LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION

P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

July 21, 1981

SNRC-601

RECEIVED JUL 22 1981 U.S. NUCLEAR REGULATORY COMMISSION

Mr. Harold R. Denton
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SHOREHAM NUCLEAR POWER STATION - Unit 1
Docket No. 50-322

Dear Mr. Denton:

Enclosed herewith are sixty (60) copies of LILCO responses to specific NRC concerns which were previously identified as requiring additional information to complete NRC review. Attachment A provides a list of the specific responses included.

If you require additional information or clarification, please do not hesitate to contact this office.

Very truly yours,

B R McCaffrey

B. R. McCaffrey
Manager, Project Engineering
Shoreham Nuclear Power Station

Enclosures

cc: J. Higgins

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FC-8935

Attachment A

Additional information is provided for the following items:

1) SER Open Item No. 36 - Containment Purge System

2) SER Open Item No. 44 - Level Measurement Errors

3) SER Open Item No. 52 - Management Organization

4) SER Open Item No. 59 - LILCO Response to Staff Position Regarding Interim Actions for Control of Heavy Loads

5) SER Open Item Nos. 62 (Reg. Guide 1.58 Rev. 1) and 63 (R.G. 1.146)

6) NUREG-0737 Item II.K.3.28 - Verify Qualification of Accumulators on Automatic Depressurization System Valves

7) NUREG-0737 Item III.D.1.1 - Primary Coolant Sources Outside the Containment Structure

Item No. 36 - Containment Purge System

The drywell and suppression chamber vent purge valves are six inches in size and 150 pound class. The governing code for design is ASME Section III Code Class 2. The valves have a design pressure of 48 psig and temperature of 90 F. The ambient temperature specified for the valves' environment is 40 - 120 F. The valve body form is globe type with quick opening trim. The valve actuator is a spring and diaphragm with a 2-inch stroke. Air to the actuator is 125 psig. Valve actuator operates on 80 psig. In addition to ASME Section III Class 2 design, the valves have also been seismically qualified (including Mark II loads) by calculation. Additional seismic evaluation was performed which required analysis to assure the function and structural integrity. The valves have been analyzed for stress and deflection due to combined seismic loads (simultaneously vertical and horizontal), pressure, and maximum operator load, and found that valve function is unimpaired. In addition, the associated solenoid valves are environmentally qualified in accordance with the requirements of NUREG-0588.

These valves were also stroke tested immediately following testing for leakage, (hydrostatic test). The bench test was performed in a test fixture which did not account for an actual system downstream piping configuration nor could a constant differential pressure be maintained across the valve during stroke test.

Shoreham Outstanding SER Issue #44
Level Measurement Errors

LEVEL MEASUREMENT ERRORS

(due to environmental temperature effects on level instrument reference legs)

Reactor vessel water level is measured by means of a produced differential pressure between a reference leg and a variable leg. The reference leg is connected to the upper part of the vessel (steam zone) and provides the constant head using an overflow type condensing chamber. The variable leg is connected to the lower part of the vessel. The produced differential pressure is therefore a function of water level.

General Electric has conducted a review of the effects of high drywell temperature on reactor vessel water level instrumentation. Although instrument accuracy is affected by varying drywell temperature and boil-off in the reference leg, there would be no impact on the scram or other level trip functions, nor would post-accident operator action be impaired.

The worst case scenario evaluated was as follows:

- o Small break LOCA occurs in drywell
- o Scram and auto ECCS (ADS and LPCI/CS) are actuated
- o Some time after LPCI/Core Spray have reestablished normal RPV water level, operator diverts or shuts off LPCI/Core Spray from RPV
- o 10-12 hours after the initiating event, PRV water level error is at its maximum
- o Upon receipt of low level alarm (Level 2), operator must re-initiate LPCI/Core Spray injection.

Were this unlikely series of events to occur it would be necessary for the operator to restart or redirect low pressure ECCS to the PRV in order to avoid core uncovery. Shoreham specific analysis verified that the operator would have from 10-15 minutes to redirect low pressure ECCS for the worst case scenario presented above, even with a nonconservative water level indication associated with long term boiloff. Operating procedures and training will specifically address the need to be aware of this phenomenon in this small break long term LOCA. Emergency procedure 29.023.09, Reactor Pressure Vessel Flooding, was submitted in SNRC-599 dated July 20, 1981, specifically assuring a conservative response to this phenomenon.

