LILCO, March 23, 1983

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DOCKETING & SERVICE BRANCH

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 (OL)
	)	
(Shoreham Nuclear Power Station, Unit 1)	)	

LILCO'S RESPONSE TO "SUFFOLK COUNTY MOTION TO STRIKE PORTIONS OF THE NRC STAFF'S PROPOSED OPINION AND FINDINGS OF FACT, LILCO'S PROPOSED OPINION AND FINDINGS OF FACT, AND LILCO'S REPLY TO THE PROPOSED OPINION AND FINDINGS OF SUFFOLK COUNTY AND THE STAFF"

In a motion served by mail upon the parties on March 8, 1983, Suffolk County has moved to strike three of LILCO's 743 initial Proposed Findings of Fact, three of the Staff's 421 Proposed Findings and their related discussion in the Staff's Proposed Opinion, and certain portions of LILCO's Reply to the Staff's and to SC's Proposed Opinion and Findings of Fact. The motion is based on the premise that each "refers to, and has the Board draw conclusions based upon data that are not in evidence on the record of this proceeding." SC Motion at 1.

Although SC's motion to strike appears to identify several distinct items, the findings at issue involve in actuality only five SNRC letters1/ relating to two contentions (Safety Relief Valves, SC-22; and Mark II, SC-21) and miscellaneous correspondence relating to definitional issues between LILCO and the

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1/ Only one item that SC moves to strike, Note 37 in LILCO's Reply, does not involve correspondence between LILCO and the Staff. Except for Note 6, below, it is not otherwise specifically addressed in this response.

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Staff on Contention 7B. Each of these items is a public document and was served on the Board and Suffolk County at the time it was transmitted to the Staff. The following table clarifies the relationships of the submittals discussed in the SC motion:

Reference at Issue	Discussion in LILCO Initial Findings	Discussion in SC Findings	Discussion in Staff Findings	Discussion in LILCO Reply Findings
		SRVs (SC-22)		
SNRC-812	H-23	--	22:28/20, 22:28/22 & Opinion at 29	Vol. 1 at 212,220
SNRC-816	H-37	--	22:28/38, & Opinion at 30	Vol. 1 at 218
		Mark II (SC-21)		
SNRC-808/ SNRC 824	G-14	--	--	Vol. 1 at 196-97 and Note 38
SNRC-831 2/	[G-10]	--	--	Vol. 1 at 203 (Note 39)
		SC-7B		
Two Letters Between LILCO & Staff	--	--	Vol. 2 at 36 (Note 13)	--

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2/ SC objects to LILCO's reference to the completion of some confirmatory piping analyses which were to be transmitted to the Staff. At the time of LILCO's Reply, there had been no submittal; thus the SNRC letter was not referenced in LILCO's Reply. Subsequent to the Reply, these reanalyses were sent to the Staff in SNRC-831. The analyses, although not the submittal, were also discussed in LILCO Finding G-10, which SC does not move to strike.

As the table illustrates on its face, the correspondence actually at issue is small -- five SNRC letters3/ and two letters between LILCO and the NRC Staff not referred to in LILCO's findings.4/ Further, it shows that of the items addressed by LILCO (SNRC-812 and -816 on SRVs, SNRC-808, -824 and -831 on Mark II), all were explicitly cited or referred to in LILCO's Proposed Findings. Since SNRC-824 and -831 were not written until after the initial filings, they could not have been referred to directly until the reply findings; however, as is shown below, their contents were presaged in the initial findings.

SC's motion should be rejected for both procedural and substantive reasons. It is overreaching,5/ untimely, and procedurally improper. It is also ill-taken on the merits, in that it presumes the use by LILCO of the cited documents for the truth of their contents, whereas the use actually made by LILCO of the correspondence at issue in its findings is not to

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3/ Copies of each of the SNRC letters are attached for the Board's convenience.

4/ LILCO does not here speak to the reference to the letters between LILCO and the Staff referred to in Staff Proposed Findings, Vol. 2 at 36 (Note 13), though LILCO agrees with the Staff's characterization there of the correspondence at issue.

5/ The motion typically asks for the striking of entire findings, when examination of the findings reveals that only one sentence, at most, in each finding is potentially affected by the correspondence at issue. For the Board's convenience, copies of the pertinent LILCO initial and reply findings, with affected areas indicated, are attached.

establish substantive propositions for which there is no other support in the record. Rather, their purpose is to enable the Board to take official notice of the existence of confirmatory documents which reflect analyses not yet finally written up or commitments still under consideration at the time the issues were being litigated.6/

A. The Motion is Untimely, Improper and Prejudicial

SC's motion was filed fully 50 days after LILCO's January 19, 1983 initial Proposed Findings of Fact, in which every document used by LILCO and now being complained of was either cited directly or presaged. The facial untimeliness of the motion is exacerbated by the fact that SC filed its own proposed findings of fact in the meantime, on January 31, 1983, and in them failed to object to any of the items of correspondence now complained of. That responsive filing, or a contemporaneous motion to strike, was the proper vehicle contemplated by the Commission's Rules of Practice, 10 CFR § 2.754, for SC's objection; and had SC used it, LILCO would have been on timely notice of a disagreement over the use of a relatively small class of documents and could either have provided

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6/ In the case of SC's objection to footnote 37 on page 195 of LILCO's Reply, the text of the Reply clearly demonstrates that qualification of the strengthened vacuum breaker disc was not a prerequisite for licensing the plant. The note was included simply to inform the Board and all parties that this effort was complete and that results would soon be submitted to the Staff.

further authentication of them in its reply findings or clarified their use. By lying in wait until after LILCO had filed its reply findings (and a rather leisurely wait at that, given that those findings were filed on February 22), SC has not only defaulted without any showing of good cause on its proper opportunity to object to the use of the documents, but has done so in a manner prejudicial to LILCO, since LILCO relied in its reply findings on the apparent absence of dissent over the use of documents referenced in its initial findings. SC has waived its proper opportunity under the Rules of Practice to object to the use of any of these documents and has not attempted any showing of good cause therefor; its raising of this matter now is prejudicial to LILCO; and the motion should be denied on this ground.

B. <u>The Letters Are Not Given Improper Substantive Weight</u>

Examination of LILCO's use of the documents now complained of in its initial and reply findings reveals clearly that in each case the document either contained a confirmatory analysis of matters discussed on the record in hearings on the SRV or Mark II issues, or memorialized a licensing commitment stated at the hearing to be under consideration. These documents are not cited for the truth of assertions not otherwise supported in the record, but for their existence. This is exactly the purpose for which official-notice provisions exist, <u>see</u> 10 CFR § 2.743(i). SC's motion suggests baldly, without reference

to any examples, that the information in them is not suitable for notice. SC is clearly incorrect: there is a substantial body of decisional law supporting the proposition that administrative agencies can and should take notice of official correspondence with regulatees.7/ It is not necessary to evaluate, much less rely on, the substance of the SNRC letters now at issue to see from LILCO's initial and reply findings that the letters merely confirm the closing-out of confirmatory analyses promised, or confirm a commitment under consideration, when the issues were being tried.

Taking the SRV-related correspondence first, SNRC-812 merely transmits to the Staff confirmatory analyses of pipe stresses in the alternate shutdown mode; the content of the preliminary analyses had already been accepted by the Staff and been the subject of litigation. See LILCO Initial Finding H-23; Reply Findings Vol. 1, at 212, 220. SNRC-816 merely transmits to the Staff a commitment to implement a lowered MSIV setpoint at the first refueling outage -- a commitment for which credit was not taken in LILCO's analyses, and which had been under consideration by LILCO at the time of the hearings. See LILCO Initial Finding H-37; Reply Findings Vol. 1 at 218.8/

7/ The U.S. Court of Appeals for the D.C. Circuit has held that a regulatory agency can and should take official notice of the reports filed with it by a regulated company. Wisconsin v. F.P.C., 201 F.2d 183, 186 (1952), cert. denied 345 U.S. 934 (1953); see Market Street R. Co. v. Commissioner, 324 U.S. 548, 562 (1944); see Midwest Television, Inc. v. F.C.C, 426 F.2d 1222, 1229 (1970).

8/ It is curious that SC objects to the Board's taking official notice of this commitment, in view of its Proposed Finding 22:48, which proposed that this change be required at Shoreham as a license condition.

As to Mark II, SNRC-808 and -824 transmit to the Staff responses to the so-called Humphrey concerns which confirm testimony of Staff witness Fields at the hearing, and in any event commit to a mode of operation conceded at the hearing to obviate the Humphrey concerns until final resolution of them. See LILCO Initial Finding G-14; Reply Findings Vol. 1 at 196-97 and Note 38. SNRC-831 merely transmits to the Staff the results of a reanalysis of piping systems at three levels on the containment -- a reanalysis which the Staff had clearly identified as confirmatory. See LILCO Initial Finding G-10; Reply Findings Vol. 1 at 202-03.

The Board does not need to evaluate the substance of any analyses forwarded by these letters to give them their proper weight. In both their uses, the letters are merely confirmatory: analyses merely confirm matters already fully litigated on the record, and licensing commitments obviate what would have otherwise been areas of technical dispute on the record.

While no further authentication of them is needed, the affidavit of Jeffrey L. Smith attesting to the truth and correctness of the SNRC letters at issue is attached out of an abundance of caution. The Board can permit the noticed use of the SNRC letters at issue without necessarily receiving the affidavit into evidence. See Carolina Power & Light Company (Shearon Harris Nuclear Power Plant, Units 1, 2, 3 and 4), LBP-78-2, 7 NRC 83, 84-88 (1978), and discussion in LILCO's Reply Findings Vol. 1 at 11-13.

