&E Inspection 75-05 reports that LILCO has asked for an emergency inspection of test facility by a national authority but was denied. Still awaiting inspection. LILCO letter to AEC (4/30/75) reports that LILCO has asked Cement and Concrete Reference Labs (CCRL) of the National Bureau of Standards to inspect concrete, aggregate, and rebar testing facilities. LILCO still takes issue, however, with the ruling that this is required by ACI 301-72 code.

I&E Inspection 75-10 reports that the inspection was performed by CCRL on 8/6/75. NRC inspector views results and has no further questions. Resolved.

15. I&E INSPECTION 75-01

VIOLATION: Contrary to Criterion 5, 10 CFR 50, Appendix B, it was found that Air Meter No. 3/12/9 had not been returned for calibration by the due date shown on the "Recall List" and that a Nonconformance and Disposition Report had not been issued as required by FQC 16.1.

RESOLUTION: Prior to completion of the inspection, LILCO reported that: 1) the air meter had been recalled, examined, and found to be within the required calibration tolerances; 2) upon notification of the deficiency, N&D report No. 246 was initiated in accordance with FQC Procedure 16.1 and because of the satisfactory results of the calibration check, it was dispositioned "Accept as is" and formally closed; and 3) LILCO also determined that failure to comply with the procedure requirements resulted from the absense of both the calibration facility supervisor and the responsible QC engineer.

LILCO letter to NRC (3/14/75) confirms the points made above and clarifies that failure to return the air meter for calibration in a timely manner was due to a misinterpretation of instructions. To prevent recurrence, LILCO reports that verbal clarification of instructions was promptly given to the responsible personnel followed by written confirmation.

I&E Inspection 75-04 confirms the above. Resolved.

16. I&E INSPECTION 75-03

VIOLATION: Contrary to Criterion 5, 10 CFR 50, Appendix B, QC documentation of Class I concrete preplacement inspections did not provide or reference specified quantative acceptance criteria for construction joint preparation and the inspection sheets were incorrectly filled out with respect to certain construction joint, keyway and water-stop inspection items.

RESOLUTION: LILCO letter to NRC (4/30/75) states that instructions have been issued by the Superintendent of Field Quality Control requiring inspection personnel to identify, by reference on the inspection report, the Drawing(s), Specification or other documents used to determine the inspection attributes.

The errors noted in preparation of concrete preplacement inspection reports consisted of recording several items as Satisfactory when they were, in fact, Not Applicable. This occurred because inspection personnel prepared final reports from notes taken during the inspection and relating primarily to Unsatisfactory items, if any. Inspection personnel are now required to complete their inspection reports at the time of the inspection. Furthermore, continuing surveillance of inspection records will be maintained to ensure compliance with instructions.

I&E Inspection 75-07 reviewed memos and various reports/documents per the corrected procedures and noted no deficiencies. Resolved.

17. I&E INSPECTION 75-03

DEVIATION: Contraty to LILCO's commitment in PSAR Section V-2.5.1.2, it has eliminated construction joint preparation for Class 1 concrete placements in the truncated cone portions of the primary containment wall.

The justification for this in the E&DCR was stated as a construction convenience which would save money, improve the work schedule and provide letter working conditons for the contractors. The E&DCR carried the notation that the change did not represent a change to the PSAR. The inspector stated that the above deviation without prior approval by the Directorate of Reactor Licensing was unacceptable. (This is not considered as a noncompliance item, but rather as a deviation from a PSAR commitment.)

RESOLUTION: Letter of LILCO to NRC (4/30/75) - LILCO believes that the roughness of the as-placed concrete is sufficient for bonding; that the preparation is not needed as long as the surface is cleaned first. They don't feel that they violated the PSAR commitment. They also say that the "construction convenience" excuse given in the above I&E report was offered by a contractor and was not an engineering justification. LILCO says they passed on the E&DCR without comment regarding the PSAR change. They say that they have been keeping a running list of changes which will appear in the FSAR.

I&E Inspection 76-02 - FSAR clarifies the alternative means of joint preparation by LILCO. Resolved.

18. I&E INSPECTION 75-04

VIOLATION: Contrary to 10 CFR Part 50.55a, nondestructive examination of the casting for reactor recirculation system valve No. B31-F023(A) did not include 100% radiographic inspection.

This infraction had the potential for causing or contributing to an occurrence related to health and safety.

RESOLUTION: LILCO letter to NRC (5/1/75) takes exception to this notice of violation. It contends that the codes and regulations in effect on the date of the order for the reactor recirculating system valves, did not require 100% radiography of the valve castings. LILCO does assure, however, that it will have those areas of the body and bonnet casting for valve B31-F023A, which were not radiographed initially, examined radiographically prior to its installation.

NRC letter to LILCO (5/15/75) declares that its position on the matter remains as previously stated. It notes LILCO's followup plans and announces that additional radiographic records will be examined during a subsequent inspection.

I&E Inspection 75-10 notes examination of various documents, memos, the valve, RRSV B31-F023A, and radiographs. Upon completion, the inspector had no further questions. Resolved.

19. I&E INSPECTION 75-05

VIOLATION: Contraty to Criterion 5, 10 CFR 50, Appendix B, the dew point of the nitrogen blanket within the RPV was not maintained within the limits established by the storage procedure and the records showing dew point vs outside air temperature were not maintained as required by the Storage Procedures.

RESOLUTION: LILCO letter to NRC (5/19/75) contends that the temperature maintenance is not an absolute requirement. According to procedures cited by the NRC LILCO argues, "Dew Point, ..., should be maintained between a high of 10°F below outside air temperature and as low as is practicable". The minimum 10°F difference between the nitrogen dew point and the outside air temperature is desirable to help prevent the accumulation of moisture within the pressure vessel, but it is not a mandatory requirement. It is LILCO's position, therefore, that there has been no noncompliance in this instance. Further, periodic checks have shown no evidence of moisture within the vessel. Per the second part of the violation being a) a plot of the dew point and air temperature against time was not available for review by the inspector; and b) a log showing periods of time when the dew point reading was within 10°F, LILCO reports that the nitrogen blanket dew point and the outside air temperature had been checked and recorded as specified by the storage procedure. The plot of the dew point and air temperature against time and log showing days when the dew point was within 10°F of the air temperature have been completed and are being maintained up-to-date.

I&E Inspection 75-07 notes that lacking records had been compiled based on readings previously taken and were maintained for current readings. In addition, periodic inspections of the vessel had shown no evidence of moisture.

The inspector also examined a memorandum dated May 14, 1975, from the Superintendent of Construction which instructed the responsible Assistant Superintendent to assure compliance with requirements of the RPV Storage Procedure and specifically to assure that a check of vessel moisture is made monthly, dew point readings are taken weekly and that the required charts are maintained. Resolved.

20. I&E INSPECTION 76-01

VIOLATION: Contrary to 10 CFR 50, Appendix B, Criterion 5, vendor documentation for pipe spool 1E11-WR232-2-01 was not reviewed in accordance with Field QC (FQC) Proceudre 9.1; pipe hangers in storage were not protected against deterioration as required by FQC 17.1; and nonconforming pipe hangers were not segregated as required by FQC 9.1.

RESOLUTION: This item was resolved prior to completion of the inspection by establishment of an area for storage of reject hangers and movement of the reject hangers into this area.

LILCO letter to NRC (2/17/76) reports that its review of vendor documentation for pipe spool 1E11-WR232-2-01 had failed to detect a missing signature on the NPP-1 Data Report. A properly signed copy of the Data Report has, however, been obtained and placed in the files. Furthermore, all safety-related pipe hangers in outdoor storage showing evidence of corrosion are being cleaned and recoated with preservative. In addition, LILCO reports that an area for storage of rejected pipe hangers has been established and the nonconforming pipe hangers have been moved into this area. To prevent recurrence, LILCO reports having all NPP-1 Data Reports reviewed for completeness and accuracy by S&W prior to being forwarded to the site. Also, it says requirements for segregation of nonconforming pipe hangers and for protection and preservation of stored pipe hangers have been re-emphasized to responsible personnel, and conformance to these requirements will be verified on future surveillance inspections of storage areas.

I&E Inspection 76-04 confirms that properly signed Data Report has been obtained. Storage inspection reports were examined as was the actual condition of the hangers in storage. Also noted is the fact that threads of the hangers in storage have been coated with a preservative and that examination and recoating of threads is a continuing program. Resolved.

21. I&E INSPECTION 76-02

VIOLATION: Contrary to Criterion 5, 10 CFR 50, Appendix B, obsolete drawings were not removed from the work area and destroyed as required by QA/QC procedures. A subcontractor was found by the NRC to have obsolete revisions of six control isometric drawings, all marked "void" in red. The subcontractor should have destroyed them, and had in fact signed a Document Transmittal Slip on 1/26/76 that they had been destroyed. Only some copies for historical files are supposed to be retained.

RESOLUTION: LILCO letter to NRC (3/26/76) outlines the new procedures enacted to enable separation of some obsolete drawings for storage in office historical files from the working area copies. It ensures that audits of the controlled drawings will be performed to verify satisfactory compliance with the document control requirements.

I&E Inspection 76-05 notes that some vendor drawings are in poor condition and/or illegible. LILCO says it has a large backlog of new revisions. Delivery to the field of these revisions has been slow and additional people are working overtime on two shifts to alleviate the problem.

I&E Inspection 76-09 notes that some files are found to be improved but an audit of 45 isometric drawings at two piping foremen's desks revealed seven to be obsolete revisions.

I&E Inspection 76-11 notes that no further obsolete drawings have been identified upon additional sample audits. Resolved.

22. I&E INSPECTION 76-06

VIOLATION: Contrary to Criterion 16 of 10 CFR 50, Appendix B, during the inspection of the placement of safety-related concrete for the radwaste building on April 27 and 28, 1976, the constructor's field quality control inspector observed that the batch plant moisture detecting equipment was not operable and he so noted this condition on the checksheet of Surveillance Inspection Plan No. 65 used for his inspection. However, the inspector failed to initiate corrective action to assure the equipment was returned to operable status as required by Paragraph 4.4 of Quality Control Procedure 10.3 and the equipment remained in an inoperable status.

RESOLUTION: LILCO letter to NRC (7/23/76) reports that a nonconformance report was issued on the inoperable equipment and closed out after the equipment was returned to operable status. To prevent recurrence, it assures that responsible FQC personnel have been reinstructed to ensure compliance with FQC procedures requiring corrective action to be initiated when nonconformances are discovered during surveillance inspections and requiring follow-up memoranda to be issued when the response to a nonconformance report is overdue.

I&E Inspection 76-09 notes examination and identification of the inoperability of the moisture detecting equipment and verifies that concrete control is based on actual test of moisture content. Instructions have been sent to the batch plant to repair or replace and calibrate the moisture detecting equipment. Resolved.

23. I&E INSPECTION 76-06

VIOLATION: Contrary to Criterion 16, 10 CFR 50, Appendix B, the constructor's field quality control organization did not initiate followup action for overdue responses to Surveillance Inspection Nonconformance Reports (SIN) as required by Paragraph 4.4.4 of Quality Control Procedure 20.2. The following are instances where the required followup memoranda were not issued to the responsible individuals with copies to the next higher supervisory level.

- SIN 28-2, Due date March 12, 1976
- SIN 28-1, Due date April 9, 1976

At the time of this inspection, these reports had not yet been answered.

RESOLUTION: LILCO letter to NRC (7/23/76) reports that a review of all open SINs by FQC revealed that responses were overdue only on those identified by the NRC and that these had been answered prior to the FQC review. To prevent recurrence, it assures that responsible FQC personnel have been reinstructed to ensure compliance with FQC procedures requiring corrective action to be iniatiated when nonconformances are discovered during surveillance inspections and requiring follow-up memoranda to be issued when the response to a nonconformance report is overdue.

I&E Inspection 76-09 notes examination of S&W Memo SNPS-QC-701 outlining the new procedures. Also inspected was the Log of Surveillance Inspections upon which this matter was considered closed. Resolved.

24. I&E INSPECTION 76-06

VIOLATION: Contrary to Criterion 16, 10 CFR 50, Appendix B, an audit performed by LILCO and documented in Audit Report FA-322, dated November 6, 1965, identified that the constructor was not filing Engineering and Design Change Reports with the applicable specifications and procedures as required by site procedures. A followup audit was performed by LILCO and documented in Audit Report FA-399, dated April 27, 1976, where it was again identified that the same conditions existed. The failure of the constructor to effect corrective action in the above instance is contrary to the requirement that nonconformances be identified and corrective action taken to preclude repetition.

RESOLUTION: LILCO letter to NRC (7/23/76) reports that the repetitive nonconformances noted on the Field Audit reports have been corrected. To prevent recurrence, it assures that FQA and FQC personnel have been instructed to ensure that action taken to prevent recurrence is stated, where applicable, in the responses on Field Audit Transmittal/Response (AT/R) forms and verified through the Field Audit program.

I&E Inspection 76-12 notes that the A T/R form had been revised to require that the response by the audited agency include "Preventive Action" to be taken in order to prevent recurrence of the nonconformance. According to LILCO, the revised forms had been in use since September, 1976. An audit of A T/R forms showed that: 1) preventive action was described; or 2) identified as not applicable in each case. Resolved.

25. I&E INSPECTION 76-06

VIOLATION: Contrary to Criterion 5, 10 CFR 50, Appendix B, not all of the safety-related Engineering and Design Change Reports which were originated during March at the site were reviewed by LILCO as required by Paragraph W3.2.2 of Procedure P-305. The following are examples of E&DCRs which were not reviewed by the quality assurance organization.

- F-2847
- F-2848
- F-2855
- F-2862

RESOLUTION: LILCO letter to NRC (7/23/76) states that the E&DCR review program in effect had been revised to require the Project Engineer (or his designee) to determine which safety-related E&DCRs required a Quality Assurance (QA) review and to forward only those to QA with E&DCR Approval Form. However, the revision to Procedure P-305 had not been completely processed and published at the time of inspection. The revised Procedure P-305 has now been published. To prevent recurrence, Engineering QA Procedure EQAP 3.3 has been reviewed to require both review of all E&DCRs forwarded to QA with an E&DCR Approval Form and a sampling inspection of other E&DCRs to provide assurance that the proper E&DCRs are being forwarded to QA for review.

I&E Inspection 76-09 notes examination of documentation regarding changes in LILCO Engineering QA Procedures, and LILCO Project Procedure, both of which address new E&DCR procedures. E&DCR Log for July, 1976 is also reviewed. Resolved.

26. I&E INSPECTION 76-07

VIOLATION: Contrary to Criterion 13, 10 CFR 50, Appendix B, the 480 volt switchgear 1R23*SWG101 and its associated transformer 1R23*T101 were not maintained in a clean condition as required by Electrical Installation Specification SH1-159. At the time of the inspection, the plastic cover installed for protection of the equipment was torn and was too small to completely cover the equipment, resulting in deposits of dust and debris on the relay contacts and transformer windings.

RESOLUTION: LILCO letter to NRC (8/19/76) reports that the 480 volt switchgear 1R23*SWG101 and its associated transformer 1R23*T101 have been properly cleaned and covered. To prevent recurrence, the contractors' personnel have been instructed to provide adequate covering or other protection to maintain cleanliness of all motor control centers, switchgear and transformers in accordance with the applicable specification. FQC is maintaining surveillance in this area to ensure continuing compliance.

I&E Inspection 76-09 notes examination of various documents concerning the violation and confirms new protection, new instructions and reports of cleaning have been accomplished. Resolved.

27. I&E INSPECTION 76-09

VIOLATION: Contrary to Criterion 13, 10 CFR 50, Appendix B, storage and preservation of equipment did not meet the requirements of Construction Site Instruction 4.6, in that:

1. Protection against corrosion of electrical connectors for the Control Rod Drive Hydraulic Control Units in the Reactor Building was not provided to preclude deterioration; and

2. Storage and storage surveillance for protection against corrosion of controls for RCIC Pump No. IE 51-P-015 in the Reactor Building was not conducted as required.

RESOLUTION: LILCO letter to NRC (10/8/76) states that the electrical connectors for the Control Rod Drive Hydraulic Control Units are being cleaned and wrapped to protect them against dirt, dust and moisture, and a surveillance inspection of these units has been completed. The electrical controllers on the RCIC pump have been cleaned, dessicant has been placed in the controllers and the controllers have been covered. A requirement for surveillance checks of electrical components of the pump has been added to the Equipment Storage History Card. To prevent recurrence, it assures that frequency of surveillance inspections by FQC has been increased to verify that storage, maintenance and inspection of electrical equipment after its removal from indoor heated storage is being accomplished in accordance with Construction Site Instruction 4.6.

I&E INSPECTION 76-12 notes inspection of newly protected HCU connectors. RCIC pump No. IE51-P-015 and RCIC lube oil cooler IE51-E-039 are noted as clean and protected. New procedures are also reviewed. Resolved.

28. I&E INSPECTION 76-11

VIOLATION: Contrary to Criterion 9, 10 CFR 50, Appendix B, fillet welds for attachment of beam seats in the reactor building at Elevation 146 did not meet the dimensional requirements of specification SH1-152.

RESOLUTION: Prior to completion of the inspection, all beam seat welds in question were re-inspected and noted as repaired. Repair and re-inspection is documented on QC Inspection Report (QCIR) dated Sptember 30, 1976. Resolved.

29. I&E INSPECTION 76-12

VIOLATION: Contrary to Criterion 5, 10 CFR 50, Appendix B, on November 9, 1976, it was observed that the refueling cavity liner surface was contaminated with iron which resulted from thermal cutting operations of carbon steel structural components adjacent to the liner.

RESOLUTION: LILCO letter to NRC (1/14/77) reports that the refueling cavity liner surface is being cleaned and reinspected with satisfactory results to date. To prevent recurrence, it assures that personnel working in the area have been instructed to exercise care when performing cutting, burning or welding operations. When possible, such operations will be conducted away from the refueling cavity liner. When it is necessary to perform such operations close to the liner, the liner surface will be protected, and it will be reinspected upon completion of the work.

I&E Inspection 78-03 reviewed documents. Also, it notes that 1) QC says stainless steel will be protected during thermal cuts of adjacent carbon steel; and 2) an S&W E&DCR provided changes in procedures to emphasize requirements. Resolved.

30. I&E INSPECTION 77-01

VIOLATION: Contrary to Criterion 3, 10 CFR 50 Appendix B, LILCO's Engineering Field Extension Office reviewed, approved, and issued a repair welding procedure RP-38, titled "Repair of Defects in Weld of Stainless Steel Forging to Carbon Steel Pipe in CRD Penetrations" which was not qualified, as issued, to the applicable code and specification.

RESOLUTION: Prior to the completion of the inspection, LILCO obtained from the responsible contractor a welding procedure qualification record which permitted qualification of the repair procedure. Furthermore, the persons responsible for implementation of the engineering QA requirements which should have eliminated such errors have been formally notified and the actions documented in a S&W memorandum to the LILCO QA manager (1/20/77).

NRC letter to LILCO (2/10/77) states that while federal regulations require LILCO to submit promptly a response to the notice of noncompliance, it notes that this item was corrected prior to the completion of the NRC inspection and, therefore, no response with respect to this matter is required. Resolved.