The following statements reflect the resolution of the outstanding items associated with SER Open Item No. 52:

1. Effective July 1, 1981 there has been a reorganization of some of the functions under the office of Vice President Nuclear. Attached are revised Figures 13.1-1 and 13.1-4 which reflect the current organization.

2. In response to the concern over off-site engineering support resources LILCO will:

   a) Obtain a Continuing Services Contract with a qualified Architect/Engineer firm to provide supplementary engineering support, to be in place prior to Fuel Load.

   b) To supplement the technical support of the nuclear organizations, a minimum of 10 engineering personnel, assigned to the Corporate Engineering Office, will be designated for nuclear support prior to Fuel Load. Their first priority will be to respond to the needs of the Shoreham Plant as required.

3. To further ensure that corporate management decisions adequately reflect a concern for or consideration of matters important to safe operation of the plant, LILCO will secure the services of an advisor to the Vice President Nuclear for matters affecting operation of the plant. This person will have substantial BWR power plant operating and management experience and will be in place prior to Fuel Load and for a year following Fuel Load or until such time as the Vice President and his staff have accrued sufficient experience in managing an operating plant.

4. At and following Fuel Load, a dedicated Communicator will be assigned to each shift. This person may be one of the Field Operations personnel not required to be on shift. Security personnel will not be used.

5. While LILCO feels that insufficient credit is being given for the real-time Digital Radiation Monitoring System, we will nontheless provide Rad/Chem Technician coverage on each shift, in addition to the previously committed Health Physics Technician.

6. Overtime Policy in compliance with NUREG-0737 (1.A1.3) will be written and submitted to the NRC. The policy will be pointed to safety related functions.

7 a) To clarify our previous position, the Operation Engineer will meet the experience qualifications required by ANSI N18.1 - 1971 for the Operations Manager. An updated resume for J. A. Notaro is attached.

   b) Since the Assistant Operations Engineer acts as alternate to the Operations Engineer, his experience qualifications will also meet ANSI N18.1 - 1971 requirements for the Operations Manager.

c) Nuclear Assistant Station Operators (NASO) will meet the requirements of ANSI N18.1 - 1971 for Licensed Reactor Operators and thus have a minimum of 1 year nuclear power plant experience.

8. To supplement the operating experience of the operations staff, one additional person with substantive BWR operating experience will be an on site advisor to Senior Plant Management. This role will be filled for one year following Fuel Load or until such time as the Plant Manager or Chief Operating Engineer and their staff have accrued sufficient experience in managing an operating plant.

9. A general revision to FSAR section 13.2 is under preparation and will be submitted by December 31, 1981.

10. LILCO will have at least one individual on site on each operating shift with substantive previous BWR operating experience, including startup and shutdown of a BWR. This experienced person will be assigned to each shift reporting to the Watch Engineer until the plant is operating at 100% power. A description of the qualifications of these individuals will be submitted as soon as they are available.

11. Supplementing previous responses to NUREG-0737, Item I.B.1.2, LILCO commits to establishing an On-site Safety Review Group. This group will be composed of a Chairman and five dedicated multi-disciplined personnel, the majority of which shall not be recent college graduates. The Chairman shall report directly to the Manager Nuclear Operations Support and shall transmit formal analyses and recommendations to him for presentation to appropriate corporate management. The Chairman shall be located off site and shall provide overall direction to the group. He shall maintain communications with the on-site group through regularly scheduled on-site meetings and other informal contacts.

The five dedicated multi-disciplined personnel shall be located on site. One member of this group shall be assigned as Group Leader to coordinate day to day assignments and activities. Principle functions of the OSRG shall include:

a) Assessment of the operating experience of the station and stations of similar design.

b) Examination of appropriate plant operating characteristics and industry/NRC issuances.

c) Review of plant activities such as maintenance, modification, operational problems, and operational analysis.

d) Surveillance of plant operations and maintenance activities to provide verification that these activities are performed correctly and with minimum human error.