If SC thinks that the Board's taking notice of these SNRC letters is so fundamentally important to the issues in the SRV and Mark II contentions that the outcome of the issues would thereby be affected, and can show good cause for having failed to file a timely objection, it can, of course, seek to reopen the record, if it can meet the test for reopening the record, Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), ALAB-138, 6 AEC 520, 523 (1973); Duke Power Company (William B. McGuire Nuclear Station, Units 1 and 2), ALAB-669, 15 NRC 453, 465 (1982). Even then, the Board has discretion in the procedures it will follow in reopening the record. Carolina Power & Light Company, supra.

CONCLUSION

For the reasons stated above, SC's Motion to Strike, dated March 8, 1983, should be denied.

Respectfully submitted,

Donald P. Irwin

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One of Counsel for Long Island Lighting Company

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DATED: March 23, 1983

Attachments: Excerpts from Findings
Affidavit of Jeffrey L. Smith
(and attachments thereto)

CERTIFICATE OF SERVICE

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

I hereby certify that copies of LILCO's Response to Suffolk County Motion to Strike Portions of the NRC Staff's Proposed Opinion and Findings of Fact, LILCO's Proposed Opinion and Findings of Fact, and LILCO's Reply to the Proposed Opinion and Findings of Suffolk County and the Staff were served this date upon the following by first-class mail, postage prepaid.

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DATED: March 23, 1983

G-10. Subsequent to issuance of Revision 5 to the Design Assessment Report, the Staff also expressed an interest in learning more about the basis for the selection of the 30 piping subsystems that were analyzed to evaluate the effects of the NUREG-0808 loads on the response of the plant's reactor building. Tr. 9886 (Terao); Suffolk County Ex. 45. In particular, the Staff was concerned with three locations on the containment wall. LILCO has committed to do a 100% reevaluation of all large bore and certain small bore piping systems at these three elevations before fuel load. Tr. 9888 (Terao). The Staff regards this reevaluation as confirmatory only, since no piping system stresses or support loads have been shown to exceed the code allowables. Tr. 9889 (Terao).

The Humphrey Concerns

G-11. In a letter to the Applicant dated July 8, 1982, the Staff identified 22 concerns raised by a Mr. John Humphrey regarding the adequacy of the design margins of the Mark II containment system; these concerns were potentially applicable to Shoreham. Suffolk County Ex. 44; Tr. 9849-50 (Fields).

G-12. In July 1982, the NRC Staff made a presentation to the ACRS on the Humphrey concerns for all BWRs, and the ACRS concluded that Humphrey's concerns did not appear significant. Tr. 9855 (Fields).

G-13. Based on the ACRS's conclusion, the Staff indicate[illegible]at it will evaluate the Applicant's responses on a confirmatory basis only. Accordingly, the Staff did not see a need to delay issuance of an operating license to await the completion of the review. Tr. 10005-06 (Eltawila).

G-14. LILCO submitted its preliminary responses on the Humphrey concerns to the Staff on August 25, 1982 and its final responses in early December. All but two of the issues were addressed in that report. The two remaining responses that relate to the RHR heat exchanger relief valve discharge lines will be submitted to the Staff in January 1983. Those responses may involve a commitment by LILCO not to use the RHR steam condensing mode during normal plant operation until it can be demonstrated that the hydrodynamic loads resulting from operation of the RHR heat exchanger in this mode are acceptable.

Steam Bypass Testing

G-15. Both preoperational and periodic tests are performed to detect any leak paths that could exist between the drywell and the wetwell. Tr. 9864 (Fields). The leakage rate is then compared with the appropriate acceptance criterion to determine acceptability. Eltawila et al., ff. Tr. 9741, at 9.

Specifically, the Staff wondered how loads measured using a "rams head" discharge pipe configuration at the test facility were translatable to Shoreham, which utilizes a "tee quencher" at the end of the discharge line. SC Ex. 34, ff. Tr. 8312, at Question 1. LILCO replied that the loads measured at the test facility would be larger than the loads on the valve internals at Shoreham, since (1) no dynamic mechanical load generated at the tee quencher is transmitted to the SRV because there is at least one anchor point between the tee quencher and the valve, (2) the first length of piping downstream of the SRV in the test program was conservatively set at twice that at Shoreham to bound the dynamic mechanical load on the valve, and (3) backpressure loads were maximized at the test facility through the use of an orifice plate and conservative pipe lengths. LILCO Response, ff. Tr. 8402, at 2-4. The Staff reviewed this response and concluded that the test facility imposed greater loads on the valves than those expected at Shoreham. Tr. 8407-08 (Wright).

H-23. The Staff noted that the test facility did not utilize spring hangers as pipe supports, but that they are used in conjunction with snubbers and rigid suports at Shoreham. The Staff asked LILCO to describe the supports used at Shoreham and to explain how the loads measured at the test facility may change given the different support configurations at Shoreham. SC Ex. 34, ff. Tr. 8312, at Question 2. LILCO responded that

the location of snubbers and rigid supports at Shoreham demonstrate that the generic test facility was prototypical, LILCO Response, ff. Tr. 8402, at 5. In addition to the supports, each SRV discharge line has one or two spring hangers, all of which are located in the drywell. Id. Since the dynamic loads resulting from liquid discharge during the alternate shutdown cooling mode of operation are significantly lower than those from high pressure steam discharge, sufficient margins should exist in the Shoreham piping system design to adequately offset the increased dead load on the spring hangers in an unpinned condition due to a water-filled condition. Id. 6-7. Nevertheless, LILCO committed to provide additional analyses on all pipes for the alternate shutdown mode, Tr. 8421 (Smith). The Staff accepted LILCO's response subject to the confirmational analyses. Tr. 8410 (Cherny). The results of these analyses, submitted in SNRC-812 on December 15, 1982, demonstrate that Shoreham complies with the requirements of NUREG-0737, Item II.D.1.

H-24. The Staff also questioned whether the flow conditions tested at the generic facility were similar to those anticipated at Shoreham. SC Ex. 34, ff. Tr. 8312, at Question 4. LILCO's response indicated that the fluid conditions of the test program matched those expected to occur at Shoreham when the plant was being operated in the alternate shutdown cooling mode. As for other event sequences, LILCO first identified

events and displayed various of them, from which BWR owners could select those best adapted to their circumstances. Id. However, most other possible modifications listed in Item II.K.3.16 either provided little relative benefit (5% or less) or tended to have adverse side-effects on plant safety. Boseman et al., ff. Tr. 7959, at 11.

H-36. LILCO evaluated the possible modifications listed in Item II.K.3.16 and studied in the GE SRV Study, and then applied the results of that study to Shoreham. Tr. 8617-22 (Smith). LILCO utilized the two principal methods evaluated in the GE SRV Study for reducing SORV events at Shoreham: (1) the use of the Target Rock two-stage SRV rather than the three-stage SRV used in the benchmark plant, and (2) use of Emergency Procedure Guidelines that permit manual implementation of a low-low set relief. The combination of the two was shown to reduce the failure of SORV events by approximately an order of magnitude. Boseman et al., ff. Tr. 7959, at 10-11; Board Findings H-32 to -35.

H-37. In addition to these features, LILCO has implemented SORV reduction measures not evaluated in the GE SRV Study, for which no specific credit has been claimed: a lowered recycling set point, worth about 1% improvement, Tr. 8655 (Hayes), and pneumatic control system modification, worth about 2% to 3%, Tr. 8656-57 (Hayes, Smith). LILCO witnesses also

indicated that they were continuing to evaluate the advantages and disadvantages of yet further potential modifications, such as lowering of the MSIV closure setpoint. Tr. 8629-31 (Smith, Hodges); Tr. 8672-73 (Smith). LILCO indicated to the Staff recently that LILCO intended to implement the lowered MSIV setpoint at the first refueling outage (Letter, Smith (LILCO) to Denton (NRC), January 7, 1983 (SNRC-816)).

H-38. SC contended that the SORV frequency reduction factors asserted by LILCO were too large. Bridenbaugh and Minor (Challenges), ff. Tr. 8709, at 5. However, the basis for SC's assertion -- a report by Southwest Research Institute -- relies solely on Target Rock three-stage valves for its reliability analysis. Tr. 8763 (Bridenbaugh); LILCO Ex. 18, ff. Tr. 9299, at 20, A-9/A-10. In addition, SC neglected a second 50% reduction factor attributable to reduction in spurious blowdown (see Board Finding H-34). SC presented no reliability surveys on two-stage SRV performance.