31. I&E INSPECTION 77-05

VIOLATION: Contrary to Criterion 16, 10 CFR 50 Appendix B, Stone and Webster (S&W) Deficiency Correction Order Nos. 10182E, 10185E and 10167E were issued in November, 1976, to correct identified instances where field routed safety-related cable installation within switchgear enclosures did not meet the separation criteria of S&W Specification No. SH1-159. The corrective actions specified and taken by S&W personnel did not include corrective actions to preclude repetition of the nonconformances. As a result, on March 2, 1977, the NRC inspector identified safety-related cable installations in switchgear enclosure 1R22*SWG-101-1 and 101-2, 1R22*SWG-103-2 and 1H11*MCB-01-29 and 01-31, which did not meet the specified separations criteria and these nonconformances were repetitious of those previously identified by S&W personnel in November, 1976.

RESOLUTION: LILCO letter to NRC (5/18/77) disputes the categorization of this item as an "infraction", arguing that it does not consider that there has been any repetition of a previously identified significant condition adverse to quality. LILCO also reports in the letter that tie wraps which bound safety-related cables to non-safety-related cables in switchgear enclosures 1R22*SWG-101-1, and 101-2, 1R22*SWG-103-2, 1H11*MCB-01-29 and 01-31, have been removed. In addition, LILCO reports that the electrical contractor has been directed to conduct a S&W approved training program for craft personnel, Construction Site Instruction 10.1 is being revised, and the final inspection requirements of FQC Procedure 12.1A are being further expanded to prevent recurrence.

NRC letter to LILCO (6/7/77) responds to LILCO's protest of "infraction" categorization. It states that although the conditions identified in the cases do vary in detail, the nonconformance in each case consists of the failure to comply with separation criteria. Had adequate corrective steps been taken following identification of the first nonconformance, they should have provided for conforming to all established separation criteria and doing so during construction rather than depending upon identification of nonconforming conditions by Field Quality Control following completion of construction. This item, therefore, remains an infraction.

I&E Inspection 77-20 notes that LILCO has conducted training sessions on cable separation criteria. Specifications and procedures have also been revised. It was also observed that nonconforming conditions identified in 77-05 had been corrected. Resolved.

32. I&E INSPECTION 77-12

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, LILCO released Receiving Inspection Reports (RIRs) for construction without evidence of source inspection by Procurement QC (PQC) as required by specification SH1-152. Furthermore, this material was released for construction without receipt of Shipping Release Tags and the S&W Certificate of Compliance (for source inspected items) which is contrary to the requirements of FQC Procedure QC-9.1, "Receiving Inspection."

RESOLUTION: LILCO letter to NRC (9/21/77) reports that specification SH1-152 requires random inspection of shop-fabricated, structural steel after the first two or three shipments. Engineering and Design Coordination Report (E&DCR) No. P2837 further clarifies this Specification to require Procurement Quality Control (PQC) Shipping Release Tags only for those components which were inspected and accepted by PQC prior to release for shipment. Field Quality Control (FQC) procedure QC-9.1 is being revised to permit the use of a Vendor's Certificate of Compliance as objective evidence of quality for those shipments which do not have a PQC Shipping Release Tag or a Stone and Webster Certificate of Compliance. Furthermore, all cognizant personnel have been directed to review the applicable FQC procedures and to reinstruct field inspectors on these requirements.

I&E Inspection 77-20 notes S&W changes in procedures to ensure documentation on PQC of materials. Also examined was documentation for a selected group of structural steel shop orders where it was observed that the Vendor's Certificate of Compliance had been supplied for all orders. Resolved.

33. I&E INSPECTION 77-17

<u>VIOLATION</u>: Contrary to Criterion 5, 10 CFR 50 Appendix B, during the period of August 16, 1976, to March 4, 1977, the disposition of N&D 922, dated August 16, 1977, and titled "Control Rod Drive Penetrations," was revised from a grind out and reweld repair of the safety-related control rod drive penetrations to the welding of sleeves over the dissimilar metal weld joints without a "revised" disposition being issued, reviewed, and approved as required by EAP-15.1, revision 3.

<u>RESOLUTION</u>: LILCO letter to NRC (11/9/77) reports that a revised disposition to N&D922 has been issued, reviewed and approved as required by EAP-15.1, Revision 3. Furthermore, responsible personnel have been instructed relative to N&D revision, review and approval requirements of EAP-15.1, Revision 3.

I&E Inspection 77-23 notes that 1) disposition of the N&D report has been revised and reviewed properly; 2) personnel have been instructed as to N&D report procedures; and 3) that LILCO is attempting to identify other deficient N&Ds. <u>Resolved</u>.

34. I&E INSPECTION 77-16

VIOLATION: Contraty to Criterion 5, 10 CFR 50 Appendix B, on February 9, 1977, Field QC had performed a final visual inspection and accepted field weld IE 41-1C179 FW06, although the transition between this weld and pump P-016 did not conform to the required profile for approximately 200° of its circumference.

RESOLUTION: LILCO letter to NRC (11/9/77) reports that N&D 1385 has been issued documenting the nonconformance and its disposition. FQC personnel have been instructed in the requirements of General Welding Procedure ASME W-100, Section B, Appendix C, Revision 1.

NRC letter to LILCO (12/23/77) states that LILCO failed to define how its actions assure that field weld transition nonconformances do not exist elsewhere.

I&E Inspection 77-23 notes 1) increased training for QC inspectors; and 2) there are several more nonconforming transactions on weld joints of Core Spray (CS) system isolation valves which are not GE furnished. This report states that the problem is broader than LILCO thinks, and must go beyond GE supplied items (LILCO has asserted that the item was supplied by a GE vendor and that the problem is confined to GE supplied items).

LILCO letter to NRC (1/4/78) says that the defect attributed to a weld end preparation on a casting furnished by a GE vendor. An investigation made of welds on other GE-furnished castings has disclosed no additional nonconforming conditions. The investigation was extended subsequent to I&E Inspection 77-23 to cover all transition welds on balance of plant equipment. All casting-to-pipe welds installed to date have been identified and a plan has been developed for the required inspections.

I&E Inspection 78-06 examined records of corrective action described in LILCO letter (1/4/78). The N&D report (No. 1592) requires QC reinspection of all Category I transition welds between unequal outside diameter pipes. Also noted are 1) S&W has conducted a training session for pipe weld QC inspectors; and 2) records show that of 408 welds reinspected, 82 were found to be unacceptable, each requiring weld build-up for repair. (These repairs have not yet been accomplished.) Resolved.

35. I&E INSPECTION 77-23

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on December 6, 1977 two safety-related, non-manual valves, numbers T46-TCV024A and T46-TCF025A, were found uncovered and exposed to the rain in the short-term storage area adjacent to the reactor building. It was determined that the valves had been withdrawn from level "B" storage on November 10, 1977. Project Procedure No. 10, revision 8, and FQC Procedure No. QC17.1, revision B, require that these valves be protected from the weather.

RESOLUTION: LILCO letter to NRC (1/26/78) reports that the two valves were moved indoors on December 7, 1977, and an initial inspection by FQC revealed that they were being properly stored and that there was no visible evidence of damage due to their being exposed to the elements. A follow-up inspection performed prior to installation again revealed no evidence of water damage. The valves in question were inspected and accepted for water-tightness at the vendor's facility prior to shipping. To prevent recurrence, LILCO assures that provisions of the correct procedures will be re-emphasized to appropriate personnel. Furthermore, on-site contractors are being reminded of their responsibility for material removed from the warehouse in transit to its final installation location. FQC will continue to monitor for contractor compliance.

I&E Inspection 78-02 notes that corrective steps and actions to prevent recurrence appeared to be as stated. No additional nonconformances of this kind were identified during a plant tour. Resolved.

36. I&E INSPECTION 77-23

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on December 6, 1977 Level A storage facilities were examined and an accumulation of dust on the shelves and materials stored therein, was observed. Also, two voltage regulator panels, Nos. 1B31-MG-001A and B (Category II) had exposed electrical components; these were covered with an oily film combined with dust. Inspection records showed that there was no inspection between July and December, 1977, for dust.

RESOLUTION: LILCO letter to NRC (1/26/78) reports that personnel access to and activities performed in the Level A storage area have been limited to those which are necessary so that the installed ventilation/filter system may maintain the required dust-free atmosphere. The accumulated dust has been removed from shelves and stored material, and voltage regulator panels 1B31-MG-001A and B (Category II) have been cleaned and provided with a protective covering.

To prevent recurrence, LILCO assures that housekeeping and inspection efforts in the Level A storage area have been increased to preclude repetition. The effectiveness of the corrective and preventive action will be monitored by FQC and further corrective steps will be taken if necessary.

I&E Inspection 78-02 examines documents and notes 1) increased frequency of Level A storage QA inspections to a weekly occurrence; and 2) use of the Level A area for work activities other than storage has been discontinued. Resolved.

37. I&E INSPECTION 78-02

VIOLATION: Contrary to Criterion 5, 10 CFR 50, Appendix B, on February 15, 1978, twenty unused and unreturned low-hydrogen type E7018 weld electrodes of various diameters were found in scattered locations inside the containment drywell where safety-related welding work activities were in progress. This was in addition to significantly greater quantities of partly used electrodes also lying loose throughout these work areas.

RESOLUTION: Prior to completion of the inspection, UNICO Chief Welding Supervisor issued a memo (2/16/78) to all site welding contractors (6) requesting submittals of action plans to "control this problem and prevent its recurrence."

LILCO letter to NRC 4/7/78) reports 2/16/78 memo as corrective action along with follow up meetings held with these contractors to assure that each was fully cognizant of the requirements of the problem and to stress procedures, etc. To prevent recurrence, LILCO assures that the frequency of FQC surveillance in both safety-related and nonsafety-related areas has been increased to monitor and measure the effectiveness of the weld material control programs.

I&E Inspection 78-06 notes examination of site instructions for increased QC, as well as daily records. Upon touring the plant and reactor containment work areas to verify improved conditions, 14 weld electrodes were found in the vicinity of some pipe supports but general electrode control has improved. Still open.

I&E Inspection 78-12 notes examination of weekly QC reports and graphs of loose electrodes found between January and July of 1978 maintained by S&W QC. It is concluded that effective corrective action has been taken in this area. Resolved.

I&E Inspection 79-06 notes that the inspector collected 28 unused welding electrodes while performing the pipe walkdown inspection. LILCO presents him with documentation dealing with weld rod control and what appears to be adequate corrective action. Pending verification, the matter is considered open again. Re-opened.

I&E Inspection 79-07 notes that the contents of one portable electrode oven were found lying loose in a group on a ledge of the reactor shield. There were no welders working in the area and the site QA auditor could not identify the source of the material. New instructions were issued May 24, 1979 requiring return of all electrodes at the end of each shift. New regulations state that three offenses of discrepancies will result in withdrawal of all qualifications for the welder involved. Failure to take disciplinary actions will be regarded as a security matter. Still open.

I&E Inspection 80-08 notes that since implementation of May, 1979 instructions, no loose weld rod has been identified during inspections. Furthermore, a full review of the program indicates that it is effective in controlling weld materials. Resolved.

38. I&E INSPECTION 78-03

VIOLATION: Contrary to Criterion 9, 10 CFR 50, Appendix B, on March 8, 1978, the temperature, as measured using a 200°F temperature indicating crayon, of the RHR pipe field weld E11 IC017-FW-03, schedule 80, 1.031" wall thickness, was less than 200°F during the in-process welding operations.

RESOLUTION: LILCO letter to NRC (5/10/78) reports that the weld had been completed and inspections will be performed in accordance with approved procedures. The completed weld will be stress-relieved in accordance with Courter Procedure NW-100, Appendix D. To prevent recurrence, LILCO assures that on future similar welds, electric resistance coils will be used where possible, to apply preheat. Site Quality Assurance will verify the use of coils and of the proper preheat during the Preweld Inspection and will verify the continued satisfactory operation of the coils during In-Process Inspection. A training session will be held to assure that appropriate supervisory personnel are familiar with the revised preheat requirements.

I&E Inspection 78-06 notes that the in-process field weld 1B21-IC175-FW14 did not have resistance heating elements installed to maintain the 200°F preheat. However, the new requirements for resistance coils had not been issued to the field as of the date of this inspection. LILCO stopped work on the weld joint pending further disposition. The placement of resistance heaters was observed the following day for field weld 1E41-IC181-FW9. This matter remains open.

I&E Inspection 78-12 notes that the inspector observed that correct welding procedures are being used. Pending additional future implementation verification, the matter still remains open.

I&E Inspection 78-16 notes that the inspector verified that preheat controls are being implemented during the welding of large bore piping. Resolved.

39. I&E INSPECTION 78-03

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on March 9, 1978, installed and inspected hangers had: 7 jam nuts missing from three hangers, 2 loose nuts and bolts on two hangers, 3 lock wires or cotter pins missing on one hanger, 1 spring pre-set piece missing from one hanger, and 8 1/4" anchor bolt spacing not meeting the 6" specified for one hanger.

RESOLUTION: LILCO letter to NRC (5/10/78) reports that all hangers previously inspected and accepted will be reinspected, and any nonconforming conditions identified will be corrected. To prevent recurrence, LILCO assures that clarification of certain hanger inspection criteria was obtained from the Field Engineering Office, and a comprehensive training session on hanger inspection criteria was held for responsible Field Quality Control and Construction personnel.

I&E Inspection 78-15 notes examination of corrective actions described in LILCO's 5/10/78 letter. In addition, the inspector re-examined the pipe supports which had previously been identified as having nonconforming conditons. These conditions were found to have been corrected. Also, additional supports hardware was examined and found to be acceptable. Further review of documentation found that procedures contain applicable acceptance criteria. Resolved.

40. I&E INSPECTION 78-05

VIOLATION: Contrary to Criterion 10, 10 CFR 50 Appendix B, on April 5, 1978, the QA manual governing Reactor Controls Incorporated installation activities for the control rod drive system did not identify those responsible for inspection of the system for conformance to drawings (except welding), nor define acceptance criteria for such inspection, nor provide for verification of completion and evaluation of such inspection, nor provide for documentation of the results. One result of the absence of such provisions was the presence of reverse slope and low points on 1 1/2 inch exhause water headers and charging water headers contrary to the slope specified on drawing FP-12C-5A, without identification and documentation by site staff for RCI engineering resolution and consideration by management.

RESOLUTION: LILCO letter to NRC (6/1/78) reports that a copy of pertinent portions of the NRC inspection report has been provided to Reactor Controls Incorporated (RCI) for their review and revision of their quality assurance manual in the light of the findings. Manual revisions submitted by RCI will be reviewed for full compliance to NRC regulations.

Construction on the referenced headers has been stopped, and the discrepancy documented and identified for engineering resolution and consideration by management.

To prevent recurrence, LILCO assures that the need for assuring compliance with all NRC requirements will be reemphasized to personnel responsible for reviewing and accepting contractors' quality assurance programs.

I&E Inspection 78-15 notes examination of documents and review of corrective action outlined in LILCO's 6/1/78 letter as well as interviews with RCI site management and site QC supervisor. Also, revised RCI procedure that specifies final inspections and other related documentation were reviewed. For incorrect pipe slope angle identified previously a "QC Hold Information" form was issued. In-process inspection of installation other than welding is generally not specified on the basis that nearly all piping is prefabricated and shipped to the site from the shop. Resolved.

41. I&E INSPECTION 78-06

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on May 3, 1978, the pipe to elbow, nuclear blass 1, field weld 1E11-IC020-FW 3 was observed in the final accepted condition with areas ground 1/16" below the pipe surface and the weld reinforcement surface was not fround parallel to the adjacent pipe surface in accordance with drawing note of Std-MP-1056-3-3.

RESOLUTION: LILCO letter to NRC (7/7/78) reports that additional weld metal will be deposited where excessive grinding occurred on the pipe surface, and the pipe and weld reinforcement surfaces will be reinspected to assure conformance to approved standards. Other similar weld areas are being reinspected, and any required corrective action, as indicated by inspection results, will be accomplished.

The remaining pipe wall thickness adjacent to weld 1E11-1C020-FW3 was measured ultrasonically and found to be below specified minimum wall requirements. This condition was initially reported to the NRC under 10 CFR 50.55(e) on May 22, 1978. The remaining pipe wall thickness adjacent to other welds being re-inspected as noted above will be similarly measured.

To prevent recurrence, LILCO assures that additional inspection criteria have been issued for ASME III Code Class 1 and 2 welds subject to In-Service Inspection, and a training session was conducted for piping-welding inspectors covering these additional requirements. The acceptance criteria for such welds are being reviewed and will be revised as warranted.

I&E Inspection 79-21 notes examination of records (E&DCR F-14070 and Courter NC Report No. 0175) which showed reinspection of welds which had been ground for surface finish and rework of nonconformances. The inspector had no further questions on the matter. Resolved.

42. I&E INSPECTION 78-12

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on August 9, 1978, the front weld (inside diameter) size on pipe spool slip-on flange 1B21-SLP-211-3-1 was less than 1/4 inch. The back weld (outside diameter) size on pipe spool slip-on flanges 1B21-SLP-211-3-1 and 1B21-SLP-203-3-1 was less than 3/8 inch.

RESOLUTION: LILCO letter to NRC (10/18/78) reports that the results of an initial sampling inspection of fillet welds on slip-on flanges revealed the need to inspect all such welds. Accordingly, all accessible slip-on flange welds will be inspected, and those found deficient will be repaired to meet ASME III requirements. Engineering will provide further instructions regarding welds not accessible for inspection when the number of these welds is determined.

To prevent recurrence, LILCO assures that the slip-on flange welds were shop-fabricated. Therefore, this condition has been brought to the attention of Procurement Quality Assurance (PQA). Additionally, the Project QA Manager has been requested to require that PQA advise the vendor of this condition and assure that the vendor takes proper corrective/preventive action.

I&E Inspection 79-03 notes examination of records and interviews with personnel re: corrective actions outlined by LILCO. Also reviewed were 1) inspections of all suspect flanges was found to be documented on 9 Courter Company Nonconformance Reports (NR 365A through I); 2) Dravo inspection/coordination documentation re: trips to the Shoreham site relative to this item; and 3) various related documentation (correspondence, NR's, QC Inspection reports, etc.). Everything seems to be intact. Resolved.

43. I&E INSPECTION 78-12

VIOLATION: Contrary to Criterion 9, 10 CFR 50 Appendix B, on August 9, 1978, the single-bevel-groove weld joint angles for pipe break restraints FWR 1, 2, and 15 were 30°, although the AWS Structural welding code D1.1, paragraph 2.9, limits such weld joint angles to a 45° minimum.

RESOLUTION: LILCO letter to NRC (10/18/78) reports that prior to the inspection, welding on pipe restraints using the 30° bevel was accomplished using technique sheets W70J and W70H, both of which required the use of a bevel angle of 45° minimum if the techniques were to be considered prequalified by AWS. A review of the essential variables of these technique sheets has shown that they are adequately supported for use with a 30° bevel angle by Welding Procedure Qualification Tests on file. Accordingly, welds made to these technique sheets are acceptable.