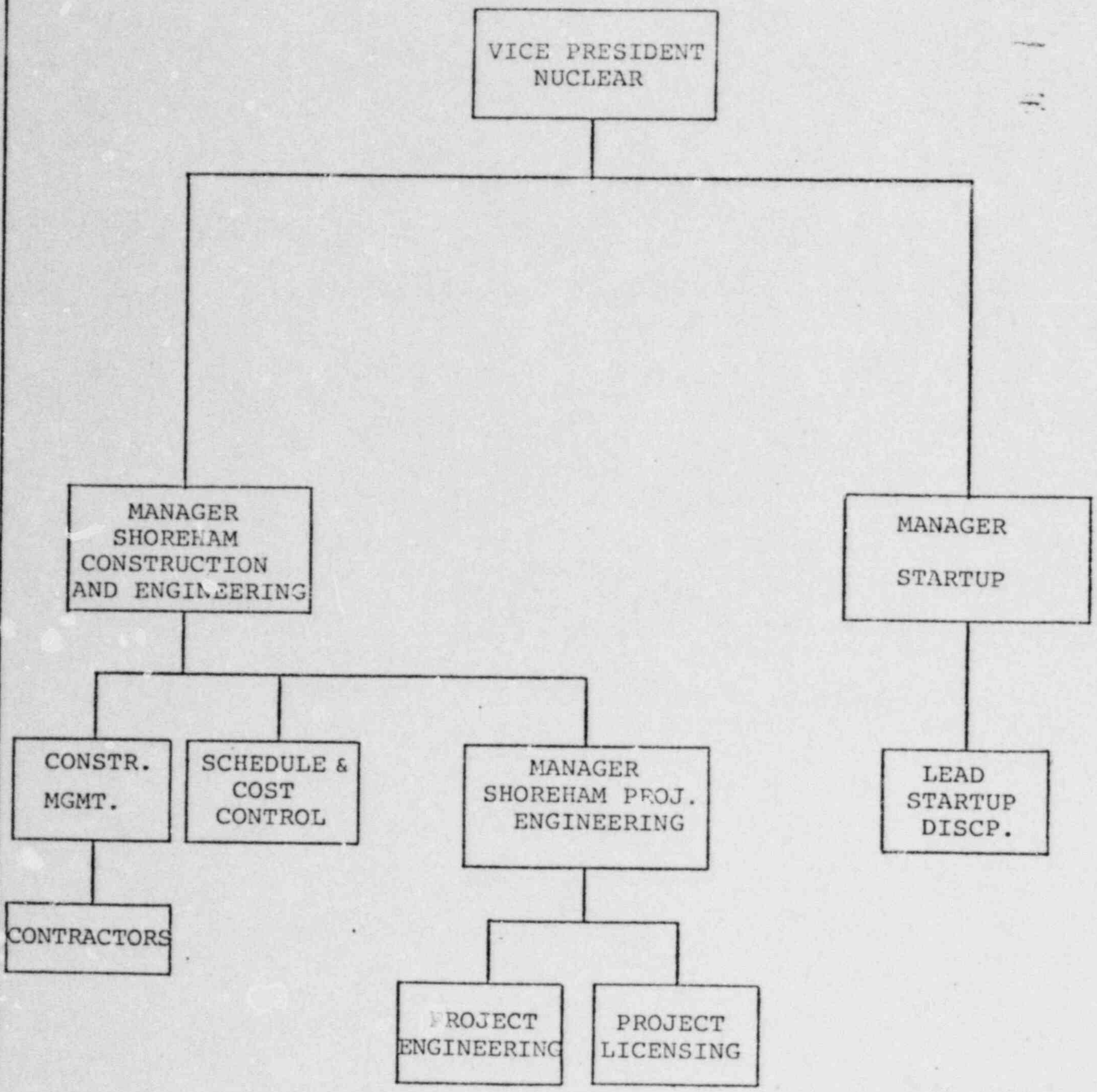

FIGURE 13.1-4
SHOREHAM PROJECT ORGANIZATION

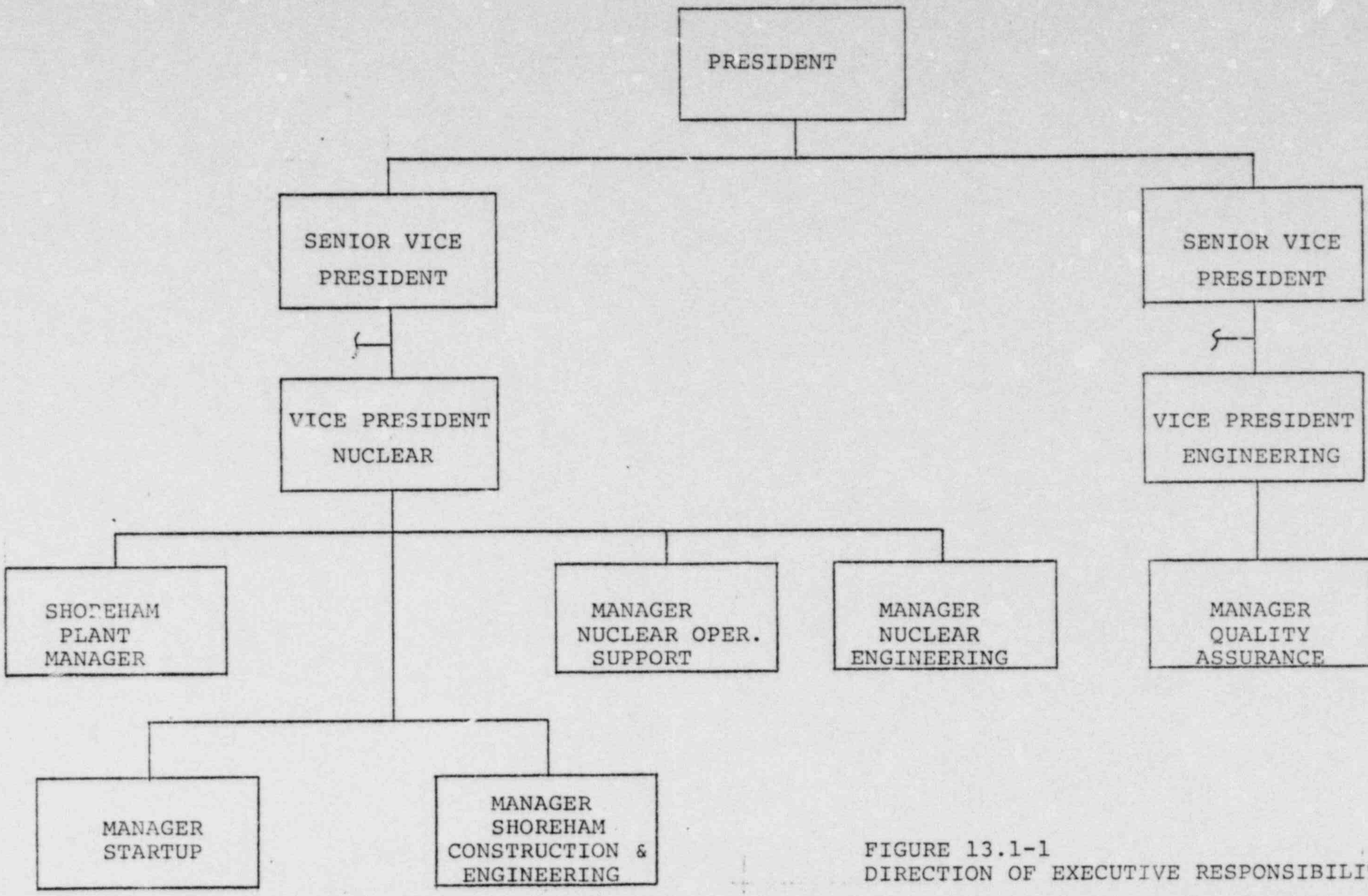

FIGURE 13.1-1
DIRECTION OF EXECUTIVE RESPONSIBILITY

JACK A. NOTARO
Operating Engineer
Long Island Lighting Company

Assigned as Operating Engineer of the Shoreham Nuclear Power Station in July, 1978. Responsible for the development and implementation of the Station's operational activities including the direction of day to day operation of the unit; startup, operation and shutdown of all station equipment; implementation of initial, requalification, and replacement training programs for licensed and unlicensed operators; and development, review, and implementation of the operations section of the Station Operating Manual.