H-39. The Staff's prefiled direct testimony agreed that LILCO had reduced the frequency of SRV challenges, contrary to the allegations of SC, by a combination of system modifications and procedural techniques. Hodges, ff. Tr. 7966, at 3. On cross-examination, Mr. Hodges confirmed that he had been the author of NUREG-0737, Item II.K.3.16 (Tr. 8022, 8488, 8491-92); that the notion there of "reduction in challenges" to

LILCO submitted its responses to the Humphrey concerns in SNRC-808 (December 9, 1982) and SNRC-824 (January 28, 1983).38/ These responses confirmed Staff witness Fields' views on these concerns, namely, that a vast majority of the concerns were not significant, and that the only area of potential significance may involve the RHR heat exchanger when operated in the steam condensing mode. See Tr. 9855 (Fields). As to the RHR heat exchanger, LILCO has reported that its complex analysis of this matter is still in progress, and that, pending its completion, it has decided to proceed in the manner of the Grand Gulf Station, see Tr. 9855 (Fields), committing not to use the RHR heat exchanger in the steam condensing mode

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38/ SNRC-824 was submitted to the Staff after the submission of LILCO's proposed opinion and findings of fact in this proceeding. The information contained in SNRC-824 updated the information contained in LILCO Finding G-14. Accordingly, LILCO requests that that proposed finding be replaced with the following finding:

> G-14. LILCO submitted its preliminary responses on the Humphrey concerns to the Staff on August 25, 1982 and its final responses in early December 1982 (SNRC-808) and late January 1983 (SNRC-824). Those responses contain a commitment not to use the RHR steam condensing mode during normal plant operation until it can be demonstrated that the hydrodynamic loads resulting from operation of the RHR heat exchanger in this mode are acceptable.

for all normal plant operations. SNRC-8:4; see LILCO Finding G-14. LILCO has made this commitment with the understanding that it may conduct an engineering effort in the future to demonstrate the ability of the RHR system to withstand loads while operating in the steam condensing mode. Id. Thus, since LILCO has implemented the Staff's recommended interim solution to the RHR problem, there is no basis in the record for SC's request for submittal of Humphrey responses, LILCO Finding 21:80, and approval of any further actions should rest with the Staff.

2. Shoreham Confirmatory Analysis

Subsection (d) of SC Contention 21 questions whether the Shoreham Mark II Containment has been demonstrated to withstand loads from simultaneous design basis transient and LOCA events. LILCO and the Staff agree that LILCO has demonstrated the design adequacy of the Shoreham Containment. LILCO Finding G-21; Staff Finding 21:23. SC contends that the Staff's review of LILCO's assessment is not complete and that, therefore, the Staff's testimony should be given little weight. SC Findings 21:24-36. In particular, SC alleges that the Staff has failed to complete its review of LILCO's confirmatory piping analyses at three elevations on the containment wall and that LILCO has

Accordingly, LILCO undertook a program to evaluate the significance of the local exceedances. Id. This evaluation included structures, piping, and other components. The results of the evaluation, documented in the DAR, indicated no exceedances of design stress allowables. Tr. 9974-75 (Malovrh). Thus, SC's request that the Board require LILCO to explain and justify the differences in the ARS, SC Finding 21:80, finds no support in the record.

c. Further Piping Subsystem Analyses

Finally, SC contends that the Staff believed LILCO's piping assessment was inadequate and thus required LILCO to complete a 100% evaluation of piping subsystems at three locations on the containment wall. SC Findings 21:70-71. SC further argues that completion of this evaluation is a prerequisite for resolving this contention. SC Finding 21:72. SC's argument misconstrues the testimony in this proceeding and finds no basis in the record. LILCO committed to perform a 100% reevaluation of all large bore and certain small bore piping systems in order to allay Staff concerns about whether earlier piping analyses were representative. LILCO Finding G-10; Tr. 9887-89 (Terao, Malovrh). The Staff regarded the

reevaluation as confirmatory, since no piping system stresses or support loads have been shown to exceed ASME code allowables. Id. The confirmatory analysis undertaken by LILCO is sufficiently circumscribed to be left to Staff review.39/

3. Steam Bypass Testing

The focus of subsection (c) of SC Contention 21 is on the ability of LILCO's test procedures to demonstrate an acceptable steam bypass rate of the drywell floor seal and downcomer vacuum breakers. SC has proposed no findings that suggest that LILCO's test procedures will fail to demonstrate the adequacy of the leakage rate. Instead, SC has reached beyond the scope of the contention, which addresses only the testing procedures and not the test results produced from the application of those procedures, to suggest that the high pressure test results may not be valid since the Staff did not verify the validity of LILCO's testing. SC Finding 21:39. Even if the Staff's review were within the scope of the contention, SC's allegation has no factual basis. As Staff witness Fields

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39/ The reevaluation, which has now been completed and will soon be forwarded to the Staff, confirms that no code allowables have been exceeded.

for LILCO's conclusions. See LILCO Findings H-15, -16, and -22 to -25. SC did not quarrel at the time with the appropriateness of the meeting, and in fact attended it. Tr. 8399-8402. LILCO's responses to the six questions were submitted the following morning, LILCO Finding H-4, and were reviewed by the Staff for approximately the same period of time normally spent on similar issues. Tr. 8608 (Wright, Hodges, Cherny). LILCO and Staff witnesses were then subjected to extensive questioning by the Board and SC on this material, including questions regarding the Staff's reason for posing each question, the form of response expected by the Staff, and the sufficiency of LILCO's response. See Tr. 8404-42, 8557-8611. Based on all these procedures the Staff concluded that LILCO had complied with the requirements of NUREG-0737, Item II.D.1. LILCO Findings H-15, -16, and -22 to -25; Staff Findings 22/28:19-27. Accordingly, the Staff's review was neither incomplete nor imprecise despite SC's allegations (SC Proposed Finding 22:18). The confirmatory piping analysis undertaken by LILCO at that time is sufficiently circumscribed to be left to the Staff's review (SC Proposed Finding 22:19 to the contrary notwithstanding) and has already been submitted to the Staff in SNRC-812 dated December 15, 1982. LILCO Finding H-23.

II.K.3.16 -- namely, to reduce SORV events. As the Staff witnesses noted, Item II.K.3.16 was designed to create a meaningful goal: a goal not defined simply by numbers, but by a desire to have all reasonable modifications identified and implemented. Staff Finding 22/28:37. Judged against this standard, LILCO complied with the requirements of Item II.K.3.16. LILCO Findings H-39 and -40; Staff Finding 22/28:40.

In an effort to assure that all reasonable modifications have been implemented, LILCO has continued to assess the efficacy and practicality of changing the water level set point for MSIV closure. <u>See</u> Tr. 8628-29 (Smith). As noted in LILCO's proposed findings, that review is now complete and LILCO has committed to implement this modification at the first refueling outage. LILCO Finding H-37; Staff Finding 22/28:38. Thus, to the extent SC's proposed findings suggest that the change in the MSIV closure set point is an idea that LILCO is not pursuing, and thus, that all reasonable modifications are not being implemented, <u>see</u> SC Findings 22/28:42 and :47, they are simply inaccurate.

SC also appears to disagree with LILCO's and the Staff's description of the performance of 2-stage Target Rock

REPLY TO SC SRV FINDINGS

While LILCO's principal reply on SC Contention 22 and SC Contention 28(a)(vi)/SOC Contention 7A(6) addresses LILCO's primary concerns with SC's proposed findings, it does not attempt to provide a finding-by-finding refutation of SC's proposed findings. Nor is that purpose of this section, which simply addresses a small group of findings, not specifically addressed in the principal LILCO reply, but which nevertheless do not accurately reflect the record. This appendix is not meant to be all-inclusive, and the fact that any given finding is not addressed here does not imply LILCO's agreement with it.

22:16. LILCO did not "acknowledge the need to perform stress analyses." Instead, LILCO committed to perform confirmatory analyses, even though it felt there was sufficient information to support its conclusions. LILCO Finding H-23. The results of those analyses, submitted in SNRC-812, support LILCO's earlier conclusion. Id.

22:17. The majority of this finding ignores the fact that the amount of time the Staff spent reviewing these responses was approximately the same as that spent actually reviewing similar submittals. See page 212 above.

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 (OL)
	)	
(Shoreham Nuclear Power Station, Unit 1)	)	

AFFIDAVIT OF JEFFREY L. SMITH

Jeffrey L. Smith, being duly sworn, deposes and says as follows:

1. My name is Jeffrey L. Smith, Manager of Special Projects for Shoreham Nuclear Power Station, Unit 1. SNRC Letters 808, 812, 816, 824, and 831 (copies of which are attached) were prepared under my supervision and direction.

2. I hereby solemnly swear and affirm that the contents of the SNRC letters referred to in paragraph one (1) above are true and correct to the best of my knowledge and belief.

Jeffrey L. Smith
Jeffrey L. Smith

STATE OF NEW YORK,
COUNTY OF SUFFOLK

Subscribed to and sworn before me this 22nd day of March, 1983.

Nancy J. Schmitt
Notary Public

My Commission Expires: 3/30/84

NANCY J. SCHMITT
NOTARY PUBLIC, State of New York
No. 52-8826330, Suffolk County
Term Expires March 30, 1984

LILCO

LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION

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

December 9, 1982 SNRC-808

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

Concerns Regarding the Adequacy of the Design Margins
of the Mark I and II Containment Systems
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Reference (1): Robert L. Tedesco letter to
M. S. Pollock dated July 8, 1982

Dear Mr. Denton:

Reference (1) requested that the Long Island Lighting Company provide a proposed program to respond to the subject concerns that were identified as being potentially applicable to Shoreham.

In response to this request, enclosed please find forty (40) copies of a report entitled "Concerns Regarding the Adequacy of the Design Margins of the Mark II Containment Systems at Shoreham Nuclear Power Station." This report addresses all items in reference (1) with the exception of items 3.3 and 3.4. A submittal will be made on these items by mid-January, 1983.

Should you have any questions, please contact this office.

Very truly yours,

J. L. Smith
Manager, Special Projects
Shoreham Nuclear Power Station

RWG:jm

Enclosure

c.c.: J. Higgins
All Parties

LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION

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

Direct Dial Number

December 15, 1982

SNRC-812

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

SER Issue II.D.1 - SRV Test Program
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Reference: (1) Letter NRC (Mr. A. Schwencer) to LILCO (Mr. M. S. Pollock) dated July 8, 1982

(2) Letter from BWR Owners' Group (T. Dente) to NRC (D. G. Eisenhut) dated 9/25/81

Dear Mr. Denton:

The reference (1) letter forwarded a request for additional information consisting of six (6) questions on the Safety Relief Valve (SRV) Operability Test Program Results, and their applicability to the Shoreham SRVs. The SRV test results have been documented in report NEDE-24988-P, Analysis of Generic BWR Safety/Relief Valve Operability Test Results which was forwarded to the staff via the Reference 2 letter.