To prevent recurrence, LILCO assures that a qualified weld technique sheet (attached to E&DCR P-3071) has been issued to cover all subsequent welding on pipe restraint joints having 30° bevels. Additionally, the requirement that all welding meet the provisions of the applicable code has been reemphasized to all welding contractors.

I&E Inspection 78-16 notes that LILCO issued N&C Report 1835 describing the noncompliance and prescribed the disposition to use "as is." Also noted is LILCO's issuance of a memo (10/17/78) to all welding contractors to be aware of the requirements of AWS D1.1 weld joints and techniques to preclude recurrance of a similar situation. The inspector interviewed piping and hanger supervisors to verify that they had received the 10/17/78 memo and were aware of its contents. Resolved.

44. I&E INSPECTION 78-15

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on September 29, 1978, S&W field forces performed welding on a Skewed Tee Joint of angle 54^{o}, on the rectangular tubular pipe support number 1E11-PSA-072 of the residual heat removal system, using non-applicalbe procedure W70G which was not prequalified for the weld joint configuration.

RESOLUTION: LILCO letter to NRC (11/24/78) reports that all previously accepted tubular pipe supports having Skewed Tee Joints were reviewed, and those having an angle less than 60^{o} have been identified. A welding procedure incorporating the essential variables of Technique W70G, but with expanded angular tolerances, is being qualified. Successful qualification of this procedure will provide justification for accpeting those wleds with angles less than 60^{o} identified in the above review.

To prevent recurrence, LILCO assures that Field Quality Control hanger inspection personnel have been reinstructed to verify the use of properly qualified welding techniques during their in-process inspections, and Construction personnel have also been advised of this problem to insure that correct weld procedures are used on future hanger installations. Further, craft and supervisory contractor personnel are receiving additional training to insure that all personnel involved in hanger welding understand the importance of utilizing the correct qualified welding techniques.

I&E Inspection 79-02 notes examination of evidence of corrective action in the form of N&C's, procedures, E&DCR's, correspondence and other records. It is confirmed that LILCO has taken steps to assure that future welding is conducted in accordance with AWS Code requirements. The previously accepted (FQC) supports have now been rejected and are being held for corrective action. LILCO has deferred action to correct existing nonconformances pending a meeting with AWS in March to clarify interpretation of related AWS Code requirements. In addition, interviews with supervisors and workers have been conducted by the inspector who has determined that LILCO employees are not conversant enough on new requirements to prove that they have been trained.

I&E Inspection 71-12 notes that the inspector has interviewed the cognizant LILCO QA engineer and examined documents relative to S&W actions taken after meeting with the AWS personnel. S&W has issued eleven new weld technique sheets that clarify joint preparation details. Some welds have been identified which have been or will be cut out and reworked to new procedures. These matters are documented in N&D reports and S&W interoffice correspondence. Resolved.

45. I&E INSPECTION 78-16

VIOLATION: Contrary to Criterion 9, 10 CFR 50 Appendix B, on October 25, 1978, general areas of undercut in excess of 1/32" deep were observed on the trucks, bridge, and trolly welds of the reactor building polar crane.

RESOLUTION: LILCO letter to NRC (1/5/79) reports that preliminary inspection of about 30% of the shop welding on the crane has revealed that welds made by automatic welding processes are acceptable, and welds containing defects are limited to those made by the manual shielded metal arc welding process. Accordingly, all load-bearing manual shop welds will be identified, and those which are accessible will be cleaned and inspected. Final corrective action will be determined upon evaluation of the inspection results.

To prevent recurrence, LILCO assures that the manufacturer of the crane and Procurement Quality Assurance have been advised of the inspection finding to permit a review of the welding process and verification of acceptable welds. Further preventive action will be taken if warranted by the evaluation of inspection results.

I&E Inspection 79-02 notes examination of evidence of corrective action which includes various correspondence, an SUW QCIR and N&D Report No. 1925. Inspection of the SMAW welds remains to be completed. The final correction to the identified defective welds remains to be specified. Still open.

I&E Inspection 80-08 notes review of various documentation related to the item including N&D Report No. 1925, QCIR's and various other reports. The inspector also observed the conditions of a random sample of repaired welds and had no further questions. Resolved.

46. I&E INSPECTION 78-16

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on October 19, 1978, the energized safety related 4160 volt switchgear 1R22*SWG103 in the 103 emergency switchgear room, elevation 25', control room building, did not comply with the requirements of procedure C.S.1 13.I. There was an accumulation of dirt inside of the energized switchgear equipment which could create conditions that would adversely affect the quality of the component and the equipment operation.

RESOLUTION: LILCO letter to NRC (1/5/79) reports that dust and dirt have been removed from the switchgear, and protective coverings have been installed to prevent debris from entering the equipment.

To prevent recurrence, LILCO assures that construction site instructions are being revised to more clearly define responsibility for maintenance of cleanliness during the construction and turnover periods, and a quality assurance instruction has been drafted defining controls of housekeeping during checkout, initial operation and pre-operational testing. Field Quality Control and Station Operational Quality Assurance will c[illegible]e to monitor activities in the area to insure that effectiv[illegible]rols have been established and implemented.

I&E Inspection 79-03 notes examination of documents and interviews with personnel confirming corrective actions described in LILCO's 1/5/79 letter. Also noted is the fact that the OQC group has performed intensive inspections of safety-related areas of the plant for housekeeping during 2/79. Furthermore, OQA has established a schedule of routine surveillance. Deficient areas have been identified. Also, OQA has reviewed the station Interim Operating Instructions and have provided comments to LILCO's start-up group for inclusion of requirements for interim maintenance/housekeeping during checkout, initial operation and pre-operational testing. Resolution of such OQA audit findings is required by the site QA program. Resolved.

47. I&E INSPECTION 79-02

VIOLATION: Contrary to Criterion 9, 10 CFR 50 Appendix B, on or about November 11, 1978, weld joint 1B21=IC175-FW6 was heated, for post weld heat treatment, at a rate exceeding the ASME III Code allowable. Specifically, the joint was heated between 640°F and 930°F at a rate of 290°F/hr. while the maximum allowable rate was 225°F/hr.

RESOLUTION: LILCO letter to NRC (3/23/79) reports that records of all welds previously heat treated in accordance with Procedure NW 100, Appendix D, have been reviewed. Those welds where the maximum allowable heating or cooling rates may have been exceeded have been identified, and an engineering evaluation has been initiated. Final corrective action will be defined upon completion of this evaluation.

To prevent recurrence, LILCO assures that an Engineering and Design Coordination Report has been issued clarifying instructions pertaining to material thicknesses to be used in calculating maximum heating and cooling rates during post weld heat treatment of welds joining components of unequal thicknesses. Procedure NW 100, Appendix D, has been revised to reflect these instructions and to incorporate the provisions of the ASME boiler and Pressure Vessel Code, Section III, 1974 Edition, Summer 1975 Addenda, and paragraph NW-4620 of the 1974 Edition.

NRC letter to LILCO (4/27/79) orders LILCO to advise the NRC of the results of the evaluation outlined in LILCO's 3/23/79 letter, when completed.

I&E Inspection 79-10 examines records of tests and LILCO reviews. The Courter Company identified 15 welds where heating/cooling rates were exceeded; these were documented on N&D reports. Engineering interpretation of material thickness for calculation purposes results in narrowing this number down to 4 welds. E&DCR F-20851 justifies these by saying that the PWHT holding time would relieve stresses introduced in the rapid heat up. Courter will visually inspect all welds for distortion of pipes. Still open.

48. I&E INSPECTION 79-04

VIOLATION: Contrary to Criterion 9, 10 CFR 50 Appendix B, in October, 1976, the Courter Company crafts, under direction of Stone and Webster, performed thermal cutting of attachment welds to remove pressure caps from nozzles N3 and N4 of residual heat removal heat exchangers No. 034A and No. 034B, without qualified and approved procedures and apparently without performing preheat required by the applicable specifications.

RESOLUTION: LILCO letter to NRC (8/15/179) states that LILCO's interpretation of the ASME code for this area shows that the ambient temperature (50°F) was sufficient to guarantee adequate pre-heat in this operation. Nevertheless, it reports that all remaining end cap metal and weld metal has been removed by grinding. The ground areas were magnetic particle inspected and all rejectable defects removed by grinding and blending into the surrounding surfaces using care not to violate minimum wall requirements. The ground and blended areas were again magnetic particle inspected and found acceptable.

To prevent recurrence, LILCO assures that Welding Procedures W 100B and W 200B, in effect at the construction site, contain the necessary guidance for use in thermal cutting. Also, an investigation by Field Quality Control has revealed that no additional ASME III nozzles have welded temporary pressure caps.

I&E Inspection 79-12 reviews documentation of corrective actions described in LILCO's 8/15/79 letter, accepts justification and considers the matter closed. Resolved.

49. I&E INSPECTION 79-05

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, as of March 30, 1979, the following Engineering Quality Assurance Procedures (EQAP's) had not been updated though their respective Change Notices were in effect in excess of a calendar year.

-- EQAP 2.3, Revision 2, Change Notice (CN) No. 1, dated April 15, 1977.

-- EQAP 2.8, Revision 0, CN No. 1, Dated April 15, 1977.

-- EQAP 3.3, Revision 2, CN No. 1, dated March 10, 1978.

-- EQAP 4.1, Revision 3, CN No. 1, dated April 18, 1977.

-- EQAP 15.2, Revision 3, CN No. 1, dated July 18, 1977.

-- EQAP 16.1, Revision 3, CN No. 1, dated January 3, 1977.

RESOLUTION: LILCO letter to NRC (5/24/79) reports that all Engineering Quality Assurance Procedures (EQAP's) are being reviewed, revised and reissued as Quality Assurance Procedures (QAP's) to reflect the changes mandated by a recent change in the LILCO organization for Quality Assurance. Applicable Change Notices are being reviewed with each Procedure and will be incorporated into the Procedure or cancelled, as appropriate, during the review.

To prevent recurrence, LILCO assures that the revised QAP 5.1, Quality Assurance Procedures, Instructions, Memoranda and Change Notices, will require that all effective Change Notices be reviewed annually with the applicable QAP and incorporated into the Procedure or cancelled, as appropriate, at that time.

I&E Inspection 79-08 notes work in progress. Still open.

I&E Inspection 79-11 notes work in progress. Still open.

I&E Inspection 80-06 reviews and confirms LILCO's corrective actions in repsonse to this noncompliance. <u>Resolved</u>.

50. I&E INSPECTION 79-06

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, LILCO failed to properly store and maintain the HPCL, CS, and RHR pumps in accordance with established instructions and specifications in that:

a) The high pressure coolant injection (HPCI) turbine and pumps were being exposed to construction dirt. Specifically, the turbine and pump gear type couplings were open and covered with dirt; in excess of seven pipe openings were open and unsealed; one shaft bearing housing was open; and the electrical junction box No. 1JB-580 was open.

b) The mechanical seal piping to the core spray pump (CS) E21-P013B, and the residual heat removal (RHR) pumps E11-P014A, B, and D were also unsealed. Although these pumps are under the construction phase maintenance program, the General Electric Specification 22A2724 applies.

RESOLUTION: Letter of response from LILCO to NRC (6/19/79) cannot be located.

I&E Inspection 79-12 notes that the inspector toured the lowest elevation of the reactor building and examined the emergency cooling equipment in the area of proper storage conditions relative to LILCO's 6/19/79 letter to the NRC. Pipe openings were covered on equipment and general onstruction debris had been reduced in the area. Resolved.

51. I&E INSPECTION 79-06

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, LILCO failed to take prompt corrective action concerning conditions adverse to quality in that:

> The inspector notified the responsible LILCO personnel of the detrimental conditions under which the HPCI unit was being stored on April 9, 1979. The inspector reinspected the area on April 10, 11, and 12, 1979 to see if any corrective actions had been taken. No corrective actions were taken and, in addition, on the reinspection of April 12, 1979, there was water raining down on the unit.

RESOLUTION: Letter from LILCO to NRC (6/19/79) cannot be located.

I&E Inspection 79-12 notes review of a memo sent to construction superintendents regarding current policy for timely correction of conditions adverse to quality. The S&W senior QC inspector was also interviewed. The NRC inspector had no further concerns. Resolved.

52. I&E INSPECTION 79-06

VIOLATION: Contrary to Criterion 9, 10 CFR 50 Appendix B, the visual inspection of the integral pipe lugs 1E21-PSR-040, drawing M-12125-27-3, Pipe Supports of Reactor Core Spray Piping, disclosed a lack of full penetration on the root of "C" weld. The fitup sketch NW-100D, "Types of Attachment Welds for Class 1, 2, and 3 Components," specifies a full penetration weld be employed. A review of the Component Checklist for IE21-PSR-040 disclosed that the final weld inspection was performed on April 5, 1979. This constitutes LILCO's failure to provide a full penetration weld.

RESOLUTION: Letter from LILCO to NRC (6/19/79) cannot be located.

I&E Inspection 80-01 notes review of corrective actions with respect to pipe lug 1E21*PSR-040 as well as actions taken to identify and correct similar nonconforming conditions on other systems. Additional documentation was also reviewed. Furthermore, the inspector discussed the measures used to identify, inspect and repair attachment welds with LILCO. Still open.

I&E Inspection 80-19 notes review of the summary sheet of all full penetration welds requiring corrective action and the NR's and E&DCR's referenced there. All corrective actions had been completed and NR's closed except NR No. 1206B for trunnion No. E11-PSA-002. The inspector also examined Courter & Company QA Procedures which establish requirements for assuring inspection of welds for full penetration when required. The inspector had no further questions on the matter. Resolved.

53. I&E INSPECTION 79-07

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, and 17A were accepted by RCI quality control on November 15, 1978 although fitup gaps of 3/16 inch existed in the completed welds. Similar fitup gaps existed in three or more other similar beam supports.

RESOLUTION: LILCO letter to NRC (9/27/79) states that all prior fabrication welding on the RPV pedistal, the CRD support beams and the support beam clips was performed to the requirements of the AWS Code. Consistent with this the installation contractor employed welders qualified to ASME Section IX and selected welding practices for installation welding which fulfilled the requirements of AWS D1.1. This AWS Code provides criteria for acceptance which permits gaps between the two members of the base material of up to 3/16 inch. The Code also requires that, when the gap exceeds 1/16 inch, the size of the weld be increased by the approximate amount of gap at the root of the fillet. Nevertheless, LILCO reports that all such beam support welds are being reinspected to assure conformance to WAS D1.1. In the event that any nonconformances are identified by this inspection, corrective action will be accomplished as specified by the Nuclear Steam Supply System Supplier.

LILCO believes that no specific preventive action is required at this time as no further welding of this nature is presently contemplated. Should a requirement develop for such welding, appropriate acceptance criteria relative to fit-up will be published and applied.

I&E Inspection 80-15 notes examination of several records pertaining to the noncompliance and had no further questions. Resolved.

54. I&E INSPECTION 79-07

VIOLATION: Contrary to Criterion 16, 10 CFR 50 Appendix B, as of May 25, 1979 S&W specification SH1-159 and associated change EDCR-F19039 permit installation of raceway which do not conform to the minimum separation criteria, and permit subsequent installation of cables in the nonconforming raceways. Documentation of each nonconformance is provided by Specification SH1-159, and future disposition of the condition is controlled by the E&DCR control system. However, corrective action to prevent repetition has not been taken and additional nonconforming installations are being made.

RESOLUTION: LILCO letter to NRC (9/27/79) disagrees with the notice of viooation claiming that Shoreham Specification SH1-159 with the associated E&DCR F-19039, permit such installation in instances where the specified separation by distance criteria cannot be met. Also, Regulatory Guide 1.75, Physical Independence of Electric Systems, allows for alternative methods of compliance. The basic separation criteria, it argues, are not waived or changed but remain a plant design basis. Therefore, no change to the FSAR is required. Nevertheless, LILCO reports that each nonconforming condition is described and documented on an E&DCR, and specific analyses are being performed to determine the safety implications in each case. Where the results of these analyses so indicate, any necessary rework will be performed. Final acceptance in each instance will be predicated upon proper disposition of the applicable E&DCR in accordance with approved procedures.

NRC letter to LILCO (12/26/79) disagrees with LILCO's positions that 1) the basic separation criteria are unchanged and 2) there is no change to the FSAR, citing Regulatory Guide 1.75 as evidence. LILCO is ordered to complete the analysis and submit it to the NRC.

I&E Inspection 80-01 notes examination of S&W letters LIL-4678, 15170 and 15199 on portions of draft separation analysis. Although analyses of cable separation as well as instrument line and small bore piping separation is in progress, this item remains open.

LILCO letter to NRC (4/16/80) describes LILCO's method for safe shutdown analysis and argues that it substitutes for some separation criteria. LILCO offers to have I&E inspect the analysis at Shoreham. Results of the analysis will be final when all Class IE wiring is installed. LILCO offers to submit these results to the NRC but says that it will take four months of preparation time.

NRC letter to LILCO (7/1/80) points out that LILCO's analysis is okay if it is approved by the Office of NRR. It requests that the analysis be submitted as soon as possible. Furthermore, if the analysis is not approved, LILCO will be required to perform any rework. According to the NRC, this compliance still stands.

LILCO letter to NRC (8/14/80) states again its suggestion to schedule a meeting with the appropriate NRC staff reviewers. It reports that LILCO's preliminary report is now available for such a review meeting. LILCO says that it requested NRC project manager J. N. Wilson set up a meeting. He has had difficulty doing this. LILCO expresses that it shares the NRC's concern but restates its position that the analytical separation analysis approach is a conservative means of assuring that any deviations from the stated criteria do not constitute a condition adverse to the quality of our construction effort or overall plant safety.

NRC letter to LILCO (8/29/80) acknowledges LILCO's 8/14/80 letter. This matter remains open.

55. I&E INSPECTION 79-07

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on May 24, 1979, installed and inspected RCIC system instrument tubing was separated by less than one foot and was not provided with physical barriers at the connection to pipe spool 1 inch - SLP-9-151-2-1.

RESOLUTION: LILCO letter to NRC (9/27/79) reports that the RCIC system instrument tubing lines have been relocated to conform to the redundant lines separation criteria of Shoreham Specification SH1-343.

To prevent recurrence, LILCO assures that inspection responsibility for separation criteria of SH1-343 has been transferred from the installation contractor to Field Quality Control. Further, a system for color coding of redundant lines has been instituted to make deviations from the four foot separation requirement more readily apparent to the inspector during line walks.

I&E Inspection 79-12 notes examination of various EDCR's (6), confirming that corrective action taken by LILCO is acceptable. Resolved.

56. I&E INSPECTION 79-12

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on August 13, 1979 battery room ventilation control panels PNL-VC16, VC17, and VC18 were installed without approved engineering drawings.

RESOLUTION: LILCO letter to NRC (8/1/80) reports that failure to obtain approved mounting drawings for Panels PNL-VC16, VC17 and VC18 has been documented on a nonconformance report. Engineering evaluation of the nonconformance and completion of any necessary rework will be accomplished by January 1, 1980. Location drawings for the above panels are available at the construction site.

