LILCO LONG ISLAND LIGHTING

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.

B001
1/AD

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,

J. L. Smith

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

RWG/law

Enclosures

cc: J. Higgins
All Parties

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

Page 1 of 3	EVENTS EVALUATED												
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
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	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
Low Pressure 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	#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

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

Dupe of ~~8208030301~~

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 identifies each test run performed and documents whether or not 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. Moore 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.