Graduated from Brooklyn Technical High School in 1965. Graduated from City College of New York in 1970 with a Bachelor's Degree in Mechanical Engineering. Received a Masters of Business Administration Degree in 1974 from Adelphi University.

Completed the General Electric Co. Boiling Water Reactor Simulator Program in July, 1976, and obtained certification as a Senior Reactor Operator.

Completed the following industry seminars and training programs:

a) BWR Design Orientation - General Electric Co.
b) BWR Technology - General Electric Co.
c) BWR Observation Training - General Electric Co.
d) Nuclear Power Plant Technology - General Physics Corp.
e) Radiation Protection - LILCO Evening Institute
f) Basic Health Physics - Brookhaven National Laboratory
g) Vibration Analysis - IRD Mechanalysis, Inc.
h) Statics, Strength of Materials, & Dynamics - LILCO Evening Institute
i) Management of Maintenance Storekeeping & Inventories - Management Dynamics Institute
j) QA for the Nuclear Industry - Stat-A-Matrix and General Physics Corp.
k) Inservice Inspection & QA During Operations - Southwest Research Institute
l) Basic Radiography - Corvair Division of General Dynamics
m) Magnetic Particle & Liquid Penetrant Testing - Magnaflux Corp.
n) Basic Ultrasonics - Automation Industries
o) Nuclear Power QA - Long Island Section of AWSC
p) Inservice Inspection Symposium - Mirror Insulation
q) Operations Quality Assurance - Stat-A-Matrix
r) Fire Fighting Training - Suffolk County Fire Department

June 1970 - Present

Employed by the Long Island Lighting Company in the Electric Production Department.

March 1973 - July 1978

Assigned to the Shoreham Nuclear Power Station in the Quality Assurance Section and subsequently promoted to Station Operating Quality Assurance Engineer responsible for the Section in July, 1974

Responsibility included initial development of the operational quality assurance program. Responsible for all aspects associated with its implementation at the station including reviews, audits, surveillance, inspections, selection and training of personnel, development of procedures and instructions, and the utilization of consultants and contractors. Additional responsibilities included licensing and inspection activities associated with the U.S. Nuclear Regulatory Commission and interfacing with external and internal organizations required to implement the operational quality assurance program.

January 1972 - March 1973

Assigned to the Electric Production Department Staff. Assigned duties included maintenance scheduling, manpower allocation, equipment testing, station performance analysis and special projects.

June 1970 - January 1972

Assigned to the Maintenance Section in the Northport Power Station. Assigned duties included assisting in outages of both a scheduled and forced nature as well as maintaining plant equipment and systems, and completing special projects.

A member of the American Society for Quality Control, the Edison Electric Institute - Quality Assurance Task Force (EEI-QATF) and the EEI-QATF Operations Subcommittee.

Item No. 59 - LILCO Response to Staff Position Regarding Interim Actions for Control of Heavy Loads

The following requirements will be implemented at Shoreham prior to the placement of new fuel assemblies in the Reactor Building:

1) Safe load paths will be defined in accordance with the guidelines set forth in Section 5.1.1 (1) of NUREG-0612 with the exception that floor markings will be limited to "where practical" due to the inherent radial and polar pathways traveled by the polar crane.

2) Procedures will be developed and implemented per the guidelines set forth in Section 5.1.1 (2) of NUREG-0612.

3) Crane operators will be trained, qualified and conduct themselves per the guidelines set forth in Section 5.1.1 (3) of NUREG-0612.

4) Cranes will be inspected, tested, and maintained in accordance with the guidelines set forth in Section 5.1.1 (6) of NUREG-0612.