Final installation inspection of the panels had not been performed by Field Quality Control at the time of the NRC inspection. Since approved mounting drawings are required by FQC in order to perform their final inspection, this discrepancy would have been observed and corrected at that time.

To prevent recurrence, LILCO assures that responsible construction supervisory personnel have been advised of the NRC inspection finding, and specification requirements for panel mounting have been emphasized to them.

I&E Inspection 80-10 notes that the inspector interviewed cognizant structural engineering design personnel, reviewed engineering documentation addressing the adequacy of the modified installation and personnel training to prevent recurrence of installation without approved engineering drawings.

LILCO letter to NRC (8/1/80) follows up with a report that the engineering evaluation of the nonconformance is complete and concludes that the installation of panels PNL-VC16, VC17 and VC18 is acceptable, and the installation details for panel PNL-VC18 have been revised accordingly.

NRC letter to LILCO (8/21/80) confirms actions taken and has no further questions. Resolved.

57. I&E INSPECTION 79-16

VIOLATION: Contrary to Criterion 2, 10 CFR 50 Appendix B, activities affecting quality were not accomplished under suitably controlled conditions in that:

a) As of October 31, 1979 periodic inspections by personnel qualified in accordance with ANSI N45.2.6 were not performed to ensure the control of items in storage as required by A.. ~ N45.2.2.

b) No mechanism exists to update the Equip~ ~t Storage History Cards at the time when equipment changes location either in the warehouse or from the warehouse to a permanent location.

c) Periodic cleanness checks are not specified for many of the components stored in the plant (e.g., Standby Liquid Control Pumps and Motors, Core Spray Motors, and Residual Heat Removal Pumps and Motors). Additionally, as noted on inspections conducted between October 3 and October 11, 1979, many components were not maintained with adequate cleanness.

d) Caps, covers or plugs were noted to have been removed and not immediately replaced on several Category I components during inspections conducted between October 3 and October 26, 1979.

e) The space heaters in panels 1H21*PNL10 and 1H21*PHL26 were found to be de-energized on October 16, 1979.

RESOLUTION: LILCO letter to NRC (2/21/80) states that it takes partial exception to the findings noted above. Field Quality Control (FQC) Procedure 17.1 and Project Procedure 10 assign responsibility for implementation of program requirements for storage inspections to FQC and require FQC inspection personnel to be qualified in accordance with ANSI N45.2.6. Inspections performed under Construction Site Instruction (CSI) 4.6 are additional inspections performed by Construction personnel to assure that maintenance functions required by that reference are properly performed by the craft personnel.

A Component Stores Requisition (CSR) is required prior to any equipment being relocated from the warehouse to an inplant location. The Chief Mechanical Supervisor is required to sign the CSR before the equipment is moved, and he then directs the modification of the Storage History Card (SHC). The CSR constitutes a record of the relocation until the SHC is updated. Records of relocation of material within the warehouse are maintained primarily for the use of warehouse personnel, and are not considered a requirement of the ANSI Standard. With respect to valves 1B21*AOV-081A&B, LILCO can now find no evidence to support the finding that the SHC's were not properly updated to reflect the location change. However, it will continue to monitor this attribute closely to ensure compliance with requirements.

Cleanness checks are required by FQC Inspection Reports, which include "Cleanliness" as one attribute in addition to others, such as protective coverings, coatings and storage levels.

Per other items of noncompliance, LILCO reports that a Quality Control Instruction is being developed to more clearly define periodicity requirements for inspections of equipment in storage whatever its location. The specific items noted in paragraph 4.d. of the inspection report, including the Battery Room where extensive construction activities are now complete, have been cleaned. Missing caps, covers and plugs have been replaced, and the space heaters in Panels 1H21*PNL10 and 1H21*PNL26 have been reenergized. An inspection of the panels revealed no damage because of the lack of heat.

To prevent recurrence, LILCO assures that the Quality Control Instruction being developed to more clearly specify scheduling of periodic inspections of all items in storage, as discussed above, will be implemented promptly to minimize recurrence of the nonconforming conditions noted in the inspection report. The Battery Rooms will be locked.

I&E Inspection 79-20 notes that at various times during the past month, the inspector noted additional openings on safety-related equipment which were not immediately covered after use. Still open.

I&E Inspection 79-23 notes same as above (79-20).

I&E Inspection 80-06 notes review of corrective actions outlined by LILCO previously. The inspector notes increased inspections, but also notes that DCO's issued on these problems of dirt and covers have not been acted upon immediately. Still open.

I&E Inspection 80-09 notes that DCO's generated have decreased in number, and past DCO's are being cleared on a continuing basis. The current program should identify all discrepancies and satisfactorily correct them. Resolved.

58. I&E INSPECTION 79-24

VIOLATION: Contrary to Criterion 16, 10 CFR 50 Appendix B, on December 3, 1974, concrete placement No. RS-4-12, which is classified as a moderately massive section, had been exposed to a temperature of 38°F on the second day after placement. This nonconformance had not been identified by Field Quality Control and corrective action had not been taken to determine whether the exposure had adversely affected the concrete and to prevent repetition of such nonconformance.

RESOLUTION: Prior to the completion of the inpsection, LILCO reviewed the curing reports for all concrete placements made from November, 1973, through February, 1980 (2156). Eight additional panels were found where either no temperature was recorded or the temperature was below the minimum. FQC conducted Windsor probe tests of nonconforming placements. Procedures and results were reviewed by the NRC. Comprehensive strength of the placements in question ranges from 5200 psi to 6900 psi, well in excess of the design strength of 3000 psi. Resolved.

59. I&E INSPECTION 79-24

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on May 19, 1979 Weld Joint No. 1G33*WD9-3-1 FW-D was welded using ER-308 filler metal although Courter and Company Welding Procedure Specification NW-100-08011AA, Revision 0, required that this joint be welded using ER-309.

RESOLUTION: LILCO letter to NRC (6/2/80) reports that the weld in question was a dissimilar metal weld. Accordingly, all Category I piping isometric drawings were reviewed and all dissimilar metal welds identified. The document package for each weld so identified was examined to verify that ER-309 filler metal had been used. The results of the examination identified one additonal weld (No. 1M50*CW3-3-99 FW-C) where ER-309 filler metal had been used. The two welds thus identified were cut out of the system and replaced using the proper filler metal.

To prevent recurrence, LILCO assures that training sessions were conducted for quality control inspectors, nondestructive examination personnel and appropriate craft and supervisory personnel to review dissimilar metal welding techniques and the proper filler metal to be used.

I&E Inspection 80-10 notes examination of various documentation including memos, N&D Report 1370 component checklists, and the attendance report and records of a training session to re-instruct responsible personnel in requirements of weld material control and adherence to weld procedures. The inspector had no further questions concerning this item. Resolved.

60. I&E INSPECTION 80-10

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on June 18, 1980, the inspector observed that the process monitor sample point at elevation 75'-0", azimuth 210° and sampling lines extending from elevation 78'-7" to elevation 96'-0" did not incorporate principles for airborne sampling identified in ANSI N13.1-1969.

RESOLUTION: LILCO letter to NRC (9/22/80) reports that the guidance contained within ANSI N13.1 has been incorporated into the installation instructions for sampling systems. Systems already installed will be reinspected and reworked as necessary to insure compliance with the revised instructions. It is considered that location of the sample panel at a higher elevation than the process sample tap as noted in the Notice of Violation is acceptable since the flow velocity and the smooth flow path will ensure that particulate dropout will not occur.

To prevent recurrence, LILCO assures that incorporation of the requirements of ANSI N13.1 into the sampling system installation instructions will provide adequate guidance for future installations.

I&E Inspection 81-17 notes that the inspector reviewed several LILCO contractor drawings which outline the sample line route. The inspector noted that 1) the sample line right angle fittings are being replaced with the larger right angle radius bends and that the Radiation Monitor has been relocated at a higher elevation (from el. 95 to 112'); and 2) the sample line penetrated the drywell at elevation 104'-0, then extended vertically down into the drywell to elevation 75'-0. No data was available or presented to show that the increase in sample line length or additional height would not have a negative effect in obtaining a representative sample.

In addition, the sample line terminates in the drywell atmosphere. This is contrary to the FSAR commitment of Section 12.3.4.2.3, which states, in part: "...The unit is an off-line type which draws a representative sample from a ventilation duct in the drywell."

As of April 15, 1982, this matter is still open.

61. I&E INSPECTION 80-10

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on June 19, 1980 the inspector observed that the redundance safety-related conduit installation for system 1G33 did not meet the separation criteria of specification SH1-159; that this nonconformance had not been documented; and the conduit had not been tagged to show that it was nonconforming.

RESOLUTION: LILCO letter to NRC (9/22/80) reports that the conduit was not identified and tagged as nonconforming at the time of the NRC inspection as Field Quality Control had not yet performed a final inspection. The condition has not been documented on a nonconformance report and the conduit tagged to identify it as nonconforming.

To prevent recurrence, LILCO assures that clarification of instructions is being promulgated requiring that Engineering and Design Coordination Reports covering conditons where conduit separation criteria cannot be met must be issued prior to installation sign off by Construction. Field Quality Control will monitor for compliance to the revised instructions.

I&E Inspection 81-05 notes that N&D No. 3398 was written to document this nonconformance and to require documentation prior to sign-off by construction. The disposition requires that such exceptions be documented on an E&DCR and identifies E&DCR F29154 as documenting nonconforming conditions in question. The inspector confirmed this action by examinations of E&DCR F29154 and QC Instruction QC1-FS1-F12.1-08G, "Inspection of Raceway/Conduit Installations." Paragraph 5.3.6 of this QC1 establishes the same requirements. Resolved.

62. I&E INSPECTION 80-14

VIOLATION: Contrary to Criterion 3, 10 CFR 50 Appendix B, vent lines and vent valves not specified on the S&W drawing (FM-25A) were installed in three locations of the High Pressure Coolant Injection System without an authorizing E&DCR to modify the drawing.

RESOLUTION: LILCO conducted additional reviews and determined that, in all, 67 valves had been installed in a similar manner. An E&DCR was initiated to modify the appropriate drawings for all these valves. Additionally, procedures were initiated at the time of system hydrotest completion to ensure that all discrepancies of this sort were identified and corrected. Personnel involved were reinstructed in the requirments for E&DCR's.

NRC letter to LILCO (10/8/80) states that although federal regulations require a response from LILCO regarding this item, it notes that the noncompliance was corrected prior to the completion of the NRC inspection and, therefore, no response is required. Resolved.

63. I&E INSPECTION 80-15

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on December 7, 1979, snubbers No. 1G33*PSSP-228, 229, 230 and 231 had been released to construction for installation and were subsequently installed without evidence of having been stroked to assure that there was no binding.

RESOLUTION: LILCO letter to NRC (10/20/80) states that the requirement for stroking all safety-related snubbers was first established by Engineering and Design Coordination Report (E&DCR) F-12159 on March 38, 1978, and was incorporated into the governing specification by E&DCR F-1748D on May 2, 1978. In April, 1978, a Senior Field Quality Control Inspector was given responsibility for verifying implementation of the stroking requirement. He initially had all previously inspected snubbers reinspected and stroked the full length of their travel. Since then he has receipt inspected all safety-related snubbers received at the Shoreham site. He has confirmed that, as part of his inspection, he witnessed the stroking of each snubber and that failure to specifically reference E&DCR F-12159 on some Receipt Inspection Reports (RIR's) was an oversight. It should be noted that all RIR's do contain reference to the governing specification.

LILCO's corrective actions reported that the responsible Inspector has prepared a supplemental Inspection Report verifying implementation of the stroking requirement on all safety-related snubbers, and he has appended this report to each snubber RIR which failed to reference E&DCR F-12159.

To prevent recurrence, LILCO assures that the requirement that all documents which provide or modify acceptance criteria be referenced on Inspection Reports generated as documentary evidence of acceptance inspections has been reemphasized to all Field Quality Control personnel.

I&E Inspection 80-19 notes an interview with the inspector responsible for receiving inspection of the snubbers who stated that stroking of the snubbers was performed by a small crew who were all aware that this was a requirement, that he personally was responsible for this inspection in must cases, although occasionally it was performed by other inspectors. To assure that the requirements might not be overlooked in case of a change of personnel, a copy of E&DCR F-12159 was inserted with the purchase order. There were no further questions on the matter. Resolved.

64. I&E INSPECTION 81-01

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, PT.315.001B and C, "125V DC Power Distribution Preop. Test: for the B and C Systems, were being performed in January 1981, while the DC Bus Current and Voltage meters and the Battery Charger DC output current and voltage meters, which are required for conducting the PT, had not been recalibrated within one year.

RESOLUTION: LILCO letter to NRC (3/23/81) states that a startup Instruction had been issued on November 12, 1980 requiring the recalibration of permanently installed meters prior to their use for recording data during any preoperational testing if calibration has not been done within the preceding twelve months. Additionally, Performance Tests performed prior to the release of this instruction were reviewed and it was determined that either all calibrations were current, no instruments were involved which required calibration, or recalibration showed the instruments to be within the required tolerance. Performance of the permanently installed meters to be used during Preoperational Test 315.001B and C was compared with and found to be in agreement with the performance of currently calibrated meters prior to the start of the test, but no formal calibration record was prepared.

Regarding corrective action, LILCO reports that testing was stopped when the conditon was identified to minimize the potential for retesting, and all required instruments used during the test were found to be within required tolerances and no retesting was necessary.

To prevent recurrence, LILCO assures that the requirement for insuring proper calibration of meter and test equipment was re-emphasized to the test personnel involved and was included in the Training Program held for the Startup organization on March 9, 1981.

I&E Inspection 81-06 notes that the inspector reviewed the instrument calibration data packages for the recalibrated instruments associated with the 125V DC preoperational tests and noted some of the same discrepancies ad discussed above. This item remains open.

65. I&E INSPECTION 81-02

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, and Criteria 55-57, 10 CFR 50 Appendix A, the requirement that containment isolation valves be located as close to containment as practical was not prescribed by documented instructions, procedures, or drawings for the small bore piping containment isolation valves. As a result, the following outside containment isolation valves were not located as close to containment as practical:

Valves 1C11*01V-1028 A and B were installed 10 to 15 feet from the containment penetration; and

Valves 1P50*MOV-103 A and B were installed approximately 40 feet from the containment penentration.

RESOLUTION: LILCO letter to NRC (4/27/81) disagrees with the NRC notice of noncompliance. LILCO argues that it does not consider that the valve installation conditions described above are nonconforming. They have completed a field inspection and engineering evaluation of the location of the CIVs outside containment for the CRD to recirculation pump purge lines (1C11*01V-1028 A and B) and for the instrument air lines (1P50*MOV-103 A and B). The results of this investigation have led to the conclusion that the intent of General Design Criteria 55-57, requiring CIVs to be located "as close as practical" to containment, and Criterion V, requiring activities affecting quality to be prescribed by documented instructions, procedures or drawings, have been fulfilled at Shoreham. Approved fabrication and installation drawings were utilized and size quality control verification applied in each case. Therefore, it concludes no corrective or preventive action is considered necessary.

I&E Inspection 81-06 notes that the inspector along with LILCO walked each of the lines in question and noted actual locations versus potentially closer locations for valve installation. LILCO agreed to perform engineering design reviews to determine if, in fact, 1P50*MOV-103A could be located closer to containment. At the conclusion of these reviews the other aspects of this issue will be addressed. This item remains open.

Still open as of April 15, 1982.

66. I&E INSPECTION 81-13

VIOLATION: Contrary to Criterion 6, 10 CFR 50 Appendix B, on August 13, 1981, Startup Manuary No. 43, located in the control room and used by persons in the control room, was not adequately controlled in that it was not updated to include:

- -- Manual Revision No. 12, Dated February 18, 1981;
- -- Startup Instruction Nol 8, Revision 0, dated February 3, 1981;
- -- Startup Instruction No. 1, Revision 5, dated May 27, 1981;
- -- Startup Instruction No. 6, Revision 1, dated March 3, 1981;
- -- Startup Instruction No. 7, Revision 1, dated June 29, 1981.

In addition, the "Controlled" Manuals Distribution List posted in revision 12, dated February 18, 1981 incorrectly assigned Startup Manual Copy No. 43 to a different recipient.

RESOLUTION: LILCO letter to NRC (10/28/81) reports that Startup Manual No. 43 was updated and the Controlled Manuals Distribution List corrected during the course of the Inspection.

To prevent recurrence, LILCO assures that all controlled LILCO Startup Manuals on site are now maintained up to date by LILCO Startup personnel to ensure that all site working copies of the Manual are current. LILCO OQA has commenced quarterly surveillance of this activity to verify that the program is properly implemented.

I&E Inspection 82-01 noted that the most recent LILCO OQA surveillance was performed December 12, 1981. On December 14, 1981 the Joint Test Group approved Rev. 15 to the Startup Manual and made the revision effective December 21, 1981. On January 18, 1982 the inspector noted that the control room and OQA copies of the Startup Manual had not been updated to include Rev. 15 and that records showed that only seven of 44 controlled copies had been updated. The licensee's representative stated that additional measures would be taken to ensure prompt updating. This item remains open.

Still open as of April 15, 1982.

67. I&E INSPECTION 81-14

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, activities affecting quality were not accomplished in accordance with instructions in that on August 5-6, 1981:

a) The inspector identified the following leads in Panel 601 lifted with no documentation in the Log or Tags hung:

   -- Leads to pressure indicator E51-PI 011;
   -- Two leads from cable E11 BBC 640;
   -- Two leads from cable E11 BBC 641; and
   -- Lead CC 75 to Terminal Board HH.

b) The authorization block was not signed on tags #1528, 1529 and 1838.

c) Two fuses were found installed in the Remote Shutdown Panel even though RED tags #30224 and 30225 specified that the fuses be pulled.

d) BLUE startup jurisdictional tags were hung concurrently with YELLOW construction jurisdictional tags on a number of components of the Reactor Building Closed Loop Cooling Water System (P42).

e) None of the four active jumper/lifted lead permits had the expected duration recorded.

f) The jumpers for Permit #81-6-1 had been removed but the permit and jumper log had not been updated.

g) The Main Control Room set of controlled station procedures was not maintained current in that it contained Rev. 1 vice Rev. 2 of SP 12.035.01, "Control of Lifted Leads and Jumpers".

RESOLUTION: Prior to completion of the inspection, LILCO corrected each of the above items except the missing expected durations of jumpers (item E).

LILCO letter to NRC (10/13/81) reports that corrective action for each item cited except item E was completed prior to the conclusion of the inspection. For item E, the expected duration has now been added to the four active jumper/lifted lead permits.