A response was provided for each of these six questions in a submittal filed with the ASLB on July 29, 1982 (refer to Attachment 1). It was determined, however, that the staff required supplemental information for question 2, involving performance of a stress analysis for each SRV discharge line, and question 4, involving a description of the events and anticipated conditions at Shoreham for which the valves are required to operate and a comparison of these plant conditions to the conditions in the test program.

As you are aware, question 2 addressed the issue of how the Shoreham unique SRV discharge piping supports may affect conclusions regarding SRV operability derived from the generic test facility.

The generic test facility was designed to be prototypical of BWR plants in terms of discharge piping configuration. It was concluded in the generic test program that the fluid transient line forces resulting from the alternate shutdown cooling mode liquid discharge are of substantially lower magnitude than those resulting from the design basis high pressure steam discharge events. On this basis it could be concluded that rigid pipe supports and snubbers which would carry the direct fluid transient loads are adequate for the liquid discharge event, since they have been designed for the more severe steam discharge events. Since spring hangers are affected most by static loads (deadweight and thermal loads), the effects of the added weight of water in the lines need to be evaluated for such supports.

In order to fully evaluate the adequacy of Shoreham's SRV discharge piping supports to assure no potentially adverse effects on SRV operability, fluid flow transient analyses as well as pipe stress and support analyses have been performed for the alternate shutdown cooling mode liquid discharge in Shoreham. The dynamic fluid forces calculated for each of the eleven discharge lines exhibited the same general characteristics as observed in the test facility; particularly, magnitudes of forces were found to be lower than those resulting from high pressure steam discharge by ratios similar to those found in the test. The eleven lines in the dry well (each of which has one or two spring hangers, as well as rigid supports and snubbers) were then analyzed using standard techniques to determine the effects of the dynamic fluid forces. These lines were also analyzed to determine pipe stresses and support loads due to the deadweight of the water in the lines, the concurrent thermal effects, and also for the effects of an assumed concurrent safe-shutdown earthquake.

All piping stresses calculated for the combination of loads described above were found to be well within ASME Faulted condition allowables. Each pipe support was also found to be within design allowables for the same combination of loads. It is noted that the spring hanger supports, which were potentially of most concern, had been designed to carry the full weight of water associated with the hydro test condition (during which time the springs are pinned). For the condition discussed herein, the spring hanger travel distances were also checked and found to be within the working range of the springs, assuring that they will not bottom out during this event.

None of the eleven Shoreham discharge lines have any spring hangers in the wet well. Additionally, since the lines are anchored at the dry well floor, loads imposed on the lines in

this area are in no way transmitted to the SRVs. However, the wet well line judged to be most likely to be strongly affected (based on support locations) was also analyzed for the same loading conditions. Again, all pipe stresses and support loads were well within design allowables. Even though it is clear that on this basis, there is no outstanding concern in this area, for completeness the remaining ten lines in the wet well are also being analyzed for these conditions. This final verification analysis will be completed prior to fuel load.

Based on the detailed analytical evaluation described above, it is concluded that the Shoreham SRV discharge piping is adequately supported to sustain the effects of a low pressure liquid discharge. Since all pipe supports are adequate and all pipe stresses are within allowable levels, the loads on the valves will not adversely affect operability of the Shoreham SRVs.

With regard to question 4, an amplified response, completely responsive to the NRC question, is included as Attachment 2 to this letter.

The information contained herein should be sufficient to allow the staff to completely close this item on the Shoreham docket.

Should you have any questions, please contact this office.

Very truly yours,

Original signed by
J. L. Smith
Manager, Special Projects
Shoreham Nuclear Power Station

RWG/law

Enclosures

cc: J. Higgins
All Parties

Attachment 1

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 (OL)
	)	
(Shoreham Nuclear Power Station, Unit 1)	)	
	)	

RESPONSE OF LONG ISLAND LIGHTING
COMPANY TO NRC REGULATORY STAFF QUESTIONS
OF JULY 8, 1982 RELATIVE TO SRV TESTING

Long Island Lighting Company has received a letter, dated July 8, 1982, from the NRC Regulatory Staff involving six questions relating to the testing of Safety Relief Valves for the Shoreham Station. The covering letter, as amplified by the oral testimony of Regulatory Staff witnesses, indicated that the Staff felt that it needed more information in the area described in the six questions attached to the letter in order to complete its review of SRV testing for Shoreham. The following submittal contains LILCO's response to the six questions.

Respectfully Submitted

LONG ISLAND LIGHTING COMPANY

Donald P. Irwin

Donald P. Irwin

Hunton & Williams
Post Office Box 1535
707 East Main Street
Richmond, Virginia 23219

DATED: July 29, 1982

1. Q. The test program utilized a "rams head" discharge pipe configuration. Shoreham utilizes a "tee" quencher configuration at the end of the discharge line. Describe the discharge pipe configuration used at Shoreham and compare the anticipated loads on valve internals in the Shoreham configuration to the measured loads in the test program. Discuss the impact of any differances in loads on valve operability.

A. The safety/relief valve discharge piping configuration at Shoreham utilizes a "tee" quencher at the discharge pipe exit. The average length of the 11 SRV discharge lines (SRVDL) is 137' and the submergence length in the suppression pool is approximately 13'. The SRV test program utilized a ramshead at the discharge pipe exit, a pipe length of 112' and a submergence length of approximately 13'. Loads on valve internals during the test program are larger than loads on valve internals in the Shoreham configuration for the following reasons:

1. No dynamic mechanical load originating at the "tee" quencher is transmitted to the valve in the Shoreham configuration because there is at least one anchor point between the valve and the tee quencher.
2. The first length of the segment of piping downstream of the SRV in the test facility was twice that of Shoreham piping, thereby resulting in a bounding dynamic mechanical load on the valve in the test program.
3. Dynamic hydraulic loads (backpressure) are experienced by the valve internals in the Shoreham configuration.

The backpressure loads may be either (i) transient backpressures occurring during valve actuation, or (ii) steady-state backpressures occurring during steady-state flow following valve actuation.

(a) The key parameters affecting the transient backpressures are the fluid inertia in the submerged SRVDL and the SRVDL air volume. Transient backpressure increases with line submergence and decreases with air volume. The transient backpressure in the test program was maximized by utilizing a submergence of 13', not less than Shoreham, and a pipe length of 112' which is less than Shoreham.

(b) The steady-state backpressure in the test program was maximized by utilizing an orifice plate in the SRVDL above the water level and before the ramshead. The orifice was sized to produce a backpressure greater than that calculated for any of the Shoreham SRVDL's.

The differences in the line configuration between the Shoreham plant and the test program as discussed above result in the loads on the valve internals for the test facility which bound the actual Shoreham loads. An additional consideration in the selection of the ramshead for the test facility was to allow more direct measure-

ment of the thrust load in the final pipe segment. Utilization of a "tee" quencher in the test program would have required quencher supports that would unnecessarily obscure accurate measurement of the pipe thrust loads. For the reasons stated above, differences between the SRVDL configurations in Shoreham and the test facility will not have any adverse effect on SRV operability at Shoreham relative to the test facility.

2. Q. The test configuration utilized no spring hangers as pipe supports. Plant specific configurations do use spring hangers in conjunction with snubber and rigid supports. Describe the safety relief valve pipe supports used at Shoreham and compare the anticipated loads on valve internals for the Shoreham pipe supports to the measured loads in the test program. Describe the impact of any differences in loads on valve operability.

A. The Shoreham safety-relief valve discharge lines (SRVDL's) are supported by a combination of snubbers, rigid supports, and spring hangers. The locations of snubbers and rigid supports at Shoreham are such that the location of such supports in the BWR generic test facility is prototypical, i.e., in each case (Shoreham and the test facility) there are supports near each change of direction in the pipe routing. Additionally, each SRVDL at Shoreham has only one or two spring hangers, all of which are located in the drywell. The spring hangers, snubbers, and rigid supports were designed to accommodate combinations of loads resulting from piping dead weight, thermal conditions, seismic and suppression pool hydrodynamic events, and a high pressure steam discharge transient.

The dynamic load effects on the piping and supports of the test facility due to the water discharge event (the alternate shutdown cooling mode) were found to be significantly lower than corresponding loads resulting from the high pressure steam discharge event. As stated in

NEDE-24988-P, this finding is considered generic to all BWR's since the test facility was designed to be prototypical of the features pertinent to this issue. Furthermore, analysis of a typical Shoreham SRVDL configuration has confirmed the applicability of this conclusion to Shoreham.

During the water discharge transient there will be significantly lower dynamic loads acting on the snubbers and rigid supports than during the steam discharge transient. This will more than offset the small increase in the dead load on these supports due to the weight of the water. Therefore, design adequacy of the snubbers and rigid supports is assured as they are designed for the larger steam discharge transient loads.

This question addresses the design adequacy of the spring hangers with respect to the increased dead load due to the weight of the water during the liquid discharge transient. As was discussed with respect to snubbers and rigid supports, the dynamic loads resulting from liquid discharge during the alternate shutdown cooling mode of operation are significantly lower than those from the high pressure steam discharge. It is believed that sufficient margin exists in the Shoreham piping system design to adequately offset the increased dead

load on the spring hangers in an unpinned condition due to a water filled condition. Nevertheless, stress analyses are being performed to confirm this assumption regarding the increased deadweight loads for all SRVDL spring hangers. It should be noted that the effect of dead load weight does not affect the ability of SRVs to open to establish the alternate shutdown cooling path since the loads occur only after valve opening.