NUREG-0612

Control of Heavy Loads at Nuclear Power Plants

Resolution of Generic Technical Activity A-36

H. George, Task Manager

Office of Nuclear Reactor Regulation

U. S. Nuclear Regulatory Commission

UNITED STATES NUCLEAR REGULATORY COMMISSION

(1) Provide sufficient operator training, handling system design, load handling instructions, and equipment inspection to assure reliable operation of the handling system; and

(2) Define safe load travel paths through procedures and operator training so that to the extent practical heavy loads avoid being carried over or near irradiated fuel or safe shutdown equipment; and

(3) Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

Certain alternative measures may be taken to compensate for deficiencies in (2) and (3) above, such as the inability to prevent a particular heavy load from being brought over spent fuel (e.g., reactor vessel head). These alternative measures can include: increasing crane reliability by providing dual load paths for certain components, increased safety factors, and increased inspection as discussed in Section .1.6 of this report; restricting crane operations in the spent fuel pool area (PWRs) until fuel has decayed so that off-site releases would be sufficiently low if fuel were damaged; or analyzing the effects of postulated load drops to show that consequences are within acceptable limits. Even if one of these alternative measures is selected, (1) and (2) above should still be satisfied to provide maximum practical defense-in-depth.

The following sections provide guidelines on how the above defense-in-depth approach may be satisfied for various plant areas. Fault trees and associated probabilities were developed and used as described in Bases for Guidelines, Section 5.2 of this report, to evaluate the adequacy of these guidelines and to assure a consistent level of protection for the various areas.

5.1.1 General

All plants have overhead handling systems that are used to handle heavy loads in the area of the reactor vessel or spent fuel in the spent fuel pool. Additionally, loads may be handled in other areas where their accidental drop may damage safe shutdown systems. Accordingly, all plants should satisfy each of the following for handling heavy loads that could be brought in proximity to or over safe shutdown equipment or irradiated fuel in the spent fuel pool area and in containment (PWRs), in the reactor building (BWRs), and in other plant areas.

(1) Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviations from defined load paths should require written alternative procedures approved by the plant safety review committee.

(2) Procedures should be developed to cover load handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of this report. These procedures should include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe load path; and other special precautions.

(3) Crane operators should be trained, qualified and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, "Overhead and Gantry Cranes."

(4) Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or More for Nuclear Materials." This standard should apply to all special lifting devices which carry heavy loads in areas as defined above. For operating plants certain inspections and load tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane which will be used.* This is in lieu of the guideline in Section 3.2.1.1 of ANSI N14.6 which bases the stress design factor on only the weight (static load) of the load and of the intervening components of the special handling device.

(5) Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, "Slings." However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load.* The rating identified on the sling should be in terms of the "static load" which produces the maximum static and dynamic load. Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used.

(6) The crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, "Overhead and Gantry Cranes," with the exception that tests and inspections should be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency (e.g., the polar crane inside a PWR containment may only be used every 12 to 18 months during refueling operations, and is generally not accessible during power operation. ANSI B30.2, however, calls for certain inspections to be performed daily or monthly. For such cranes having limited usage, the inspections, tests, and maintenance should be performed prior to their use.)

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* For the purpose of selecting the proper sling, loads imposed by the SSE need not be included in the dynamic loads imposed on the sling or lifting device.

SER Open Item Nos. 62 and 63

Safety Evaluation Report, Open Items No. 62 (R.G. 1.58, Rev. 1) and No. 63 (R.G. 1.146) are associated with Generic letter 81-01, "Qualification of Inspection, Examination, and Testing and Audit Personnel", dated May 4, 1981. As requested in the Generic letter, LILCO commits to the following:

1. Regulatory positions C.5, 6, 7, 8, and 10 of Regulatory Guide 1.58, Rev. 1.

2. Regulatory Guide 1.146.

These commitments will be fully implemented by six months after fuel load.

II.K.3.28 VERIFY QUALIFICATION OF ACCUMULATORS ON AUTOMATIC DEPRESSURIZATION SYSTEM VALVES

The applicant should address the following issues as part of the qualification program for the ADS accumulators and the associated equipment.

1. The applicant should define the number of times the ADS valves must be capable of cycling using only the individual accumulator inventory and the length of time these accumulators are required to perform their function following an accident.

Response to Item 1:

This information was provided in our previous response to this item and in response to FSAR Question 212.97.

2. The criteria for the allowable leakage limits should be provided along with the bases for the criteria. The leakage limit should include some margin to account for a possible increase in leakage resulting from the effects of a harsh environment or a seismic event, unless it can be demonstrated that the leak rate will not increase following an accident. The bases for the allowable leakage criteria should include a discussion of the basis upon which the margin will be selected or justification for not applying a margin to the allowable leakage limit.