To prevent recurrence, LILCO assures that with respect to items 1, 2 and 3, the requirements of 10 CFR 50, Appendix B, and the LILCO Startup Manual pertinent to the Lifted Wire/Temporary Jumper Program were re-emphasized at the Startup General Staff Meeting on September 8, 1981, and appropriate written guidance for implementing the Program has been provided to all Test Engineers and Technicians involved. For item 4, LILCO reviews of simultaneous jurisdictional tagging have shown this to be an isolated instance. A memorandum has been issued to site Construction and Startup personnel re-emphasizing the requirements of the Shoreham Startup Manual regarding Yellow and Blue jurisdictional tagging. With reference to item five, the Jumper/Lifted Lead Permit form has been modified to include an "expected duration" block to be completed during preparation of the permit. Relative to item 6, the importance of returning the tags on the lifted leads to Operations personnel in accordance wtih SP 12.035.01 has been re-emphasized to Plant Staff personnel, and for item seven, further instruction has been provided to the individuals responsible for procedure distribution. Finally, the Operational Quality Assurance Section has commenced surveillance to assure proper program implemention in this area.

I&E Inspection 81-20 notes that the inspector reviewed documents pertaining to the corrective actions outlined in LILCO's 10/13/81 letter. He also reviewed the plant staff Jumper Log and toured the plant to observe tagging and the condition of panels and motor control center cubicles regarding jumpers and lifted leads. No discrepancies were identified in the areas of danger tags or jurisdictional tags. In the area of jumpers and lifted leads, the inspector did note several additional discrepancies, all of which LILCO corrected. This matter remains open pending further verification that the requirements for jumpers and lifted leads are being implemented.

Still open as of April 15, 1982.

68. I&E INSPECTION 81-16

VIOLATION: Contrary to Criterion 5, 10 CFR 50 Appendix B, on September 11, 1981, the inspector observed that Project Procedure P304 does not require report of a possible reportable deficiency.

RESOLUTION: LILCO letter to NRC (10/26/81) reports that Project Office personnel involved in reviewing and reporting deficiencies have been instructed to document and report Potentially Reportable Deficiencies as outlined in the NRC's "Guidance - 10 CFR 50.55(3), Construction Deficiency Reporting", dated April 1, 1980, and Project Procedure P304 will be modified to incorporate reporting Potentially Reportable Deficiencies as required by the NRC Guidance.

LILCO believes that no additional preventive action is considered necessary.

I&E Inspection 82-01 notes that the inspector examined Project Procedure P-304 which had been revised to include provisions for reporting of potentially significant procedures. The inspector also reviewed records and reports of recent potentially reportable deficiencies and confirmed that such items were reviewed and reported in accordance with Procedure P-304. There were no further questions on this matter. Resolved.

69. I&E INSPECTION 81-22

VIOLATION: Contrary to Criteria 3 and 5, 10 CFR 50 Appendix B,

a) On December 14, 1981 there was no Yellow-Lined Master in the Startup Resource Center Yellow-Lined Master File for drawings ESK-11R4204 or ESK-6T2301.

b) As of December 14, 1981, although the affected Pre-operational Tests had been completed, there was no stamping or other documentation on the below listed drawings in the Startup Resource Center Yellow-Lined Master File to indicate that the latest revisions had been reviewed by the Test Engineer: ESK-5R2303, ESK-5R2304, ESK-6P2108, and ESK-6P2111.

c) As of December 14, 1981 numerous superseded drawings were not retained in the Startup Resources Center Yellow-Lined Master File, including:

ESK-11R4204, original and Rev. 1
ESK-6T2301, Rev. 2
ESK-6G1133, Rev. 5
ESK-6P2111, Rev. 3

d) As of December 14, 1981, numerous superseded drawings retained in the Startup Resource Center Yellow-Lined Master File were not makred "VOID", including:

ESK-11R4201, Rev. 3
ESK-11R4202, Rev. 2
ESK-5R301, Rev. 11, 11A, 12, and 13
ESK-6R4308, Rev. 3
ESK-5R2304, Rev. 6, 6A, 6B, and 6C
ESK-6G1104, Rev. 3
ESK-6G1114, Rev. 2

RESOLUTION: Still open as of April 15, 1982.

70. I&E INSPECTION 82-02

VIOLATION: Contrary to Criterion 17, 10 CFR 50 Appendix B, LILCO failed to maintain records to furnish evidence of the fit pump tensioner hydraulic pressure of the jet pump hold down beam preload force used in the fabrication of the jet pumps during the summer of 1981, as required by the EQA Manual.

RESOLUTION: Still open as of April 15, 1982.

71. I&E INSPECTION 82-02

DEVIATION: LILCO has failed to uphold its commitments to the NRC in that the Shoreham FSAR, paragraph 4.4.6 states that the loose parts monitoring system meets the requirements of Regulatory Guide 1.133 and yet contrary to the requirements of that document,

a) as of January 13, 1982, instrument cables for different channels were not physically separated inside the drywell (which is accessible during fullpower operation) in that they were run in the same conduits and they utilized the same electrical penetration;

b) as of January 13, 1982, there was no alarm or annunciator from the loose parts monitoring panel to audibly or visually alert control room personnel that the alert level had been reached.

(This is not considered a noncompliance item, but rather as a deviation from FSAR and NRC commitments).

RESOLUTION: Still open as of April 15, 1982.

ATTACHMENT 6

STAFF REPORT ON QUALITY ASSURANCE

BY THE PRESIDENT'S COMMISSION

ON THE ACCIDENT AT THREE MILE ISLAND

OCTOBER 1979

ATTACHMENT 6

Staff Reports To The President's Commission On

THE ACCIDENT AT THREE MILE ISLAND

Reports Of The Technical Assessment Task Force. Vol. IV

REPORT OF THE
TECHNICAL ASSESSMENT TA^{C}Y FORCE

ON

QUALITY ASSURANCE

BY

William M. Bland, Jr.
Dwight Reilly

October, 1979
Washington, D. C.

TABLE OF CONTENTS

IV. FINDINGS AND CONCLUSIONS

A. SPECIFIC FINDINGS

For the convenience of the readers, the specific findings are listed here in the same order that they were identified in the text. The most important parts of the specific findings have been combined into the major findings that are listed in the conclusions section of the report.

The specific findings, by report section, are as follows:

Requirements

- o Quality assurance requirements as stated in 10-CFR-50 Appendix B, appear adequate for those systems to which they apply.

- o Quality assurance requirements apply only to a narrow portion of the plant defined as safety-related or safety grade. Many items vital to the safe and reliable operation of the plant are not covered by the quality assurance program because of this definition.

- o There is no requirement for independent, on-site quality, or safety assessment operations. Surveillance testing by the utility is audited infrequently. Regulations allow review to be done by in-line supervision and other personnel directly responsible for operations.

- o Reliability/safety analysis requirements are applied to specific safety-related hardware as specified in Appendix A of 10-CFR-50 utilizing a questionable "single failure" criterion.

- o Safety and reliability requirements and analyses are not required to be applied to many plant systems which may be "vital" to the safe operation of the plant, but are not labeled "safety-related."

- o Lack of requirements by NRC in the safety and reliability disciplines has resulted in little motivation to form a strong safety and reliability engineering capability in NRC and the utility industry.

- o Present NRC design, safety, and reliability requirements do not generally address human factors and the man-machine interface.

NRC Organization and Responsibilities

- o There is no assignment within the NRC organization for overview of critical functions such as: problem reporting, failure analysis, and corrective action; systems engineering; and the role of the operator and human factors in plant safety.

- The fragmenting of quality assurance responsibilities among the various NRC organizations weakens the ability of this discipline to ensure an adequate utility quality program.

- The NRR Division (DOR) responsible for overviewing the operating reactor is not part of the licensing design review, construction, or startup monitoring process.

- No NRC organization is identified as being responsible for auditing the project management, engineering, and inspection functions of the NRC.

- NRC project managers and quality assurance personnel in the NRC Division of Project Management and Operating Reactors are primarily concerned with initial licensing and changes thereto within the scope of the FSAR and Standard Review Plan. Little overall assessment of utility management, engineering, or operations is evident.

- The NRC project manager does little engineering analysis and is not a significant factor in the review of nonconformances, procedures, or system engineering aspects of the plant.

- Project management experience gained during design construction and startup of the plant is lost upon transfer of responsibility for the plant to DOR. There appears to be little effort by the project manager in DPM to transfer licensing and startup experience to other NRC groups.

- There is no NRR review of proposed operating procedures as part of operating license approval.

- The Division of System Safety overview of the nuclear power plant is primarily concerned with the design of safety-related components and subsystems within the framework of the Standard Review Plan.

- The DSS does not include nor does the Standard Review Plan require significant consideration of non-safety-related systems, systems interactions, operating procedures, or human factors in the evaluation of the nuclear plant.

- The DSS has not adequately recognized potential system and system-operator problems even when these problems were brought to their attention; possibly because of the emphasis applied to component and subsystem design aspects and to the design base accidents by the NRC.

- The DSS makes little use of plant experience data in developing requirements for and in the conduct of their overview process.

- The NRC Office of Inspection and Enforcement and its regional office conduct a detailed, documented inspection program for those utility systems and activities covered by applicable regulations, regulatory guides, utility FSAR, operating license, and technical specifications.

- o Region I on-site inspections appear to miss signals and symptoms that indicate potential plant operating problems and weak utility management.

- o In Region I, there is little physical inspection or direct observations of operations such as surveillance testing of the operating reactors during NRC plant visits.

- o Region I inspections did not detect the emergency feedwater valve procedure change leading to technical specification violation in about 15 visits to TMI-2 from August 1978 to March 1979.

- o The role of quality assurance does not appear to be an important factor in the I&E plan. No I&E audit was made of the TMI-2 quality assurance plan to see that that plan was implemented to support the operating phase from the beginning. An I&E audit about 18 months after operating license issuance found many deficiencies in the implementation of the quality assurance plan. In their investigation of the TMI accident, I&E did not interview any Met Ed quality assurance personnel in the 200 interviews held.

- o Sufficient I&E staff may not be available to conduct an adequate overall plant surveillance (inspection) activity.

- o There is little I&E assessment of the utility's management capabilities.

- o Although one inspector receives all reports concerning TMI-2, he has no responsibility for the execution or the quality of execution of all TMI-2 sections.

<u>Utility (Met Ed) Organization and Responsibilities</u>

- o The Met Ed organizational structure, quality assurance plan, and independent review groups meet basic NRC requirements.

- o As implemented, the TMI independent assessment program involving quality assurance and the review committees, PROC, GRC, and GORB, looked only at NRC required safety-related functions and therefore could not assure safe operation of the overall plant.

- o Lack of quality assurance or other TMI independent assessment of non-safety-related hardware and procedures was a factor in the accident.

- o Because of the limited purview of the review mechanisms, it is possible that Met Ed management was not fully cognizant of plant conditions and operations.

- Although the TMI internal audit program meets NRC requirements and is well done, Met Ed management did not assure that corrective action identified by the audits was initiated and completed in a timely fashion.

- Significant misunderstanding exists among NRC and TMI-2 personnel regarding the meaning and application of terms such as "safety-related," and "safety grade," and similar terms.

- Misunderstanding exists among NRC and TMI management and project personnel as to what specific hardware is considered safety-related at TMI-2 and what specific document defines that hardware.

- The lack of clear designation of safety-related equipment and, specifically, what that means contributed to inadequate hardware and procedure review and failure analysis and corrective action that are necessary to assure safe operation of the plant.

Procedures

- There is essentially no NRR review of detailed utility procedures. Reviews are limited to assuring that a proper list of procedures is available and a utility procedure review system is in place.

- I&E review of procedures is limited by intent to about 5 percent of operating and emergency procedures, and changes to procedures identified by the utility as impacting the technical specification.

- The PORC is the primary procedure review organization. Current PORC membership and review practices appear to preclude adequate independent review of procedures associated with safety-related systems.

- Lack of TMI quality assurance overview of the preparation and conduct of surveillance procedures can preclude detection of omissions, mistakes, and unsafe practices by the utility.

- A small utility quality control staff precludes adequate verification (inspection) of maintenance and repair of safety-related systems and components.

- There is no independent review or verification of maintenance and repair procedures involving systems not identified as safety-related, but which may be important to safe and reliable plant operations.

Nonconformance Reporting Systems

- There is no systematic problem reporting, rigorous failure analysis, corrective actic , problem trend evaluation, and information distributing system applicable to all plant hardware systems, procedures, and operations that are important to plant safety and reliability.

- NRC requirements contained in 10-CFR-21 limit reporting of events by the licensee to essentially those functions and hardware considered safety-related.

- The format and content of license event reports as required by NRC do not provide appropriate identification and classification of the problems and their causes; or provide sufficient information for effective utilization by other utilities.

- No NRC organization has had the assigned responsibility to systematically assure a thorough review of each LER, the failure analysis contained therein, the corrective action taken by the utility, and the possible application of the information to other plants.

- There is little evidence of use by NRC or the industry of operating experience and failure history contained in LERs to upgrade requirements, designs, procedures, and training.

Configuration Control

- The limited NRC overview of utility changes to plant configuration does not assure NRC a current understanding of plant systems and operations.

- NRC personnel involved in the original plant design review during the licensing process are not required to review plant changes.

- I&E personnel responsible for plant overview and acceptance of LER corrective actions are not directly involved in the configuration change process.

- The TMI system for reviewing and controlling changes to safety-related systems appears adequate. There is a lack of rigorous control and independent review of hardware configuration and changes thereto for other systems important to plant safety.

- At the time of the accident, TMI did not have a rigorous drawing control system in place that assured plant operators had an adequate understanding of the as-built configuration of all the facility.

- The I&E inspection program calls for examination of the utility configuration control system once every 3 years. It did not assure an adequate document (drawings and procedures) control program at TMI.

Assurance Functions of Nuclear Power as Related to Other Programs

- The overall quality assurance, safety, and reliability programs and practices utilized by NRC and GPU/Met Ed are not commensurate with the requirements, procedures, and practices of other programs where safety and reliability are critical concerns.

- Management, engineering, quality assurance, safety, and reliability practices and philosophies are available to minimize the probability of failures in the nuclear industry.

B. CONCLUSIONS

A review of the independent assessment program for nuclear power plants as defined by NRC quality assurance regulations and requirements has been accomplished by examining the major elements of the NRC and one of its five regional offices and one utility company (Met Ed). This somewhat limited review has resulted in two general conclusions and several major findings. The major findings relate to the specific findings listed in the previous section of this report. The findings are supported by the results of the analysis by the Department of Energy (DOE) (reference 122).

It is concluded that the overview and independent assessment performed by NRC were limited only to those items which were identified as safety-related, including intensive analyses of recovery from postulated accidents which resulted in a narrow overview of the utility. This narrow view was further confined by the application of the forerunner of the Standard Review Plan which programmed the review effort by NRC to carefully defined areas. Further, this narrow and confined review was bothered by a focusing problem brought about by doubts about the interpretation and application of the term "safety-related" to equipment; this further affected related procedures, inspection, maintenance, and problem resolution. Combining this narrow view with a weak NRC-to-utility management interrelationship, left voids that prevented the NRC from knowing the "health" of the utility. More important, the NRC did not have an independent assessment activity to "tell them that they didn't know."

It is further concluded that the management utility joined the NRC's narrow and confined view on the safety items and virtually ignored other vital parts of plant operation. This viewpoint is shared in an analysis by DOE (reference 123). These other parts were those whose performance not only supported the safety-related items, but were those that were also vital to assuring that the plant would reliably perform. This illustrated that the utility management had not exhibited the desire or capacity to go beyond the NRC requirements to provide a well-designed, maintained, and staffed plant capable of reliable performance that would not jeopardize the health and safety of the public and its own workers. Like the NRC, the utility management had no independent assessment system to tell them that their plant was "sick."

The major findings are as follows:

- The NRC organization, procedures, and practices, as now constituted, do not provide for the combined management, engineering, and assurance review of utility performance necessary to minimize the probability of equipment and operator failures necessary to ensure the safe operation of the nuclear power plant.

- A lack of an independent on-site quality assurance or safety assessment of plant operations and of equipment not considered safety-related contributed significantly to the accident at TMI.

- There was lack of detailed safety and failure modes analysis on all plant systems necessary to ensure the reliability and safety of the facility.

- Systems engineering, interactions between systems, and the interaction between the equipment and its operators have not generally been considered in the NRC overview process.

- A comprehensive nonconformance, problem reporting, failure analysis, corrective action system applicable to all systems and operations that affect plant safety and reliability does not exist. The current LER system also does not assure adequate dissemination and utilization of useful failure data through the industry.

- Current utility and NRC practices do not assure proper preparation, review, and execution of operating and maintenance procedures.

- NRC has a very limited view of changes made to plant configuration. Utility control of safety-related equipment changes appear adequate; control of non-safety-related equipment configuration is inadequate.

- Full use is not being made of management, engineering, safety, reliability, and quality assurance practices which are in use in other industries where safety and reliability are critical concerns.

ATTACHMENT 7

REPORT FOR THE COMBINED UTILITY ASSESSMENT OF THE ADEQUACY OF THE LILCO QA PROGRAM FOR NUCLEAR APPLICATION

FEBRUARY 27, 1981

ATTACHMENT 7 APR 19 1982

REPORT FOR THE COMBINED UTILITY ASSESSMENT OF THE ADEQUACY OF THE LILCO QA PROGRAM (For Nuclear Application)

Conducted: February 23-27, 1981

Summary:

This is the fourth of a planned series of reciprocal combined utility assessments which are intended to provide an independent evaluation of Quality Program adequacy and, to the extent possible, implementation. As requested by LILCO, the scope of this assessment was accomplished in the following areas:

Organization and Staffing
Quality Program
Procurement
Audit Program
Corrective/Preventive Action
Trends
Training
QA Records
Document Control
Reportables 50.55(e)/Part 21
Instructions, Procedures and Drawing

In attempting to fulfill this charter, the assessment team reviewed applicable LILCO Quality Program commitments in QA manuals, FSARs, policies and procedures of departments participating in the Quality Program for nuclear activities. Each team member developed checklists, as the basis for conducting the assessment.

The assessment was conducted by interviews and discussions with appropriate management and staff personnel.

The assessment was performed at the Shoreham Site and in the Hicksville offices of LILCO and involved the following organizations, as related to the QA Program:

Shoreham Project
Purchasing
Quality Systems Division
Field Quality Assurance Division
Operational Quality Assurance Section
Meter and Test Department - Protection Division
Nuclear Operations Support D.
Nuclear Engineering
Nuclear Fuel Division

A list of all personnel contacted during the assessment is attached to this report.

Assessment Conclusions:

Attached to this report are "ASSESSMENT SUMMARIES", which contain the teams assessments of an area or subject, the items observed during the assessment process, and the actions the team believes should be considered by LILCO.

The content of the 16 assessment summaries are self-explanatory. Any areas not covered by an in-depth assessment may have received limited assessment or were considered acceptable.

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NUSCO
Assessment Team Leader

T G Bassett 2/27/81
T. Bassett — Date
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T. Crouse — Date
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Dean Schaefer	Supervisor of Corrosion and Instrument Division
William Stoll	Manager of Purchasing Department
V. L. Elefante	Manager of Material and Equipment Purchasing Department
R. M. Kascsak	Manager of Nuclear Systems Engineering
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F. X. Schoner	QA Coordinator, Quality Systems
A. Maksimchak	Protection Division

Program Element: QA Staffing

I. Areas Examined

Size of QA Department Staff

II. Assessment

The QA Department staff is authorized 17, present size of staff is 11. The OQA Section staff is authorized 8, with a present staff of 3. There is a need to assure that all quality functions are sufficiently staffed.