3. Q. Report NEDE-24988-P did not identify any valve functional deficiencies or anomalies encountered during the test program. Describe the impact on valve safety function of any valve functional deficiencies or anomalies encountered during the program.

A. No functional deficiencies or anomalies of the safety relief or relief valves, not only for Target Rock two-stage valves but also for all other types of valves tested, were experienced during the testing by Wyle Laboratories for compliance with the alternate shutdown cooling mode requirement. All the valves subjected to test runs, valid and invalid, opened and closed without loss of pressure integrity or damage. Anomalies encountered during the test program were all due to failures of test facility instrumentation, equipment, data acquisition equipment, or deviation from the approved test procedure.

The test specification for each valve required six valid runs. Under the test procedure, any anomaly caused the test run to be judged invalid. In testing for the Target Rock two-stage SRV, only one anomaly of any sort occurred: on water test run No. 302, the test system GN_2 regulator failed, resulting in a test which did not comply with the procedural test requirements. The Wyle Laboratories test log sheet for the Target Rock two-stage valve tests is attached.

Each Wyle test report for the respective valves
fies each test run performed and documents whether
the test run is valid or invalid and states the reason
for considering the run invalid. No anomaly encountered during the required test program affects any valve safety or operability function.

All valid test runs are identified in Table 2.2-1 of NEDE-24988-P. The data presented in Table 4.2-1 for each valve were obtained from the Table 2.2-1 test runs and were based upon the selection criteria of:

(a) Presenting the maximum representative loading information obtained from the steam run data,

(b) Presenting the maximum representative water loading information obtained from the 15° F subcooled water test data,

(c) Presenting the data on the only test run performed for the 50° F subcooled water test condition.

OPERABILITY TEST REPORT
FOR
TARGET ROCK 6X10 SRV
FOR
LOW PRESSURE WATER TESTS
FOR
GENERAL ELECTRIC COMPANY

GENERAL ELECTRIC
NUCLEAR ENERGY BUSINESS GROUP
J. J. Moon 2-10-82
APPROVED DATE
3682-77-1
VPF NO.
D810270
TRANSMITTAL NO.

175 Curtner Avenue
San Jose, California

TABLE I

TEST LOG FOR SRV TR-1

Test No.	Test Media	Load Line Configuration	Test Date	Remarks
301	Steam	I	3/17/81	Acceptable
302	Water	I	3/17/81	GN_2 Regulator failed. Data not acceptable.
303	Water	I	3/17/81	Acceptable
304	Steam	I	3/17/81	Acceptable
305	Water	I	3/18/81	Acceptable
306	Steam	I	3/18/81	Acceptable
307	Water	I	3/18/81	Acceptable
308	Water	I	3/18/81	Special test at elevated temperature and low pressure requested by G.E.

4. Q. The purpose of the test program was to determine valve performance under conditions anticipated to be encountered in the plants. Describe the events and anticipated conditions at Shoreham for which the valves are required to operate and compare these plant conditions to the conditions in the test program. Describe the plant features assumed in the event evaluations used to scope the test program and compare them to plant features at Shoreham. For example, describe high level trips to prevent water from entering the steam lines under high pressure operating conditions as assumed in the test event and compare them to trips used at Shoreham.

A. The purpose of the test program was to determine valve performance, under conditions anticipated to be encountered in the plants, which could result in liquid or two phase flow through the valves. The alternate shutdown cooling mode is the only anticipated event which is expected to result in liquid at the valve inlet. Consequently, this was the event simulated in the SRV test program. This conclusion and the test results applicable to Shoreham are discussed below. The alternate shutdown cooling mode has been described in the response to NRC question 5.

The SRV inlet fluid conditions tested in the BWR Owners' Group SRV test program, as documented in NEDE-24988-P, are representative of the fluid conditions expected to occur in the alternate shutdown cooling mode of operation at Shoreham. These fluid conditions at the SRV inlet are 15° F to 50° F subcooled liquid at 20 psig to 250 psig.

The BWR Owners' Group, in their enclosure to the September 17, 1980 letter from D. B. Waters to R. H. Vollmer, identified thirteen events which could result in liquid or two phase SRV inlet flow. These events were identified by evaluating the initiating events described in Reg. Guide 1.70, Rev. 2, with and without the additional conservatism of a single active component failure or operator error postulated with the event sequence. Of these thirteen events, only eight are applicable to the Shoreham plant because of its design and specific plant configuration. For these eight events, the Shoreham specific features, such as trip logic, power supplies, instrument line configuration, alarms and operator actions, have been compared to the base case analysis presented in the BWR Owners' Group September 17, 1980 submittal and subsequent discussions with the NRC Staff. This comparison has demonstrated that in each case, the base case analysis is applicable to Shoreham in that the base case assumptions are applicable. For example, the base case analysis for the reactor level 8 failure/HPCI overfill event included a level 8 trip scheme with two out of two logic, two variable instrument legs and one power supply inputting to one HPCI turbine trip mechanism with one turbine stop valve. This scheme is the same as the Shoreham design.

As discussed above, the Shoreham plant features are represented in the base case analysis performed in the BWR Owners' Group evaluation. This evaluation concluded that the alternate shutdown cooling mode is the only expected operating event involving liquid or two phase flow and therefore requires testing. The alternate shutdown cooling mode fluid conditions tested in the BWR Owners' Group test program accurately bound the Shoreham plant specific fluid conditions expected for this event.

5. Q. The valves are likely to be extensively cycled in a controlled depressurization mode in a plant specific application. Was this mode simulated in the test program? What is the effect of this valve cycling on valve performance and probability of the valve to fail open or to fail closed?

A. The BWR safety/relief valve (SRV) operability test program was designed to simulate the alternate shutdown cooling mode, which is the only expected liquid discharge event for Shoreham. The sequence of events leading to the alternate shutdown cooling mode is given below.

Following normal reactor shutdown, the reactor operator depressurizes the reactor vessel by opening the turbine bypass valves and removing heat through the main condenser. If the main condenser is unavailable, the operator could depressurize the reactor vessel by using the SRV's to discharge steam to the suppression pool. If SRV operation is required, the operator cycles the valves in order to assure that the cooldown rate is maintained within the technical specification limit of 100° F per hour. This would require on the order of 1-10 cycles of the SRV. When the vessel is depressurized, the operator initiates normal shutdown cooling by use of the RHR system. If that system is unavailable because the valve on the RHR shutdown cooling suction line fails to open, the operator initiates the alternate shutdown cooling mode.

For alternate shutdown cooling, the operator opens one SRV and initiates either an RHR or core spray pump utilizing the suppression pool as the suction source. The reactor vessel is filled such that water is allowed to flow into the main steam lines and out of the SRV and back to the suppression pool. Cooling of the system is provided by use of an RHR heat exchanger. As a result, an alternate cooling mode is maintained.

In order to assure continuous long term heat removal, the SRV is kept open and no cycling of the valve is performed. In order to control the reactor vessel cooldown rate, the operator is instructed to throttle the injection valve into the vessel. Consequently, no cycling of the SRV is required for the alternate shutdown cooling mode, and no cycling of the SRV was performed for the generic BWR SRV operability test program.

The ability of the Shoreham SRV to be extensively cycled for steam discharge conditions has been confirmed during steam discharge qualification testing of the valve by the valve vendor. This qualification testing for the Target Rock two-stage valve used in Shoreham has been previously identified in the Shoreham response to NRC question 212.51. Based on the qualification

testing of the SRV's, the cycling of the valves in a controlled depressurization mode for steam discharge conditions will not adversely affect valve performance and the probability of the valve to fail open or closed is extremely low.

6. Q. Describe how the values of valve C_v's in report NEDE-24988-P will be used at Shoreham. Show that the methodology used in the test program to determine the valve C_v will be consistent with the application at Shoreham.

A. The flow coefficient, C_v, for the Target Rock 6 x 10 two-stage pilot operated safety relief valve (SRV) utilized in Shoreham was determined in the generic SRV test program (NEDE-24988-P). The average flow coefficient calculated from the test results for the Target Rock two-stage valve, model 7567F, is reported in Table 5.2-1 of NEDE-24988-P. This test value has been used by LILCO to confirm that the liquid discharge flow capacity of the Shoreham SRVs will be sufficient to remove core decay heat when injecting into the reactor pressure vessel (RPV) in the alternate shutdown cooling mode. The C_v value determined in the SRV test demonstrates that the Shoreham SRVs are capable of returning the flow injected by the RHR or CS pump to the suppression pool.

If the operator were to place the Shoreham plant in the alternate shutdown cooling mode, he would assure that adequate core cooling was being provided by monitoring the following parameters: RHR or CS flow rate, reactor vessel pressure and reactor vessel temperature.

The flow coefficient for the Target Rock two-stage valve reported in NEDE-24988-P was determined from the SRV

flow rate when the valve inlet was pressurized to approximately 250 psig. The valve flow rate was measured with the supply line flow venturi upstream of the steam chest. The C_v for the valve was calculated using the nominal measured pressure differential between the valve inlet (steam chest) and 3' downstream of the valve and the corresponding measured flowrate. Furthermore, the test conditions and test configuration were representative of Shoreham plant conditions for the alternate shutdown cooling mode, e.g. pressure upstream of the valve, fluid temperature, friction losses and liquid flowrate. Therefore the reported C_v values are appropriate for application to the Shoreham plant.