3. The applicant should commit to periodic leak testing of the ADS accumulator system to assure emergency supply for the required number and duration of valve actuations, as defined in Item 1 above.

4. The applicant shall propose technical specifications that specify leak test frequency, allowable leak rate, and the actions to be taken if the leakage limit is exceeded.

Response to Items 2, 3, and 4 :

In order to verify the leak-tight integrity of the ADS short term accumulator system, the applicant will leak test each redundant train at the same frequency as the plant integrated leak rate test program. This test should be performed such that the static head on the SRV's is not sufficient to unseat them. For train A, this test will be accomplished by closing valves 1P50*MOV-103A and its adjacent outboard manual globe valve, and 1P50*MOV-105A and venting the system piping between 1P50*MOV-103A and its associated outboard globe valve. With the minimum normal system pressure at 90 psig, the short term accumulator header, thereby isolated, will be required to maintain a pressure greater than or equal to 70 psig, as measured on header pressure indicator 1P50-PI-116A. This test will be performed as a function of time adequate to satisfy the short term ADS requirements. The train B header will be similarly tested.

If at the end of the short term accumulator header leak test, it is found tha 70 psig cannot be maintained, each ADS short term accumulator shall have its pressure checked locally to verify it is greater than or equal to 70 psig via a test gauge. Any short term accumulator system unable to maintain 70 psig will be repaired/modified, and retested to verify leak-tight integrity.

5. Since the ADS accumulator system is important to safety, it must meet the requirements of the GDC 2 and 4. The ADS accumulator system, and associated control circuitry, from the ADS valve operator out to and including the accumulator system isolation check valve should be seismically and environmentally qualified. Acceptable methods for demonstrating this qualification are given in SRP Sections 3.9.2, 3.10, and 3.11, as supplemented by the Category I requirements of NUREG-0588.

Response to Item 5:

The seismic and environmental qualification criteria for Shoreham are described in FSAR Sections 3.10 and 3.11. In addition, comprehensive description and status reports for both the seismic and environmental qualification program have been submitted by SNRC-575 dated May 28, 1981, and SNRC-576 dated May 27, 1981 respectively. As described therein, Class IE electrical equipment is qualified in accordance with NUREG-0588 Category 2.

The ADS accumulator system is qualified in accordance with the program outlined above and, therefore, meets the requirements of GDC 2 and 4.

6. The applicant will perform a leak test prior to initial operation. The applicant should address the action to be taken if the leakage rate during the pre-operational testing exceeds that established in Item 2 above.

Response to Item 6:

The applicant will perform a leak test prior to initial operation. Should the leakage rate exceed that established in Item 2, the system will be repaired/modified as required.

NUREG-0737 Item III.D.1.1 - Primary Coolant Sources Outside the Containment Structure

A leakage reduction and control program will be developed to monitor leakage from systems outside containment that could contain radioactive fluids during a serious transient or accident. The systems to be included in the immediate and continuing leakage reduction programs are:

Core Spray
HPCI
RHR
RCIC
Hydrogen Recombiners - Primary Containment Atmospheric Control
Reactor Water Clean-Up (RWCU)
Coolant Sampling and Post Accident Sampling
Reactor Building Equipment Drains (RBED)
Reactor Building Floor Drains
Instrumentation Lines
Reactor Building Standby Ventilation System (RBSVS)
Post-accident Monitoring System (PAMS)

The leakage reduction and control program will consist of regular periodic visual inspections together with detailed visual inspections and quantitative leakage measurements. Detailed system reviews will be conducted on a once per refuel cycle basis as outlined in Table III.D.1.1-1.

Detailed system walkdowns will be conducted of liquid systems and quantitative leakage measurements will be performed on a once per refueling outage cycle. General visual inspections for leakage or signs of leakage will be conducted on a more frequent basis by operations personnel on accessible portions of the systems. These inspections will be conducted with the systems pressurized to normal or test modes. Any excessive leakage will be recorded and appropriate maintenance work requests generated.