III. Conditions

FQA - Authorized 8
Present 6

QSD - Authorized 8
Present 4

OQA - Authorized 8
Present 3 - LILCO
10 - S&W

IV. Areas (Actions) to be Considered

Continue the effort to add required QA personnel to ensure all aspects of the quality program requirements are covered.

The QA Department is actively pursuing recruitment of additional quality assurance people, and are experiencing difficulty in the selection of candidates which is somewhat due to inconsistencies in present salary structures in relation to industry scales, the local cost of living and housing costs and a highly competitive labor market for qualified QA people.

Consideration should be given to staffing the OQA function with LILCO employees rather than S&W. Permanent LILCO employees continuing thru operations would eliminate future retraining.

Program Element - QA Reporting Chain

I. Areas Examined

Reporting Chain for Operations QA Engineer

II. Assessment

The OQAE reporting functionally to the Plant Manager is not in concert with current NRC thinking, as noted in NUREG-0731 Published 9/80, which recommends that the **onsite QA function should report to the offsite QA Manager.**

III. Conditions

The present organization shows that the OQAE "Functionally" reports to the Plant Manager and has a "Review and Audit" reporting chain to the Manager, QA.

NUREG-0731 "Draft", Published September, 1980, states the following. "The reporting of the functional areas of radiation protection, quality assurance and training should assure independence from operating pressures. ....overall management and technical direction in these areas may well be concentrated at the home office." Also, a Representative Plant Organization depicts the onsite QA function reporting to the offsite QA Manager.

IV. Areas (Actions) to be Considered

LILCO should reassess the reporting chain for the OQAE.

Program Element - Quality Program

I. Areas Examined

FSAR 17.1-9, Revision 17, September 1979

Engineering QA Manual

QA Manual, Draft

- Purchasing Department

- Fuels Purchasing Department

Nuclear Operations Support Division

Operations QA

Project Management

- Nuclear Fuel

Quality Assurance Department

Plant Management

II. Assessment

Discussions with personnel representing the organizations involved in the Shoreham activities, indicates inconsistent understanding of their organization's role in the implementation of the Quality Program.

III. Conditions

Interviews with representatives of the above organizations, revealed that some were uncertain as to whether or not they are or will be involved in the implementation of the Quality Program. The uncertainty could exist due to one, or a combination, of the following factors:

1) the effects of the recent major reorganization,

2) untimely indoctrination and training by the QA Department and/or

3) inadequate <u>intra</u>-department training in the details of each department's responsibilities within the program.

IV. <u>Areas (Actions) to be Considered</u>

Senior management should consider investigating, to determine the cause or causes of such middle-management uncertainty. Having identified this (or these), positive corrective action should be taken.

Program Element - Quality Program

I. Areas Examined

Engineering QA Manual

Operations QA Manual

FSAR 17.2-1 Revision 17, September 1979

QA Manual, Draft

II. Assessment

LILCO should act to establish one QA Program. The QA Manual presently in Draft form, subsequent to resolution of comments is to be used as a basis for a FSAR Amendment and will then allow procedure issues in a timely manner prior to licensing hearings.

III. Conditions

The "Draft" QA Manual has been written to replace the present EQA and OQA manuals. The review cycle for the Draft Manual is not complete. The EQA and OQA manuals infer that two QA Programs exist at LILCO. In reality, LILCO is working toward an operational status with one QA Program.

IV. Areas (Actions) to be Considered

1) Utilize the QA Manual as the QA portion of the FSAR.

2) Submit the QA Manual as a Topical Report to the NRC for approval. An approved Topical may then be utilized as the QA portion of the FSAR.

Program Element - QA Program

I. <u>Areas Examined</u>

Engineering QA Manual

Operational QA Manual

Final Safety Analysis Report

II. <u>Assessment</u>

There are some inequities in the quality sections of the FSAR and the Engineering and Operational QA Manuals, specific to the requirements of 10CFR50 Appendix B.

III. <u>Conditions</u>

1. The FSAR (17.1-26) states that QA records may be retained in separate "or" remote locations. Protection from tornadoes, rodents and insects was not included in the coverage of the protection of QA records.

2. The NSSS is apparently not required to comply with ANSI N45.2.9 (Reference Engineering QA Manual Section 17.3.8c).

3. Section 17 of the EQAM is not compatible with the FSAR (17.1-26) in that, LILCO's QA records responsibilities are not delineated and the records retention program is not described.

4. Section 6 of the EQAM does not include all responsible LILCO departments, e.g., Purchasing, Meter and Test.

5. The FSAR 17.1-11, 12 & 13, does not provide for reviews to assure that design bases were included in procurement documents.

6. Section 4 of the EQAM does not require the review of procurement documents prior to bid or contract placement (reference ANSI N45.2.13).

7. Sections 4.3.1 and 4.3.7 of the OQAM are not clear and provisions for,
   a) review for the inclusion of regulatory requirements and design bases and,
   b) assurance of the right of access,

   are not included.

8. Section 5 of the EQAM and OQAM are not clear as to whether the requirements apply to all LILCO organizations.

9. The responsibility for the review of site and offsite procedures (page 17.2-12) was not carried forward from the FSAR to the two QA Manuals.

The above items applicable to the EQAM and OQAM, were discussed with the QA Department personnel and areas which were noted as not contained in the proposed draft QA Manual were identified for future consideration.

Encl

IV. Actions to be Considered

Future revisions to the FSAR and QA Manual should address the conditions in Section III.

Program Element - Control of Purchased Materials, Components and Services

I. Areas Examined

A. Meter and Test Department

B. Purchasing Department

C. Nuclear Engineering Department

D. QA Department

II. Assessment

QA Department

Provisions for the implementation of the requirements of Regulatory Guide 1.144 relative to the assessment of the performance of suppliers were not in effect. This requirement is relatively new and work is underway to cover it, when LILCO assumes this responsibility. The reassessment of suppliers is currently being performed by the QA agent.

III. Conditions Observed

A. Meter and Test Department

The department's personnel exhibited satisfactory control of purchased calibration services.

B. Purchasing Department

Supplier evaluation, assessment and qualification are not functionally assigned to the department. The completed purchase requisition and release form are the authorizations to place contracts.

C. Nuclear Engineering Department

Not Assessed

D. QA Department

The primary responsiblity for this function is assigned to LILCO's QA agent. The evaluation of each supplier annually as required by Reg. Guide 1.144 for the chosen triennial audit option is being performed. Suppliers surveyed (qualified) by the QA Department are added to the approved suppliers list and the annual evaluation is performed by the Agent. The QA Department's draft procedure QAP 7.2 adequately addresses this area.

IV. Actions to be Considered

QA Department

The division should complete the development and initial implementation of the procedure (QAP 7.2) for the assessment of suppliers and the annual evaluation as required by Reg. Guide 1.144.

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Program Element: Audit Program

I. <u>Areas Examined</u>

The 1980 audit activities of Quality Systems, Field QA, and Operations QA.

II. <u>Assessment</u>

The audit program being implemented by the three audit organizations meets LILCO's regulatory commitments.

III. <u>Conditions</u>

1. The FQA audit program is exceptionally comprehensive, effective and efficient.

2. The Quality Systems and Operations QA audit schedules provide for timely audit of new activities; however, Nuclear Engineering and Nuclear Operations Support Division could not support an audit due to a lack of internal procedures.

IV. <u>Actions to be Considered</u>

1. Expedite issuance of Nuclear Engineering and Nuclear Operations Support Division procedures.

2. The Nuclear Operations Corporate Policy 16, does not provide for sufficient overlap of audit responsibilities during early years of operation. These audit responsibilities should be reassessed prior to plant operation.

Program Element - Corrective/Preventive Action

I. <u>Areas Examined</u>

The operational phase corrective and preventive action programs that are used in conjunction with the nonconformance control system, audit program, and trending activities.

II. <u>Assessment</u>

The corrective action program is being adequately implemented in accordance with existing procedures; however, measures to more quickly provide preventive action should be considered.

III. <u>Conditions Found</u>

1. The procedure for nonconformance reports do not include provisions for specific corrective action to prevent recurrence. These nonconformances are reviewed by OQA on a periodic basis for trend identification and the need for corrective/preventive action.
2. Operations QA issued seven Corrective Action Requests in 1980; four of which addressed unresponsiveness on the part of Project, Startup, or Plant Staff.

IV. <u>Areas (Actions) to be Considered</u>

1. Provide procedural provision for recurrence control as needed on each LILCO nonconformance report; or
2. Increase the frequency of performing trend analysis and subsequent issuance of a CAR.

Program Element - Trends

I. Areas Examined

Identification of adverse quality trends.

II. Assessment

Trending activities are being adequately performed in accordance with applicable procedures and regulatory commitments.

III. Conditions Found

1. Quality Systems and Field QA are reviewing various documents on a quarterly basis to identify repetitive adverse occurrence.

2. Operations QA has performed its first required annual trend report.

3. Field QA has inputed its past audit reports to a computer and can sort or trend its audit results by activity, auditor, organizations, etc.

IV. Areas (Actions) to be Considered

None

Program Element - Training, Headquarters

I. Areas Examined

Nuclear Engineering

Nuclear Operations Support Division

Purchasing

Nuclear Fuels Division

II. Assessment

LILCO should formulate an overall plan to identify required formal training for those personnel assigned to the headquarters organizations, who are dedicated to support Shoreham. The plan should include specific actions to be taken by each organization and a schedule for accomplishing the actions.

III. Conditions

Presently, each of the headquarters organizations are performing or contributing to the overall Shoreham effort and have not as yet, provided procedures to reflect their plans or specific actions to provide indoctrination and technical training to personnel who are or will be involved in safety related activities.

IV. Areas (Actions) to be Considered

1) Identify the overall requirements for headquarters training.
2) Provide procedures to formally prescribe the application and scheduling of training.

Program Element - Training, Headquarters (Reporting of Status and Adequacy)

I. Areas Examined

Engineering QA Procedures

QA Department

II. Assessment

The periodic review and reporting requirement by the QA Manager, relative to the EQA Training Program needs to be analyzed and, possibly, revised.

III. Conditions

EQAP 2.1, paragraph 4.1(e) requires that the QA Manager review and report periodically to the Senior Vice President, Engineering on the status and adequacy of the EQA Training Program. This is not consistently being done.

IV. Areas (Actions) to be Considered

Comply with the requirement.

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Program Element - Training, Shoreham Site

I. Areas Examined

Operating QA Section's Procedures

Plant Staff Procedures

Training Procedures for Both

II. Assessment

The site procedures reviewed, appear to be comprehensive with but one relatively minor exception, i.e., it appears that not all procedures are covered within the training procedures.

III. Conditions

It is clear that a great number of operating procedures are available at the site which govern the activities of the Plant Staff. Further, training procedures are prepared to guide plant personnel in how, how frequently, what content, how documented, who, etc., should receive training to comply with the operating procedures. A few, primarily administrative procedures, e.g., the procurement procedure, may have inadvertently been overlooked as far as their coverage in the training procedures.

IV. Areas (Actions) to be Considered

The operating procedures should be reviewed to insure that each is covered within the training program.

Program Element - QA Records

I. <u>Areas Examined</u>

A. Meter and Test Department

B. Purchasing Department

C. Nuclear Engineering Department

D. QA Department

II. <u>Assessment</u>

A. Meter and Test Department

The control of QA records was satisfactory except for two areas which are identified in III A below.

B. Purchasing Department

The control of QA records was being handled in accordance with existing pre nuclear procedures which did not comply with the requirements of ANSI N45.2.9, the Nuclear Operations Corporate Policies and the QA Manual.

C. QA Department

The QA Department's philosophy (as indicated in Quality Assurance Procedure 17.1) of treating QA record files, as working (in process) records, is not in compliance with ANSI N45.2.9. The specifics are noted in III D below.

III. Conditions Observed

A. Meter and Test Department

The department's control of QA records was satisfactory except for the following:

1) the validation of record changes by the originating organization, and

2) the periodic audit of records to ensure that records are in their assigned locations and are in good condition. The department's records index was in compliance with the requirements of ANSI N45.2.9 and a full time security system was in effect. A duplicate copy of the department's records had been distributed to the Permanent Plant File and the filing system in use was described in procedures and was implemented.

B. Purchasing Department

The control of QA records was not being carried out in accordance with the requirements of ANSI N45.2.9 in that,

1) an index of QA records generated and processed by the department did not exist,

2) the location within the storage area for the QA records was not specified,

3) full time security was not provided, and,

4) periodic audits were not performed.

1 Start

A central file was being used for extended retention of purchasing records and a filing system for all purchasing records was in effect. Copies of purchase orders were being sent to Project Management at the site for use and distribution. Duplicate copies of many records were being sent to the permanent plant file.

C. Nuclear Engineering Department

Not Checked

D. QA Department

QAP 17.1 does not cover the following areas of the control of QA records:

1) storage of records,
2) adding supplemental records and correcting records,
3) control of access to the records,
4) accountability of records,
5) filing of supplemental records and disposition of obsolete records,
6) protection of records, and,
7) periodic audit of records to verify records are filed correctly and are in good condition.

IV. <u>Actions to be Considered</u>

A. Meter and Test Department

The department should develop and implement methods of

1) obtaining validation of record changes by the originating organization and,
2) the performance of periodic audits of the records keeping function and files.

B. Purchasing Department

A coordinated effort should be initiated in conjunction with all LILCO nuclear departments under the guidance of an assigned coordinator to develop and implement procedures for the control of QA records.

C. QA Department

QAP 17.1 should be updated to adequately cover the control of QA records as required by ANSI N45.2.9.

Program Element - Document Control

I. Areas Examined

A. Meter and Test Department

B. Purchasing Department

C. Nuclear Engineering Department (Hicksville)

D. Quality Assurance Department (QSD)

II. Assessment

The existing procedures for some of the activities assessed, appear to require revision to address the control for timely receipt and acknowledgement of distributed documents.

III. Conditions Found

A. The Meter and Test Department's control of procedures appears adequate, however, follow-up to ensure that distributed documents have been received, should be addressed.

B. Nuclear Engineering (Hicksville)
Not assessed because no procedures exist except for Project Procedures.

C. QA Department, the Quality Systems Division appeared to have an adequate program, except that a follow-up system to ensure timely receipt and acknowledgement of documents was not addressed.

IV. Areas (Actions) to be Considered

Procedures which describe document distribution should address follow-up systems and the disposition of obsolete documents.

I. Areas Examined

Personnel awareness and implementation of procedures for reporting of 10CFR 50.55(e) and 10CFR21 items to the NRC.

II. Assessment

The Shoreham Project personnel are aware of the requirements for, and methods to report items.

III. Conditions Found

1. The procedures for reportable items to the NRC are adequate.

2. Documentation reflecting status of items reported to the NRC is adequate.

3. Documentation of the reportability decision by the Project is adequate, including those items that were deemed not reportable.

4. Training of Project personnel on reporting requirements is limited.

IV. Areas (Actions) to be Considered

Even though the present reporting situation is satisfactory, the importance of this area should be reinforced by repeated (yearly) mandatory training sessions.

I. Areas Examined

A. Meter & Test Department

B. Purchasing Department

C. Nuclear Engineering Department

D. Nuclear Operations Support Division

E. QA Department

F. Field QA Division

G. OQA Division

H. Nuclear Operations - Plant

I. Nuclear Engineering - Fuel

II. Assessment

With few exceptions as noted in Section III, the contacted departments appear to have adequate procedures for the present scope of their QA activities. However, scopes of activities for all off-site departments may change as Shoreham becomes operational. This change in activities is also noted in NOC Policies which will become effective six months prior to fuel load. There is no present apparent overall effort to plan and issue procedures which will be required to provide timely implementation of the NOC Policies, and the Quality Program commitments in consideration for transition into the operating phase.

III. Conditions

A. The Meter & Test Department's procedures do not address the following QA activities: Processing nonconformances, Receipt Inspection, Periodic review of procedures, validation of changes to QA records, and the filing of records. There may also be a need to include in procedures, the basis for technical reviews of purchase requisitions

B. The Purchasing Department had previously initiated draft procedures to implement the QA Program, however those procedures have not been issued as of this date.

C. The present functions of the Nuclear Engineering Department, are being accomplished in accordance with Project Procedures. Possible future involvement in design activities during the operational phase, has not been addressed at this time.

D. The Nuclear Operations Support Division does not presently require any additional procedures. However, procedures which might be required during the operational phase such as FSAR updates and amendments to the operating license, have not been addressed at this time.

E. The QA Department and the OQAE appear to have adequate procedures for their present functions and are actively developing procedures for future activities.

F. Field QA Division

The division's procedures were reviewed during the assessment of the QA Department. Conversations with the division QA Manager, and the above assessment review revealed that all Field QA activities were adequately addressed in procedures and instructions.

G. OQA Division

The division's QA procedures are included in the plant program for maintaining the status of procedures. The plant's method and OQA's method for ensuring that QA procedures and instructions are reviewed periodically, are not covered procedurally by OQA.

H. Nuclear Operations - Plant

Nearly 2,000 procedures are in existence covering activities affecting quality. A cursory review of the list of plant procedures did not indicate any areas which were not covered.

I. Nuclear Engineering - Fuel

Quality activities are not presently covered by procedures. (Design control, organization, document control, QA records, etc. will be performed by the section.)

IV. Areas (Actions) to be Considered

A. Required procedures should be identified and schedules for their issue should be established to assure compliance with the QA Program.

B. Thought should be given to assigning this task to one group, to provide for tracking and monitoring the task to completion and to coordinate the compatability of the required procedural interfaces.

ATTACHMENT 8

LILCO'S NEW RELEASE AND VIEWGRAPHS

DESCRIBING ITS SHOREHAM PHYSICAL INSPECTION PROGRAM

ATTACHMENT 8

LILCO
LONG ISLAND LIGHTING
250 OLD COUNTRY ROAD
MINEOLA, N.Y. 11501

NEWS RELEASE

CORPORATE COMMUNICATIONS DEPARTMENT
JAN K. HICKMAN, Manager
(516) 228-2308

FOR IMMEDIATE RELEASE
May 12, 1982

Long Island Lighting Company today offered Suffolk County the opportunity for a full, independent physical inspection of all 32 safety systems at the utility's Shoreham Nuclear Power Station.

The scope of the review, which includes a design verification program, is believed to be unprecedented for a third-party inspection of a nuclear power plant in this country or abroad.

If the County accepts the proposal, about 25 inspectors from an independent consulting firm will begin the inspection shortly. The firm will be selected from among several consultants approved by the County last year.

The 25 engineers and designers are expected to spend up to 30,000 man-hours physically re-inspecting each of the reactor's safety systems, as well as safety test results, blueprints, construction drawings and other documentation.

The physical inspection and design verification process is expected to take three to five months of full-time work, and cost \$2 to \$3 million. The consulting firm's inspection forces will be based at the Shoreham site.