Attachment 2

NRC QUESTION 4

The purpose of the test program was to determine valve performance under conditions anticipated to be encountered in the plants. Describe the events and anticipated conditions at Shoreham for which the valves are required to operate and compare these plant conditions to the conditions in the test program. Describe the plant features assumed in the event evaluations used to scope the test program and compare them to plant features at Shoreham. For example, describe high level trips to prevent water from entering the steam lines under high pressure operating conditions as assumed in the test event and compare them to trips used at Shoreham.

RESPONSE TO NRC QUESTION 4

The purpose of the S/RV test program was to demonstrate that the Safety Relief Valve (S/RV) will open and reclose under all expected flow conditions. The expected valve operating conditions were determined through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. Single failures were applied to these analyses so that the dynamic forces on the safety and relief valves would be maximized. Test pressures were the highest predicted by conventional safety analysis procedures. The BWR Owners Group, in their enclosure to the September 17, 1980 letter from D. B. Waters to R. H. Vollmer, identified 13 events which may result in liquid or two-phase S/RV inlet flow that would maximize the dynamic forces on the safety and relief valve. These events were identified by evaluating the initiating events described in Regulatory Guide 1.70, Revision 2, with and without the additional conservatism of a single active component failure or operator error postulated in the event sequence. It was concluded from this evaluation that the alternate shutdown cooling mode is the only expected event which will result in liquid at the valve inlet. Consequently, this was the event simulated in the S/RV test program. This conclusion and the test results applicable to Shoreham are discussed below. The alternate shutdown cooling mode of operation has been described in the response to NRC Question 5.

The S/RV inlet fluid conditions tested in the BWR Owners Group S/RV test program, as documented in NEDE-24988-P, are 15°F to 50°F subcooled liquid at 20 psig to 250 psig. These fluid conditions envelope the conditions expected to occur at Shoreham in the alternate shutdown cooling mode of operation.
The BWR Owners Group identified 13 events by evaluating the initiating events described in Regulatory Guide 1.70, Revision 2, with and without the additional conservatism of a single active component failure or operator error postulated in the events sequence. These events and the plant-specific features that mitigate these events, are summarized in Table 1. Of these 13 events, only 8 are applicable to the Shoreham plant because of

its design and specific plant configuration. Five events, namely 2, 5, 6, 10, and 13 are not applicable to the Shoreham plant for the reasons listed below:

a. Event 2 will only result in steam conditions at the S/RV inlet because the Shoreham plant has safety relief valves located higher than the MSIVs.

b. Events 5 and 10 require initiation of the HPCS system. This system is not present in the Shoreham design.

c. Event 6 requires initiation of the RCIC system with head sprays. The Shoreham plant design does not include head sprays.

d. Event 13 is not applicable for the Shoreham plant because there are no procedures or specific design features that lead to break isolation in the event of a large break accident.

For these 8 remaining events, the Shoreham specific features, such as trip logic, power supplies, instrument line configuration, alarms and operator actions, have been compared to the base case analysis presented in the BWR Owners Group submittal of September 17, 1980. This comparison has demonstrated that in each case, the base case analysis is applicable to Shoreham because the base case analysis does not include any plant features which are not already present in the Shoreham design. For events, 1, 3, 4, 8, 9, 11, and 12, Table 1 demonstrates that the Shoreham specific features are included in the base case analyses presented in the BWR Owners Group submittal of September 17, 1980. It is seen from Table 1, that all plant features assumed in the event evaluation are also existing features in the Shoreham plant. For example, the base case analysis for Event 3, the reactor Level 8 failure/HPCI overfill event, included a Level 8 trip scheme with 2 out of 2 logic, 2 variable instrument legs and 1 power supply inputting to 1 HPCI turbine trip mechanism with a turbine stop valve. All features included in this base case analysis are similar to plant features in the Shoreham design. Furthermore, the time available for operator action, is expected to be longer in the Shoreham plant than in the base case analysis for each case where operator action is required.

Event 7, the alternate shutdown cooling mode of operation, is the only expected event which will result in liquid or two-phase fluid at the S/RV inlet. Consequently, this event was simulated in the BWR S/RV test program. In Shoreham, this event involves flow of subcooled water (approximately 20°F subcooled) at a pressure of approximately 50 psig. The test conditions clearly envelope these plant conditions.

As discussed above, the BWR Owners Group evaluated transients including single active failures that would maximize the dynamic

forces on the safety relief valves. As a result of this evaluation, the alternate shutdown cooling mode is the only expected event involving liquid or two-phase flow. Consequently this event was tested in the BWR S/RV test program. The fluid conditions and flow conditions tested in the BWR Owners Group test program conservatively envelope the Shoreham plant-specific fluid conditions expected for the alternate shutdown cooling mode of operation.

TABLE 1 - EVENTS EVALUATED

PLANT FEATURES	EVENTS EVALUATED												
	#1 FW Cont. Fail., L8 trip failure	#2 Press. Reg. Fail.	#3 Transient HPCI, L8 trip failure	#4 Transient RCIC, L8 trip failure	#5 Transient HPCS, L8 trip failure	#6 Transient RCIC Hd. Spr.	#7 Alt. Shutdown Cooling	#8 MSL Brk OSC	#9 SBA, RCIC, L8 trip failure	#10 SBA, HPCS, L8 trip failure	#11 SBA, HPCI, L8 trip failure	#12 SBA, Depress. & ECCS Over., Operator error	#13 LBA, ECCS Overf Brk Isol
High Water Level 7 Alarm	X / S		X / S	X / S	X / NA				X / S	X / NA	X / S	X / S	X / NA
High Drywell Pressure Alarm													
FW Level 8 Trip	X / S	X / NA											
RCIC Level 8 Trip			X / S	X / S	X / NA				X / S	X / NA	X / S		X / NA
HPCS Level 8 Trip				X / NA	X / NA				X / NA	X / NA			X / NA
HPCI Level 8 Trip			X / S	X / S					X / S		X / S		X / NA
HPCI/S and RCIC Initiation on Low Water Level	X / S	X / NA	X / S	X / S	X / NA	X / NA		X / S	X / S				X / NA
HPCI/S Initiation on High Drywell Pressure			X / S	X / S					X / S	X / NA	X / S	X / S	X / NA
RCIC Initiation on High Drywell Pressure													X / NA

TABLE 1 - EVENTS EVALUATED (continued)

PLANT FEATURES	#1 FW Cont. Fail., L8 trip failure	#2 Press. Reg. Fail.	#3 Transient HPCI, L8 trip failure	#4 Transient RCIC, L8 trip failure	#5 Transient HPCS, L8 trip failure	#6 Transient RCIC Hd. Spr.	#7 Alt. Shutdown Cooling	#8 MSL Brk OSC	#9 SBA, RCIC, L8 trip failure	#10 SBA, HPCS, L8 trip failure	#11 SBA, HPCI, L8 trip failure	#12 SBA, Depress. & ECCS Over., Operator Error	#13 LBA, ECCS Overf Brk Isol
Low Prèssure ECCS Initiation on High Drywell Pressure												X/S	X/NA
Low Pressure Initiation on Low Water Level													X/NA
FW Pumps Trip on Low Suction Pressure	X/S												
HPCI Trip on High Backpressure			X/S								X/S		
RCIC Trip on High Backpressure				X/S					X/S				
Turbine Trip on Vessel High Level	X/S	X/NA											
MSIVs Closure on Low Turbine Inlet Pressure	X/S	X/NA						X/S					
MSIVs Closure on High Steam Flow		X/NA						X/S					
MSIVs Closure on High Steam Tunnel Temperature								X/S					

TABLE 1 - EVENTS EVALUATED (continued)

PLANT FEATURES	EVENTS EVALUATED												
	#1 FW Cont. Fail., L8 trip failure	#2 Press. Reg. Fail.	#3 Transient HPCI, L8 trip failure	#4 Transient RCIC, L8 trip failure	#5 Transient HPCS, L8 trip failure	#6 Transient RCIC Hd. Spr.	#7 Alt. Shutdown Cooling	#8 MSL Brk OSC	#9 SBA, RCIC, L8 trip failure	#10 SBA, HPCS, L8 trip failure	#11 SBA, HPCI, L8 trip failure	#12 SBA, Depress. & ECCS Over., Operator Error	#13 LBA, ECCS Overf Brk Isol
MSIV Closure on High Radiation								X / S					
Reactor Scram on Turbine Trip	X / S	X / NA											
Reactor Scram on Neutron Flux Monitor		X / NA											
Reactor Scram on MSIVs Closure		X / NA											
Reactor Scram on High Radiation								X / S					
Reactor Scram on High Drywell Pressure									X / S	X / NA	X / S	X / S	X / NA
Reactor Scram on Low Water Level													X / NA
Reactor Isolation on Low Water Level													X / NA

*KEY:
- X - Feature considered in Base Case Analysis
- S - Feature in Shoreham Design
- NA - Not Applicable

LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION
P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

January 7, 1983 SNRC-816

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

SER Open Issue II.K.3.16
MSIV Setpoint Changes
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Reference: (1) LILCO letter, SNRC-563, (J. P. Novarro) to the NRC (H. R. Denton), dated 5/15/82

Dear Mr. Denton:

As a member of the BWR Owners Group, LILCO participated in the evaluation of NUREG-0737, Item II.K.3.16 "Reduction of Challenges to the SRVs" (Reference 1). The Shoreham Nuclear Power Station incorporated several of the recommended modifications to reduce the number of challenges to the safety/relief valves.

These modifications, in conjunction with the use of the two stage Target Rock SRVs, result in an estimated stuck open relief valve (SORV) frequency which is approximately an order of magnitude less than the BWR 4 benchmark plant. The present Shoreham design therefore meets the intent of NUREG-0737, II.K.3.16.