Systems containing gases are to be tested by use of tracer gases (helium, freon, or DOP), by pressure decay testing or metered makeup tests.

All maintenance work requests written to correct or investigate leakage on systems included in the program will be marked as "LEAKAGE RELATED". This will alert maintenance personnel to assign high priority to this work and will flag these requests for analysis and record keeping.

The Technical Support Group will administer the leakage reduction and control program. Responsibilities will include the following:

a. Coordinate the once per refueling outage detailed inspections and quantitative leakage measurements, and maintain records generated from the inspection.

b. Review all leakage related maintenance work requests and maintain a record of status for equipment exhibiting leakage. This review will also consider initiating modifications to reduce system leakage.

c. Maintain records and provide a written report annually to the Plant Manager on the following:

   1) Actual system leakage rates as determined by the once per fuel cycle measurements for each system in the program.

   2) Analysis of the data indicating the reason for any high value and identifying corrective action to be taken.

   3) Status of pending leakage related maintenance work requests or modifications.

The Shoreham leakage reduction and control program will be implemented prior to fuel load. Before beginning full power operation, LILCO will submit to the NRC staff a report of actual measured leakage rates from all systems included in the program and all preventative maintenance performed as a direct result of the evaluation of this leakage. The report will also identify leakage criteria to be applied during the first fuel cycle as the basis for instituting corrective action in the form of preventative maintenance. Prior to the start of the second fuel cycle, LILCO will revise the criteria as necessary based on the experience gained during Shoreham's first fuel cycle. The revised criteria shall then be used as the basis for long term leakage monitoring activity at Shoreham.

The following systems which may contain radioactive fluids during a serious transient or accident were not included in the Shoreham leakage reduction and control program for the reasons indicated:

-Suppression Pool Cleanup Mode of Fuel Pool Cleanup System - source lines from suppression pool automatically isolate upon accident conditions.

-Drywell Equipment and Floor Drains - system isolates automatically upon accident conditions.

-Drywell Air Cooling System - system isolates automatically upon accident conditions.

-RBNVS - system secures upon accident conditions; reactor building ventilation handled by RBSVS which is included in leakage reduction and control program.

In regard to the North Anna and related incidents described in the NRC letter dated 10/17/79, LILCO is following the recommendations outlined in I&E Circular No. 79-21, "Prevention of Unplanned Releases of Radioactivity". LILCO is taking the following preventative measures:

1. Review of Operating Procedures involving transfer of radioactive liquids.

2. Review of "as-built" systems having the potential of inadvertent releases

3. Review of Surveillance Procedures to address testing of systems and temporary piping which could cause an inadvertent release.

4. Include I&E Circular 79-21 on Required Reading List.

TABLE III.D.1.1-1

SUMMARY OF SNPS-1 LEAKAGE REDUCTION & CONTROL PROGRAM

SYSTEM	TEST METHOD	FREQUENCY	RESPONSIBLE GROUP
1. Core Spray	Detailed Inspection and Measurement	Once per refueling cycle	Technical Support
	Visual (Note 1)	During Operability Surveillance	Operations
2. HPCS	Detailed Inspection and Measurement	Once per refueling cycle	
	Visual (note 1)	During Operability Surveillance	Operations
3. RHR	Detailed Inspection and Measurement	Once per refueling cycle	
	Visual (Note 1)	During Operability Surveillance	operations
4. RCIC	Detailed Inspection	Once per refueling cycle	
	Visual (Note 1)	During Operability Surveillance	Operations
5. Primary Containment Atmosphere Control - Hydrogen Recombiners	Metered Air Make-up	Once per refueling cycle	
6. RWCU	Leakage Inspection and Measurement	Once per refueling cycle	
7. Coolant Sampling & Post Accident Sampling	Leakage Inspection and Measurement	Once per refueling cycle	
	Visual (Note 1)	When samples are drawn	Radiation/Chemistry
8. RBED	Meter for abnormal input rate	Daily	Operations
9. Reactor Building Standby Ventilation [illegible]	DOP and Freon Testing (Note 2)	Once per refueling cycle	
10. PAMS	Leakage Inspection and Measurement	Once per refueling cycle	Technical Support Group

NOTE 1: Accessible portions only
2: Filters only