The review of the reactor's safety systems will involve the physical re-examination of all 50,000 feet of safety related piping. The plant's 2,400 critical pipe supports will also be reinspected. These supports range in weight from 50 pounds to ten tons.

Also scheduled for scrutiny are the strength of the reinforced concrete

-MORE-that comprises the reactor's "primary containment," and the location and configuration of safety related electrical and mechanical equipment.

A team of engineers from LILCO, Stone and Webster, the plant's architect-engineer; and General Electric, the reactor's designer, will provide support to the 25 inspectors. The utility group will supply design information such as blueprints, construction drawings and all other information that might be requested by the inspectors.

The methods of physical inspection and design verification will include, but are not limited to, the following:

- Independent ultrasonic testing of welds that connect safety related piping.
- Field examination of safety related electrical components, to make sure that the location and installation of the equipment conforms to the design documents.
- Field inspection of all safety related piping systems, including comparison of design drawings to actual installation.
- Analysis of safety systems' pre-operational test results. The results will be compared with the test criteria that the systems must meet.
- Monitoring of the reactor containment building's "bottom line" safety test -- the structural acceptance test of Shoreham's primary containment. In this test, slated for June, the reactor containment will be pressurized to simulate accident conditions. The independent inspectors will monitor this five-day test of the reactor containment's ability to withstand an accident.

In addition, the reactor's primary containment will be tested using the "Windsor Probe" technique. In this test, sharp metal probes are shot into the concrete containment, which ranges in thickness from four to

-MORE-seven feet. The depths of the probes' penetrations are measured to assess the strength of the reinforced concrete.

LILCO officials said that if any deficiencies are identified during the course of the inspection, they will be corrected. If the County agrees to the inspection program, progress reports and final results will be forwarded to the County and to the Nuclear Regulatory Commission (NRC), the Federal government agency that regulates the construction and operation of nuclear power plants in this country.

Officials also pointed out that the reactor's 32 safety systems, as well as non-safety related components, have already been examined, several times, by quality assurance inspectors. The NRC has also conducted independent inspections of the plant. The Shoreham plant currently meets all Federal and industry safety codes and regulations.

A list of the plant's 32 safety systems is attached.

-30-

SYSTEMS FOR INDEPENDENT PHYSCAL INSPECTION

Nuclear Boiler System
Reactor Water Recirculation System
Control Rod Drive Hydraulic Control System
Standby Liquid Control System
Neutron Monitoring System
Reactor Remote Shutdown System
Reactor Protection System
Process Radiation Monitoring System
Area Radiation Monitoring System
Residual Heat Removal System
Core Spray System
Main Steam Isolation Valve Leakage Control System
High Pressure Coolant Injection System
Reactor Core Isolation Cooling System
Reactor Water Cleanup System
Fuel Pool Cooling & Cleanup System
Reactor Building Standby Ventilation Control Room Chilled Water System
Main Steam System
Demineralized & Makeup Water System
Service Water System
Reactor Building Closed Loop Cooling Water System
Compressed Air System
Emergency Diesel Generators System
Reactor Primary Containment System
Primary Containment Inerting System
Reactor Building Ventilation System
Reactor Building Standby Ventilation System
Primary Containment Cooling System
Primary Containment Atmospheric Control System
Miscellaneous Heating, Ventilating and Air Conditioning Systems
Diesel Generator Ventilation System
Control Room Heating, Ventilating and Air Conditioning Systems

PHYSICAL INSPECTION/DESIGN VERIFICATION

I. OBJECTIVES:

1. Design Verification (by physical inspection) to demonstrate that the Plant is built according to design requirements.

2. Construction Quality Verification (by physical inspection, quality records review, and non-destructive testing) to demonstrate that Quality Assurance has been effective at Shoreham.

3. Preoperational Test Verification to insure that the system testing results conform to the Plant's safety requirements.

PHYSICAL INSPECTION/DESIGN VERIFICATION

II. SCOPE:

1. 100% of Safety Systems Included
2. ASME Piping
3. ASME Welds
4. Pipe Supports
5. Mechanical/Electrical Components
6. Concrete Strength of Primary Containment
7. Primary Containment Testing

DESIGN VERIFICATION OF SAFETY SYSTEMS

1. SCOPE: ALL SAFETY SYSTEMS (32)

2. METHOD: COMPARE THE "AS-BUILT" PLANT WITH THE DESIGN DRAWINGS FOR:

   A) CONFIGURATION

   B) COMPONENT LOCATION

   C) PROPER IDENTIFICATION

3. DOCUMENTS USED

   A) APPROVED FLOW DIAGRAMS (FM's) - 54

   B) E&DCR's - ALL APPLICABLE E&DCR's

ASME PIPING DESIGN VERIFICATION

1. **SCOPE:** REVERIFICATION OF SAMPLE OF PIPING "AS-BUILT" DRAWINGS IN ASME SAFETY SYSTEMS

2. **METHOD:** FIELD WALK SELECTED PIPING ISOMETRIC DRAWINGS AND VERIFY:

   A) PIPE DIAMETER

   B) ROUTING

   C) PIPE SUPPORT LOCATIONS

   D) COMPONENT IDENTIFICATION

3. **DOCUMENTS USED**

   A) LATEST ISOMETRIC DRAWINGS

   B) LINE-WALK PROCEDURES

REINSPECTION OF ASME WELDS

1. SCOPE: REINSPECTION OF SAMPLE OF WELDS IN ALL SAFETY SYSTEMS

2. METHOD: A) ULTRASONICALLY TEST CLASS 1 AND 2 WELDS

   B) VISUALLY INSPECT CLASS 3 WELDS

3. DOCUMENTS USED

   A) SHOREHAM PSI PROCEDURES AND ACCEPTANCE CRITERIA

   B) ORIGINAL QA WELD DOCUMENTATION PACKAGES

PIPE SUPPORT VERIFICATION

1. SCOPE: ALL LARGE BORE SAFETY RELATED PIPE SUPPORTS - ~2,400 TOTAL

2. METHOD: FIELD INSPECTION OF ALL SAFETY RELATED LB SUPPORTS AND COMPARISON WITH DRAWINGS FOR:

   A) PROPER LOCATION

   B) PROPER CONFIGURATION PER DESIGN DRAWINGS

   C) HARDWARE INSTALLATION (NUTS, BOLTS, ETC.)

3. DOCUMENTS USED

   A) PIPE SUPPORT DRAWINGS (BZ's)

   B) APPLICABLE QA PROCEDURES

MECHANICAL/ELECTRICAL COMPONENT VERIFICATION

1. SCOPE: REVERIFY THAT A SAMPLE OF MECHANICAL AND ELECTRICAL COMPONENTS HAS BEEN INSTALLED IN ACCORDANCE WITH DETAILED DESIGN DRAWINGS

2. METHOD: FIELD INSPECTION OF SELECTED COMPONENTS AND COMPARISON TO DESIGN DRAWINGS FOR:

   A) DIMENSIONAL CHECKS OF PIPE AND COMPONENT SUPPORTS.

   B) VERIFYING THAT BREAKERS, RELAYS, JUNCTION BOXES HAVE BEEN INSTALLED CORRECTLY.

   C) INSPECT ELECTRICAL PANELS FOR INSTRUMENTS, SWITCHES, ETC.

3. DOCUMENTS USED

   A) PIPE SUPPORT DRAWINGS

   B) PANEL AND ELEMENTARY DRAWINGS

PRIMARY CONTAINMENT CONCRETE STRENGTH

1. SCOPE: ALL CONCRETE PLACEMENTS (POURS) OF THE SHOREHAM PRIMARY CONTAINMENT (36)

2. METHOD: TEST EACH POUR FOR COMPRESSIVE STRENGTH USING THE WINDSOR PROBE TECHNIQUE

3. DOCUMENTS USED

   A) QA DOCUMENTATION
      (POUR CARDS)

   B) APPLICABLE QA PROCEDURES

PRIMARY CONTAINMENT TESTING

1. SCOPE: Test Performance, and Results of the Primary Containment Structural Acceptance Test (SAT)

2. METHOD: The Contractor Will Review the SAT Test Procedures and Establish Hold Points for His Personnel to Witness the Test, Data Acquisition, and Results Recording. He will Review the Results to Insure That the Containment meets its Design Requirements

3. DOCUMENTS USED

   a) SAT Test Procedures

   b) Test Data Forms

MATERIAL QUALITY RECORD VERIFICATION

1. SCOPE: REVIEW OF SAMPLE OF PIPE SPOOL QA DOCUMENTATION PACKAGES FOR PROPER MATERIAL CERTIFICATION

2. METHOD: QA INSPECTORS WILL REVIEW DOCUMENTATION AGAINST CODE AND SPECIFICATION REQUIREMENTS FOR:

   A) PHYSICAL PROPERTIES

   B) CHEMICAL PROPERTIES

   C) SUBSUPPLIER CERTIFICATIONS

3. DOCUMENTS USED

   A) ASME CODE

   B) SPECIFICATION SH1-56

   C) SPECIFICATION SH1-75

PRE-OPERATIONAL TEST VERIFICATION

1. SCOPE: ALL PRE-OPERATIONAL TESTS (PTs) COMPLETED AND ACCEPTED THROUGH JULY 1, 1982.

   - THERE ARE 180 PTs AND ATs
   - ~70% WILL BE COMPLETE BY 7/1/82

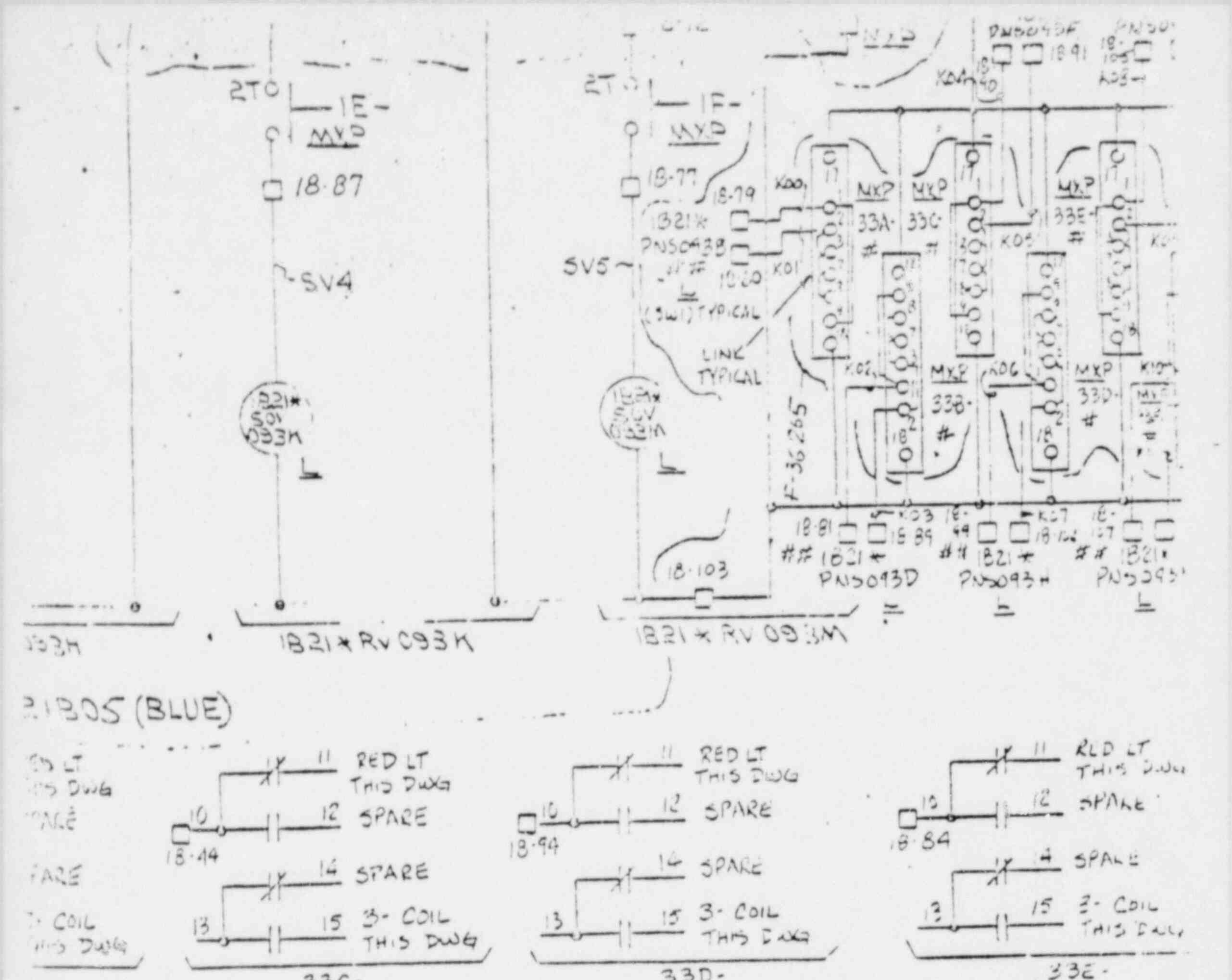
2. METHOD: REVIEW ACTUAL PT TEST DATA AND COMPARE TO PT ACCEPTANCE REQUIREMENTS TO INSURE THAT SAFETY SYSTEMS ACTUALLY PERFORM AS DESIGNED

3. DOCUMENTS USED

   A) APPROVED* PT PROCEDURES

   B) APPROVED* PT TEST RESULTS

* APPROVED BY JOINT TEST GROUP (JTG)



CCF B21/1

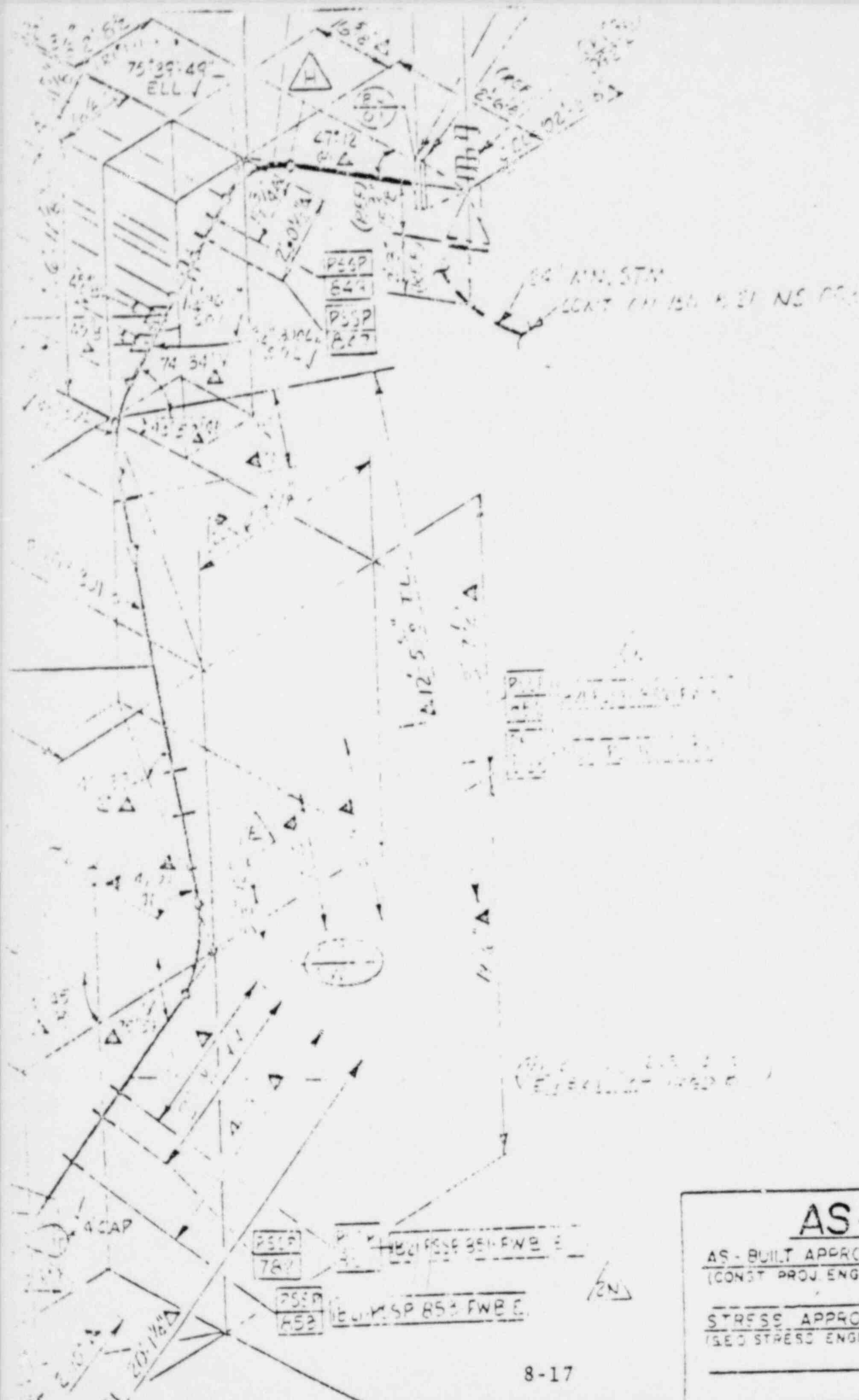
NOTES:

1. APPLICABLE LOGIC: NONE
2. # R.B. DENISON INC., PROXSWITCH, MODEL WE-77/E
3. ## REFERENCE ANDERSON, GREENWOOD & CO. DRAWING 04-3553 (G&W NO 6.53-B)
4. LETTERS IN PARENTHESIS ARE ANDERSON, GREENWOOD & CO., IDENTIFICATION

	SPARES
[illegible]	IT,1B,2B
[illegible]	IT,1B,2B
[illegible]	IT,1B,2B
[illegible]	IT,1B,2B
[illegible]	IT,1B,2B
[illegible]	IT,1B,2B