During the ASLB hearing litigation on SC Contention 28a (vi), LILCO indicated that it was investigating the feasibility of another modification which would further reduce the number of SRV challenges; i.e., changing the MSIV isolation setpoint from the present low reactor water level (L2) to the low-low level (L1). This investigation, which is complete, has determined that it is feasible for Shoreham to modify the setpoint so that isolation occurs at level 1, without impacting the safe operation of the unit.

As previously stated, Shoreham has already incorporated several design changes which have reduced the SORV frequency by an order of magnitude and, as shown in reference 1, the incremental benefit of the proposed MSIV isolation setpoint change will result in a minor improvement to the total SORV event frequency. This small incremental benefit is due to the fact that the previous modifications have already remedied common SORV initiators that were factored into the estimate of the benefits of the proposed modification taken singly. The benefits are not additive; each subsequent modification results in a smaller incremental improvement to the predicted SORV event frequency. Not withstanding this small incremental reduction in total SORV event frequency resulting from the incorporation of this change, LILCO commits to its implementation.

The aforementioned investigation, determined that several actions are necessary to support the actual physical incorporation of the hardware to implement the MSIV logic setpoint revision. Although the feasibility investigation determined that the FSAR Chapter 15 bounding analyses remain unaffected, two of the less critical transients will require revision. Specifically, the loss of offsite power and the loss of feedwater transients must be reanalyzed to reflect the lowered isolation setpoint.

The engineering revisions involve several organizations. The NSSS vendor must revise all related documentation such as elementary and one line drawings, logic diagrams, and the master parts list and provide formal permission to revise the MSIV isolation logic setpoint.

Upon receipt of the formal approval and the technical details of the revision, LILCO's architect/engineer will revise the applicable plant drawings and authorize the change. The Start-Up or the Plant Maintenance group will then implement the actual hardware modification. To support the modified MSIV isolation setpoint logic, the Plant Staff must revise the applicable station procedures. In addition, the operators must be trained using these procedures to ensure familiarity with the modified plant parameters resulting from the change. In view of the efforts involved in this modification, the implementation of the MSIV logic setpoint revision can not support the start-up testing schedule.

In summary, the Shoreham design already meets the intent of NUREG-0737. Previous modifications have reduced the predicted SORV frequency by an order of magnitude vis-a-vis the benchmark BWR-4 plant. The modification of the isolation logic requires hardware

modifications to systems that will not be an extended shutdown when access is permissible and the MSIVs are not required to perform a safety function.

In light of the above, and since the proposed MSIV isolation setpoint revision does not provide a significant contribution to the overall plant safety, LILCO intends to incorporate this plant modification during the first refueling outage.

In the course of the ASLB hearing litigation, LILCO committed to complete a feasibility evaluation for this modification. LILCO believes the information stated herein and the commitment to modify the MSIV isolation setpoint logic adequately fulfills this commitment.

Very truly yours,

Original signed by

J. L. Smith
Manager, Special Projects
Shoreham Nuclear Power Station

RJT/jm

cc: J. Higgins
All Parties

LILCO

LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION

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

Direct Dial Number

January 28, 1983 SNRC-824

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

Humphrey Concerns 3.3 and 3.4
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

References: (1) Robert L. Tedesco letter to M.S. Pollock, dated July 8, 1982
(2) J.L. Smith letter to H.R. Denton, SNRC-808, dated December 9, 1982

Dear Mr. Denton:

Reference (1) requested that Long Island Lighting Company provide a proposed program to respond to the subject concerns that were identified as being potentially applicable to Shoreham.

In Reference (2), LILCO transmitted its responses to those items, with the exception of items 3.3 and 3.4. Reference (2) committed to a submittal in January of 1983 relative to these two items.

Since the time that Reference (1) was transmitted, continued analytical investigation has been conducted by Stone & Webster Engineering Corporation into the submerged structure loads and piping/pipe support loads associated with RHR heat exchanger relief valve discharge events during the Steam Condensing mode (SCM) of RHR system operation. This analysis is complex due to: a) the various potential modes of system failure, and b) the response characteristics of system components. To date, we have been unable to demonstrate that the existing piping, supports, and structures can withstand the present conservatively predicted loadings which may be imposed by this event.

As a result, LILCO has elected to delete the SCM as an operating mode of RHR, for all normal plant operations. This would allow LILCO to utilize available engineering resources more efficiently for the purpose of completing construction and supporting the current schedule for fuel load.

Accordingly, a standing order will be issued to operating personnel prohibiting the use of SCM during normal plant operations. Additionally, manual isolation valves which isolate RHR system pressure control valves 1E11*PCV007A, B, level and pressure controllers 1E11* LT002A, B and 1E11*PT003A, B respectively, and upstream manual isolation valves which isolate RHR system pressure control valves 1E11*PCV003A, B will be maintained normally closed and administratively controlled to preclude inadvertent operation in the SCM. The Station Test Procedure for SCM operation will not be conducted until an analytical justification has been developed.

A review of the Shoreham licensing application was conducted and it has been determined that there are no design basis transients or accidents which, for the purpose of mitigation, require the use of the SCM of the RHR system.

However, for non-design basis events involving multiple failures (e.g. Station Blackout), the SCM flow path of the RHR system will be retained as an available option for use by operating personnel when all other means of core/containment cooling have been exhausted. For such an event, reactor pressure would be less than the relief setpoint of 450 psig and the system would only be operated in a local manual mode.

It is our belief that, given the above commitments, this issue is removed as a licensing concern, with the understanding that LILCO may conduct an engineering effort in the future to demonstrate the ability of the RHR system to withstand the potential loads while operating in the SCM.

If you should have any questions concerning this matter, feel free to contact this office.

Very truly yours,

Original signed by

J. L. Smith
Manager, Special Projects
Shoreham Nuclear Power Station

DWD/jpb

cc: All Parties
J. Higgins, Site NRC

SNRC-831

LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION
P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

Direct Dial Number

February 18, 1983 SNRC-831

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

Mark II Hydrodynamic Loads Confirmatory Program
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Reference 1) SSER for Seismic and Dynamic Qualification of Safety Related Electrical And Mechanical Equipment, dated December 27, 1982

2) Plant Design Assessment for SRV and LOCA Loads Revision 5, December 1981 Shoreham Nuclear Power Station - Unit 1

Dear Mr. Denton:

In response to reference 1, item (3), LILCO hereby submits its report entitled, "Mark II Hydrodynamic Loads Confirmatory Program, Pipe Mounted Equipment Evaluation, Phase I." This report presents the current qualification levels for all motor operated valves (MOV's), on the 30 piping subsystems discussed in Reference 2, and the acceleration levels calculated for the Generic Long Term Program (LTP) confirmatory loads.

As stated in this report, all MOV's on the 30 piping subsystems have been evaluated and found to be adequately designed to accomodate the final generic (LTP) hydrodynamic loads.

LILCO believes this information sufficient to constitute closure of Phase I, Pipe Mounted Equipment concerns. Should you have any further questions regarding this matter, please feel free to contact this office.

Very truly yours,

Original signed by

J.L. Smith
Manager, Special Projects
Shoreham Nuclear Power Station

DWD:bc

Attachment

cc: J. Higgins
All Parties Listed in Attachment 1

ATTACHMENT 1

Lawrence Brenner, Esq.
Administrative Judge
Atomic Safety and Licensing
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Washington, D.C. 20555

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MARK II HYDRODYNAMIC LOADS CONFIRMATORY PROGRAM
PIPE MOUNTED EQUIPMENT EVALUATION - PHASE I
SHOREHAM NUCLEAR POWER STATION - UNIT 1
LONG ISLAND LIGHTING COMPANY

The objective of this report is to present additional information on Shoreham equipment evaluation results as a supplement to the Shoreham Design Assessment Report (DAR) Revision 5 (Reference 1) Appendix L "Mark II Hydrodynamic Loads Confirmatory Program."

As stated in the DAR, the Shoreham Mark II hydrodynamic loads confirmatory program has evaluated the plant with respect to the final generic Long-Term Program (LTP) hydrodynamic loads. The LTP hydrodynamic loads, the scope and procedure of the confirmatory program, and the evaluation results have been discussed in Reference 1. The evaluation concluded that the Shoreham reactor building structures, piping, and equipment had been adequately designed to accommodate the final generic LTP hydrodynamic loads, with the exception of a number of motor-operated valves (MOVs). These MOVs are generally the same valves that had acceleration values due to the design bases hydrodynamic loads which exceeded original qualification levels.

Since the evaluation results became available, analytical efforts as well as a requalification test program have been completed to demonstrate that the integrity and operability of the valves can be assured. This report presents the current qualification levels for all MOVs on the 30 piping subsystems discussed in the DAR and the acceleration levels calculated for the LTP confirmatory loads.

As indicated in Reference 1, the Shoreham reactor building structural dynamic analysis results had clearly shown that the most significant final generic load is the CO-basic load. The load definition is a direct application of the 4TCO test data on the Shoreham pool boundary with a conservative spatial distribution. NUREG-0808 (Reference 2) has acknowledged the conservative nature of the load definition and allows credit to be taken for the pool size effect and pool temperature range (Reference 3). Shoreham has performed a plant unique assessment and concluded that a CO-basic load reduction ratio of 0.7 can be applied for the pool size effect. Shoreham has conservatively elected not to take credit for the pool temperature range effect for results reported to date.