NUCLEAR SAFETY RELATED
Q.A. CAT. II

AC ELEMENTARY DIAG 120V
DOWNSTREAM DW VAC BKR VV'S



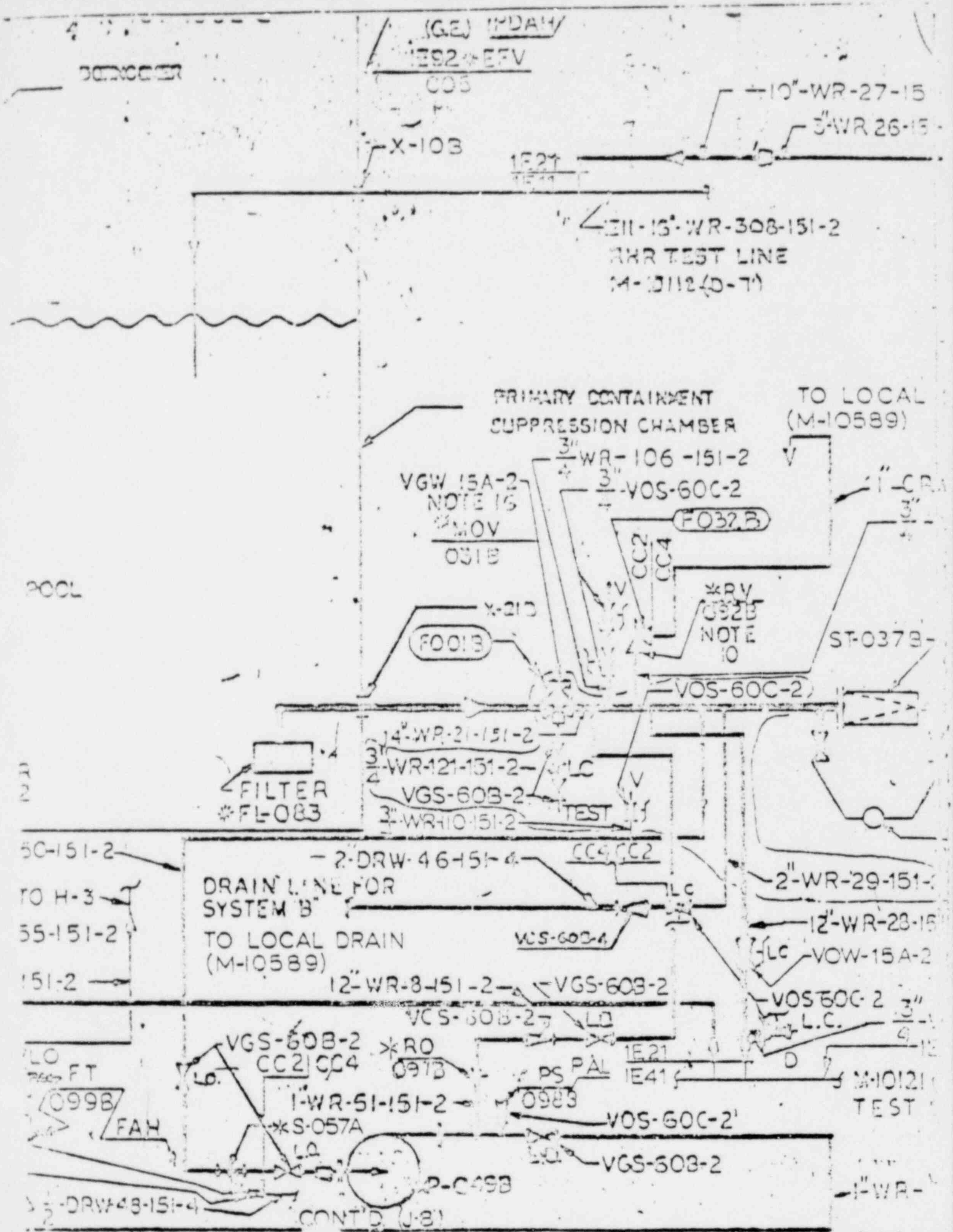
AS - BUILT

AS - BUILT APPROVAL:
(CONST. PROJ. ENG. ☐)(OR HIS DESIGNEE)

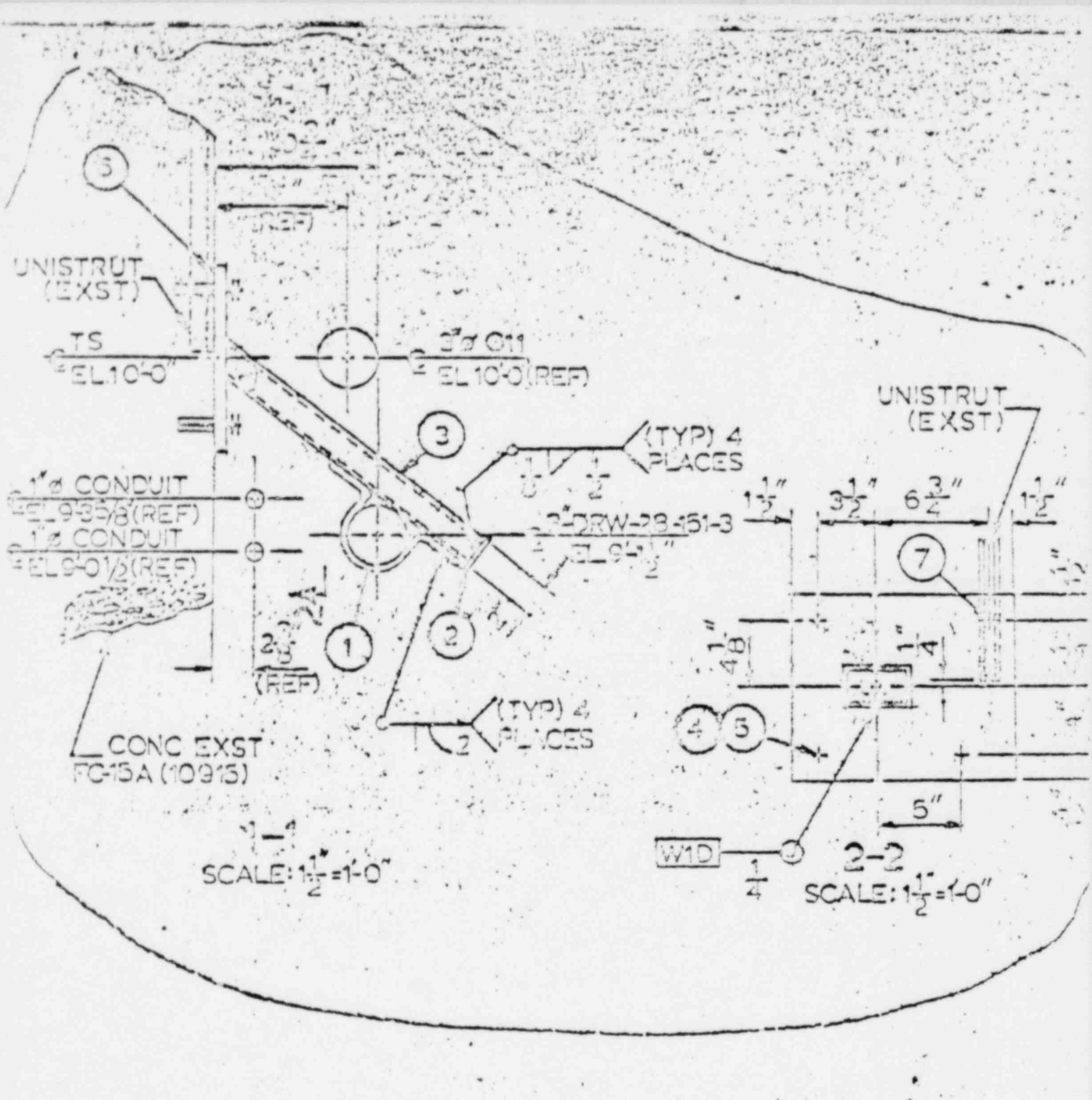
______________ DATE ____

STRESS APPROVAL:
(S.E.D STRESS ENGINEER)

______________ DATE ____



CORE SPRAY & RHR LOOP LEVEL SYSTEM



SUPPORT	FX (LBS)	FY (LBS)	FZ (LBS)	MX (FT-LBS)	MY (FT-LBS)	MZ (FT-LBS)
PSR1836	± 107	- 307	—	—	—	—

8-19

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ATTACHMENT 9

LILCO'S INDEPENDENT DESIGN REVIEW

OF THE CORE SPRAY SYSTEM

ENCLOSURE F

INDEPENDENT DESIGN REVIEW COMMITMENT

SHOREHAM NUCLEAR POWER STATION INDEPENDENT DESIGN REVIEW

Introduction

In order to provide added assurance that all aspects of Shoreham's safety-related piping design and installation have been properly performed from both a technical and Quality Assurance standpoint, the Long Island Lighting Company (LILCO) has obtained the services of an outside consultant to conduct an independent stress analysis and support design review on selected portions (one loop) of the safety-related, seismic Category I, Low Pressure Core Spray System (E21).

This independent review will be conducted in such a manner as to ensure that all individuals involved are free of substantive interest in either the Long Island Lighting Company or its agents - namely, the General Electric Company (GE) and Stone & Webster Engineering Corporation (SWEC).

All services to be performed will be in accordance with the terms and agreements to be defined in an appropriate Engineering Services scope of work. This work order will further serve as a basis for program review and approval by the NRC Staff or its consultants.

Objectives

The purpose of this review will be to independently verify and confirm that for the as-installed piping, components and supports:

1) the design criteria, design bases, implementation, and documentation were consistent with FSAR/DAR requirements.

2) the design data used by interfacing internal and external design organizations and disciplines were properly controlled and consistently applied.

3) the design modifications resulting from field and engineering change requests were properly reconciled or incorporated into the final design.

4) the Quality Assurance program properly monitored and documented the design, procurement, and installation procedures utilized on Shoreham.

5) the "as-built" documentation reflects the actual plant configuration and has been properly reconciled to the as-designed condition.

Scope of Work

The scope of audit will focus on a detailed review of selected E21 system piping, pipe supports, pipe mounted mechanical equipment, containment penetration sleeves, and associated components within the Core Spray System boundaries that are shown darkened on SWEC Drawing #11600.02-FM 23A (refer to attached flow diagram).

The significant areas for review shall emphasize the actual details of the technical engineering process and will include:

1) The adequacy of the design requirements as delineated in the GE and SWEC LPCS system specifications and associated standards, procedures and drawings for consistency with design criteria, NRC requirements, and FSAR commitments. This will include a review of the applicable design documentation for piping system design, pipe material, pipe support, pipe-mounted mechanical components, floor-mounted mechanical components, containment penetrations, and any other interface necessary to complete those designs. Design document review will include, but not be limited to:

   o Design Specification for Piping Engineering and Design (SH1-171)
   o Specification for Field Fabrication and Erection of Piping (SH1-056)
   o Design and Fabrication of Nuclear Power Plant Piping Supports (SH1-068)
   o Design and Erection Tolerances for Pipe Supports (EMD 81-02)
   o Procedure for Seismic Cat I As-Built Piping Review and Reconciliation (Project Procedure No. 42)

2) Based upon the design requirements defined in the technical specifications, the independent reviewer will review the adequacy of the design and analysis for the as-installed configuration of piping, components, and supports necessary for system operating function. For each qualification analysis the reviewer will verify and confirm the adequacy of:

   - the mathematical and/or computer model used
   - the <u>input</u> of design bases, loading conditions & loading combinations
   - the applicable codes and standards
   - the conformance to acceptance criteria
   - the appropriate interface requirements
   - the resolution of design change requests
   - the final reports, drawings and conclusions made

   The technical review will consist of the following minimum activities:

   A. Pipe Stress Analysis Review

2\. A. 1. Input Data Check

- o Internal piping pressure
- o Thermal load cases
- o System operating modes
- o Seismic spectra and anchor movements
- o SRV spectra and anchor movements
- o Fluid transient and other occasional loads, as applicable

2\. Piping Model Check

- o Piping configuration geometry
- o Piping section physical and material properties
- o Support and restraint stiffness, function, location, and orientation
- o Fittings, nozzles, and valves
- o System boundaries, and code classifications per 10 CFR 50

3\. Pipe Stress Related Calculations

- o Stress intensification factors or indices
- o Flow induced transient loads, pool drag/impact loads
- o Valve model natural frequency
- o Thermal transients (ΔT_1, ΔT_2)
- o Support, restraints, penetration, nozzle load summaries

4\. Stress Reports Issued and Conclusions

- o Certification (for ASME III, Class 1)
- o Load cases
- o Load combinations
- o Pipe stress code compliance
- o Support, restraint, penetration, nozzle allowable check
- o Valve acceleration, end loads check
- o Functional capability check

B. Pipe Support Analysis Review

1\. Input Data Check

- o Support loads generated for all essential load cases
- o Support types and locations
- o Piping deflections generated for all essential load cases
- o Pipe stress at integrally welded attachments

F 4

2. B. 2. Design Calculations

- o Member sizing, stiffness, stability
- o Weld calculations
- o Stress allowables
- o Vendor allowables for standard hardware
- o Computer models
- o Expansion bolt allowables and base plate flexibility

3. Review of Design Drawings and As-Built Installation

- o Proper function, location and orientation specified
- o Proper clearances specified
- o Proper structural member and weld sizes
- o Proper adjustments of spring and snubber components
- o Interferences

This analytic review shall be done for normal/upset service levels which consist of the appropriate loading combinations identified in the specifications. In addition, the above review shall assume validation of all referenced computer codes, and dynamic response spectra for seismic and hydrodynamic events. It is not anticipated that the independent reviewer perform detailed calculations or analysis. It is sufficient that the existing documentation and calculations be reviewed to determine their validity; however, the independent reviewer may perform any calculations as he feels necessary.

3. The technical design and analysis of vendor qualified mechanical and electrical components that have previously received third party review by SWEC or G.E. to current NRC (SQRT) guidelines will be verified for satisfaction of interface requirements. Furthermore, selected representative pipe-mounted and floor-mounted mechanical and electrical component's qualification reports will be independently assessed to ensure that their mechanical and structural design requirements are satisfied.

4. The QA/QC process and documentation review shall include a review of QA/QC records for, and a physical inspection of the system covering the following minimum activities:

   A) the designation of safety-related items to determine whether the systems' structures and components have been properly classified in accordance with 10CFR; 50.

   B) the training and qualification records of construction personnel.

C) the records concerning the identification and control of installed material, parts, and components

D) the records concerning the control of special processes

E) the Nonconformance and Disposition (N&D) Report process and, for the installed items under the scope of review, the nonconformance, disposition, corrective action measures, and close-outs taken

F) the results of SWEC Engineering Assurance (EA) and LILCO audits, follow-ups and close-outs

G) the engineering and field design and drawing change preparation, review, approval and incorporation

H) the as-built system design verification process for piping, pipe-mounted components and supports. This will involve a field walkdown and visual inspection of the system, including field welds, support locations, etc.

I) the records of receiving inspections and test results

J) the records of material certification

K) the verification of the torque of bolts

L) the nondestructive test records

M) the adequacy of the LILCO and SWEC QA programs and their implementation based on the adequacy of the above

In summary, the overall program scope will be structured to insure that the design process properly converted the Shoreham design basis specified in its FSAR into documents which accurately reflect existing field conditions and satisfy all qualification requirements.

Results

The output from this study will be a final report summarizing all documents reviewed, overall evaluation of the design and its control for the selected scope of review, description of conservatisms identified and a record of all potential findings, and observations. A potential finding is identified as a weakness in the design or design control process. A potential finding which is determined to be accurate and has potential for significant impact on the design adequacy will be identified as a finding. If the potential finding does not have the potential for significant impact, it will be identified as an observation. Findings and observations may be classified as technical, design, traceability or procedural. In each instance, the potential finding will be verified as accurate by LILCO and the design organization involved and submitted to an internal finding review committee within the independent review organization composed of senior technical management for further assessment. If the committee determines the potential finding to be a reportable finding or observation, LILCO and the design organization involved may propose a remedial action plan which will then be assessed for adequacy and acceptance by the independent review committee. All reportable findings will be identified to the NRC by LILCO in accordance with 10CFR50.55(e) or 10CFR 21. The final report will contain records of all potential findings, observations and findings along with their associated remedial actions, if any. For each, the report will document an assessment of its extent, evaluation of safety impact, results of any further analysis (if required), and recommendations based on review of proposed remedial actions.

Furthermore, an interim report is to be provided midway through the scheduled design review process and is intended to reflect review completeness, progress and preliminary results to date. All review reports, comments, observations, and correspondence shall become Project QA Records of the reviewing organization. If requested by LILCO, copies of any or all of this information will be transmitted.

Independence

The General Electric Company, (GE) and Stone & Webster Engineering Corporation (SWEC) will make available to the independent reviewer all documents and design calculations requested by the independent reviewer which are pertinent to perform this review. Both the interim and final reports concerning this review shall be sent unedited and concurrently to LILCO, the NRC staff, and SWEC. Prior review or editorial control of the written final report by any of these parties is not permitted. All individuals involved in this review shall be free of substantive interest in LILCO or its agents. For example, the following are precluded:

a) An immediate family member who is employed by LILCO or who is engaged directly or indirectly in the design or construction of the Shoreham Nuclear Power Station.

b) A cumulative ownership and creditor interest in LILCO which exceeds 5% of their gross family annual income.

c) Any work experience in design, construction or quality assurance of the Shoreham Nuclear Power Station or with LILCO currently or within the past 5 years.

In addition, each individual participating in the review process will agree in writing:

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a) Not to disclose to third parties (not involved in the project) information revealed in the course of his assignment without prior authorization of LILCO.

b) The return to LILCO of all proprietary documents, drawings, and calculations used in the course of his assignment.

c) To notify the LILCO Project Manager if, during the term of this project, he fails to satisfy the independence requirements above.

ATTACHMENT 10

GUIDELINES FOR INDEPENDENT INSPECTION

TASK C - QA/QC PROGRAMS

ATTACHMENT 10

GUIDELINES FOR INDEPENDENT INSPECTION

TASK C - QA/QC PROGRAMS

Subtasks	Description
C-1	Determine whether and how the QA/QC programs of LILCO and its major contractors for design and construction activities comply with each of the 18 criteria of Appendix B to 10 CFR 50 and the guidance provided in applicable Regulatory Guides during the time span encompassed by Shoreham design and construction activities;
C-2	Assess the organizational independence of inspection and construction supervision;
C-3	Assess the adequacy of the training and qualification of welders, the qualification of welding procedures, and the traceability of welding records to welders;
C-4	Evaluate the internal audit frequency, the reporting of audit results to management, and the timeliness of audit follow-ups and resolutions;
C-5	Determine the adequacy of documentation of tests and inspection results;
C-6	Verify the acceptable calibration and accuracy

Subtasks	Description
	of measuring and test equipment have been and are being used;
C-7	Review the training and qualification records for personnel performing NDT activities;
C-8	Review the records concerning the identification and control of installed materials, parts and components, including the acceptance of incoming equipment and the release of uninspected equipment for installation;
C-9	Review the records of identification, segregation, review and release of non-conforming items and the records concerning the adequacy of disposition of such non-conformances;
C-10	Evaluate the chronological records of detection and resolution of all safety-related problems detected during construction, including corrective action measures;
C-11	Determine the adequacy of selection, evaluation and source inspection for suppliers of items and services important to safety;
C-12	Determine the effectiveness of LILCO's and its major contractors' QA/QC programs and their implementation based on all of the above;

Subtasks	Description
C-13	Determine the root causes of any QA/QC program breakdowns which contributed in whole or in part to the deficiencies identified in Tasks A and B;
C-14	Determine whether and how the LILCO QA/QC program for operations comply with the QA requirements in Appendix B to 10 CFR 50, including:

(i) the qualification of the Shoreham QA/QC staff;

(ii) the availability of QC personnel on off-shifts;

(iii) the availa' ility of "as-built" drawings;

(iv) the select.. of replacement materials and parts for safety-related items;

(v) the applicability of the QA/QC program to replacement electrical and instrumentation components, modules, and equipment;

(vi) the handling and installation of replacement parts and materials for safety-related items;

(vii) the program for procurement of non-safety-related replacement materials and parts;

(viii) a comparison of the LILCO operation QA/QC program to NRC Regulatory Guides cited in the FSAR related to QA/QC acitivties.