As it was discussed throughout the DAR, the structural analyses have generally employed simplifying assumptions that are conservative in nature. An example is the treatment of axisymmetric hydrodynamic loads such as the CO-basic load definition. The support excitation to a piping subsystem that is attached to the containment wall is a one-directional radial excitation. The design analyses generally employed have been conservatively performed with the full amplitude of radial excitation applied in two perpendicular horizontal directions. This substantial conservatism in the CO-basic load analysis has been removed in the piping analyses for the 30 piping subsystem evaluated herein.

The acceleration values at the MOVs have been calculated with the Shoreham pool size effect taken into account for the CO load definition and the tangential component of excitation removed for axisymmetric loads. The results are summarized in Table 1 for the 60 MOVs on the 30 piping subsystems evaluated. Also shown are the qualification levels for both the valve operators and assemblies.

The operator qualification levels were arrived at by test. The assembly qualification levels reflect the acceleration values used in the original qualification stress analyses ratioed up by the faulted condition factor of safety.

For the 60 valves evaluated, all calculated accelerations were found to be acceptable. For one valve, 1E11*MOV-039A, the calculated horizontal acceleration exceeds the valve assembly qualification level, but the combined effect with a lower vertical acceleration results in acceptable calculated stresses.

Based on the information presented herein, it is concluded that all Shoreham equipment will be proven to be within qualification levels. Phase II of this program is underway to address all valves on the remaining piping subsystems attached to the primary containment at locations of high amplitude ARS. It is expected that the program will provide positive confirmation that the qualification of Shoreham MOVs conform with the requirements of the Mark II LTP hydrodynamic load definitions.

References:

1. Shoreham Nuclear Power Station Unit 1
   Plant Design Assessment Report for SRV and LOCA
   Loads, Revision 5, December 1981

2. Mark II Containment Program Load Evaluation and
   Acceptance Criteria, NUREG-0808, August 1981

3. Mark II Containment Lead Plant Program Load
   Evaluation and Acceptance Criteria, NUREG-0487,
   Supplement 2, February 1981

TABLE 1

CONFIRMATORY VS. QUALIFICATION LEVEL ACCELERATIONS

Valve Mark No.	AX No.	Confirmatory at Operator C.G. G_H	Confirmatory at Operator C.G. G_V	SQRT No.	Qualification Level Operator G_H	Qualification Level Operator G_V	Qualification Level Assembly G_H	Qualification Level Assembly G_V
1E11*MOV032A	8C-1	9.8	1.4	88AD-7	10.0	10.0	>10.0	>10.0
1E11*MOV031A	8C-1	6.9	1.2	88AD-7	10.0	10.0	>10.0	>10.0
1E11*MOV036A	8F-1	2.7	1.9	88AD-10	7.0	7.0	>7.0	>7.0
1E11*MOV037A	8F-1	4.4	4.8	88AD-9	7.0	7.0	>7.0	>7.0
1E11*MOV038A	8F-1	4.7	1.0	88AD-2	10.0	10.0	>10.0	5.2
1E11*MOV039A	8F-1	8.3	0.5	88V-16	10.0	10.0	7.4	9.5
1E11*MOV040A	8F-1	7.1	4.2	88V-20	10.0	10.0	>10.0	>10.0
1E11*MOV041A	8F-1	5.5	1.6	88V-6	10.0	10.0	>10.0	>10.0
1E11*MOV042A	8F-1	3.5	2.5	88AD-5	10.0	10.0	>10.0	>10.0
1E11*MOV035A	8G-2	1.5	1.1	88V-20	10.0	10.0	>10.0	>10.0
1E11*MOV033A	8G-2	2.3	1.3	88V-20	10.0	10.0	>10.0	>10.0
1E11*MOV034A	8G-2	1.5	1.1	88AD-6	10.0	10.0	5.5	3.9
1E11*PCV003A	8G-2	1.8	1.9	318-2	3.0	3.0	>3.0	>3.0
1E11*MOV037B	8H-1	5.3	5.6	88AD-9	7.0	7.0	>7.0	>7.0
1E11*MOV036B	8H-1	6.7	3.2	88AD-10	7.0	7.0	>7.0	>7.0
1E11*MOV050	8H-1	9.5	3.1	88V-21	10.0	10.0	>10.0	4.7
1E11*MOV040B	8H-1	9.1	3.5	88V-20	10.0	10.0	>10.0	>10.0
1E11*MOV038B	8H-1	3.8	1.1	88AD-2	10.0	10.0	>10.0	5.2
1E11*MOV039B	8H-1	4.7	3.8	88V-16	10.0	10.0	7.4	9.5
1E11*MOV053	8L-1	2.5	1.4	88V-11	10.0	10.0	8.5	5.4
1E11*MOV054	8L-1	1.7	1.0	88V-11	10.0	10.0	8.5	5.4

TABLE 1 (CONTINUED)

CONFIRMATORY VS. QUALIFICATION LEVEL ACCELERATIONS

Valve Mark No.	AX No.	CONFIRMATORY AT OPERATOR C.G.		SQRT No.	QUALIFICATION LEVEL OPERATOR		ASSEMBLY	
		GH	GV		GH	GV	GH	GV
1E51*MOV042	2A-2	1.6	1.2	88V-2	10.0	10.0	>10.0	4.3
1E51*MOV048	2A-2	1.1	1.6	253-3	10.0	10.0	7.3	3.3
1E51*MOV041	2A-2	1.0	1.3	88V-2	10.0	10.0	>10.0	4.3
1E51*MOV047	2A-2	1.0	3.3	253-3	10.0	10.0	7.3	3.3
1E51*MOV032	2C-1	2.4	2.4	88V-6	10.0	10.0	>10.0	>10.0
1E51*MOV031	2C-1	3.2	1.5	88V-6	10.0	10.0	>10.0	>10.0
1E41*MOV042	11G-1	1.4	0.6	88V-17	10.0	10.0	3.9	3.9
1E41*MOV048	11G-1	0.7	2.5	253-3	10.0	10.0	7.3	3.3
1E41*MOV041	24A-1	3.9	1.6	88V-13	10.0	10.0	5.6	2.3
1E41*MOV047	24A-1	6.4	0.3	253-3	10.0	10.0	7.3	3.3
1B21*MOV085	25J-1	1.6	1.4	253-1	10.0	10.0	3.2	4.5
1B21*MOV083	25J-1	1.9	1.8	253-1	10.0	10.0	3.2	4.5
1B21*MOV084	25J-1	1.6	2.6	253-1	10.0	10.0	3.2	4.5
1P41*MOV033A	33A-1	0.7	0.6	197-1	8.0	8.0	>8.0	>8.0
1P41*MOV033B	33A-1	0.5	0.4	197-1	8.0	8.0	>8.0	>8.0
1P41*MOV033C	33A-1	0.8	0.4	197-1	8.0	8.0	>8.0	>8.0
1P41*MOV033D	33A-1	0.5	0.6	197-1	8.0	8.0	>8.0	>8.0
1P41*MOV042A	33A-1	1.0	1.8	197-3	8.0	8.0	6.6	6.6
1P41*MOV042B	33A-1	0.8	1.6	197-3	8.0	8.0	6.6	6.6
1E11*MOV031C	8C-1	7.5	0.6	88AD-7	10.0	10.0	>10.0	>10.0
1E11*MOV032C	8C-1	4.6	6.2	88AD-7	10.0	10.0	>10.0	>10.0

TABLE 1 (CONTINUED)

CONFIRMATORY VS. QUALIFICATION LEVEL ACCELERATIONS

Valve Mark No.	AX No.	CONFIRMATORY AT OPERATOR C.G.		SQRT NO.	QUALIFICATION LEVEL OPERATOR		QUALIFICATION LEVEL ASSEMBLY	
		G_H	G_V		G_H	G_V	G_H	G_V
1E11*MOV047	8N-1	4.0	1.3	88AD-8	10.0	10.0	>10.0	>10.0
1E32*MOV021C	60A-1	1.9	1.6	253-4	10.0	10.0	8.1	8.1
1E32*MOV022C	60A-1	1.6	1.9	253-4	10.0	10.0	8.1	8.1
1B21*MOV063	60A-1	1.3	0.9	253-1	10.0	10.0	3.2	4.5
1B21*MOV068C	60A-1	1.3	0.6	253-1	10.0	10.0	3.2	4.5
1E32*MOV021D	60B-1	2.4	1.2	253-4	10.0	10.0	8.1	8.1
1E32*MOV022D	60B-1	2.1	0.7	253-4	10.0	10.0	8.1	8.1
1B21*MOV064	60B-1	1.8	1.2	253-1	10.0	10.0	3.2	4.5
1B21*MOV068D	60B-1	1.2	0.9	253-1	10.0	10.0	3.2	4.5
1E32*MOV022A	60E-1	1.5	1.1	253-4	10.0	10.0	8.1	8.1
1E32*MOV021A	60E-1	2.8	1.6	253-4	10.0	10.0	8.1	8.1
1B21*MOV061	60E-1	1.9	0.9	253-1	10.0	10.0	3.2	4.5
1B21*MOV068A	60E-1	1.4	0.7	253-1	10.0	10.0	3.2	4.5
1B21*MOV068B	60F-1	1.4	0.9	253-1	10.0	10.0	3.2	4.5
1E32*MOV021B	60F-1	2.4	2.1	253-4	10.0	10.0	8.1	8.1
1E32*MOV022B	60F-1	1.8	1.8	253-4	10.0	10.0	8.1	8.1
1B21*MOV062	60F-1	1.3	0.9	253-1	10.0	10.0	3.2	4.5
1E41*MOV049	11G-1	1.5	2.1	253-1	10.0	10.0	3.2	4.5