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

SNRC-577

May 27, 1981

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

Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Dear Mr. Denton:

Forwarded herewith are sixty (60) copies of LILCO's responses to the Safety Evaluation Report (SER) Outstanding Issues listed in Attachment 1.

Please note that our responses to the Outstanding Issues listed in Attachment 2 will be forwarded to you under separate cover by June 1, 1981.

Very truly yours,

J. P. Novarro

J. P. Novarro
Project Manager
Shoreham Nuclear Power Station

CC:mc

Enclosures

cc: J. Higgins

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Attachment 1 - SER Outstanding Issues

Number	Issue
1	Pool Dynamic Loads
4	Piping Vibration Test Program - Safety related snubbers
5	LOCA Loadings on Reactor Vessel Supports and Internals
6	Downcomer Fatigue Analysis
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15	Inservice Testing of Pumps and Valves
16	Leak Testing of Pressure Isolation Valves
17	SRV Surveillance Program
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ATTACHMENT 2 - SER OUTSTANDING ISSUES

SCHEDULED FOR SUBMITTAL BY JUNE 1, 1981

Number	Issue
9	Environmental Qualification
20	Appendix G-IV.A.2.a - Nil Ductility Temperature
21	Appendix G-IV.A.2.c - Pressure Temperature Limit
22	Appendix G - Impact Testing
23	Appendix G-IV.B - Minimum Upper Shelf Energy
35	Containment Isolation
37	Secondary Containment Bypass Leakage
38	Fracture Prevention of Containment Pressure Boundary
51	Fracture Toughness of Steam Line and Feedwater Materials
57	TMI-2 Requirements (Third and final submittal scheduled for 5/30/81)
58 *	Reactor Vessel Materials Toughness

* Based upon discussions with J. N. Wilson, NRC Project Manager, Shoreham, it is our understanding that Outstanding Issue Number 58, Reactor Vessel Materials Toughness, does not require a LILCO submittal.

Item 1 - Pool Dynamic Loads

Shoreham has followed all NUREG requirements with respect to pool dynamic loads with the exceptions noted below. These requirements are outlined in NUREG-0487 together with Supplement 1, Supplement 2 and NUREG-0484 Rev. 1. Deviations from these requirements yet to be approved are:

1. In lieu of Supplement 2 of NUREG-0487, Shoreham has used an interim confirmatory chugging load definition whose power spectral density (PSD) essentially bounds the lead plants interim load definition. Please refer to our response to SER Open Item 28. In addition, Shoreham has committed to perform a subsequent design evaluation step using the final generic load.

2. To calculate quencher air clearing loads, Shoreham has used a generic load definition based on actual full scale T-quencher test results. Please refer to our response to SER Open Item 29. It is our understanding that review of this generic load definition has been completed and official NRC approval is imminent.

3. Shoreham has committed to use measured T-quencher arm and tie-down loads in those areas where Karlstein test data exceeds the generic T-quencher load specification proposed in the PP&L Design Assessment Report. Shoreham will commit to increase the quencher support bending moment used in the design assessment by a factor of 1.25. Please refer to our response to SER Open Item 34.

Item # 4 - Safety Related Snubbers

Prior to installation, all safety related snubbers are operationally stroked to determine that they are not frozen, seized or jammed. After installation and prior to system pre-operational testing, snubbers are visually inspected for signs of damage or impaired operability as well as for adequate swing clearances. In addition, their location, orientation, position setting and configuration are verified to be in accordance with approved design drawings. Fluid level and leakage are not applicable to Shoreham as hydraulic snubbers are not employed in any safety related systems.

The snubber thermal movements for those systems that have operating temperatures in excess of 250°F will be verified during the Startup Test phase under STP-811 - "System Thermal Expansion Test Procedure". This includes verification of the expected snubber thermal movements and swing clearances. Discrepancies or inconsistencies will be dispositioned prior to proceeding to the next test plateau. A preliminary list of the snubbers is attached. The final list will be included in the station technical specifications.

?/DATE	REVIEWER/CHECKER/DATE	INDEPENDENT REVIEWER/DATE
?ECT/TITLE		QA CATEGORY/CODE CLASS

PRELIMINARY

POOR ORIGINAL

SAFETY RELATED MECHANICAL SNUBBERS.

NO	SNUBBER	SYSTEM	AREA	CATEGORY	REMARKS.
1	2	3	4	5	6
1	PSSP0817	1B21	20	2	NOTES:- (A) FOR AREA SEE PHIS REPORT FOR SNUBBER LOCAT
2	PSSP0819	1B21	20	2	(B) FOR CATEGORY SEE NOTES 1, 2 & 3
3	PSSP0820	1B21	20	2	(1) THE FIRST SNUBBER AWAY FRO EACH REACTOR VESSEL NOZZLE.
4	PSSP0822	1B21	20	2	(2) EACH SNUBBER WITHIN FIVE FEET OF HEAVY EQUIPMENT.
5	PSSP0823	1B21	20	2	(VALVE, PUMP, TURBINE, MOTOR ETC)
6	PSSP0825	1B21	20	2	(3) EACH SNUBBER WITHIN TEN FEET OF THE DISCHAR
7	PSSP0826	1B21	20	2	FROM A SAFETY RELIEF VALV
8	PSSP0828	1B21	20	3	
9	PSSP0829	1B21	20	3	
10	PSSP0830	1B21	20	3	
11	PSSP0831	1B21	20	3	
12	PSSP0833	1B21	20	3	
13	PSSP0834	1B21	20	3	
14	PSSP0835	1B21	20	3	
15	PSSP0836	1B21	20	2	
16	PSSP0837	1B21	20	3	
17	PSSP0838	1B21	20	3	
18	PSSP0839	1B21	20	3	
19	PSSP0840	1B21	20	3	

PREPARER/DATE	REVIEWER/CHECKER/DATE	INDEPENDENT REVIEWER/DATE
SUBJECT/TITLE		QA CATEGORY/CODE CL-33

1	2	3	4	5	6
20	PSSP0841	1B21	20	3	POOR ORIGINAL
21	PSSP0842	1B21	20	3	
22	PSSP0843	1B21	20	3	
23	PSSP0844	1B21	20	3	
24	PSSP0847	1B21	20	2	
25	PSSP0848	1B21	20	2	
26	PSSP0849	1B21	20	2	
27	PSSP0850	1B21	20	2	
28	PSSP0851	1B21	20	2	
29	PSSP0852	1B21	20	2	
30	PSSP0855	1B21	20	2	
31	PSSP0858	1B21	20	2	
32	PSSP0792	1B21	20	2	
33	PSSP0793	1B21	20	2	
34	PSSP0794	1B21	20	2	
35	PSSP0859	1B21	20	2	
36	PSSP0860	1B21	20	2	
37	PSSP0864	1B21	20	2	
38	PSSP0865	1B21	20	2	
39	PSSP0273	1B21	20	2	
40	PSSP0274	1B21	20	2	
41	PSSP0275	1B21	20	2	
42	PSSP0281	1B21	20	2	

PREPARER/DATE	REVIEWER/CHECKER/DATE	INDEPENDENT REVIEWER/DATE
SUBJECT/TITLE		QA CATEGORY/CODE CLASS

1	2	3	4	5	6
43	PSSP0798	1B21	20	2	
44	PSSP0797	1B21	20	2	POOR ORIGINAL
45	PSSP0796	1B21	20	2	
46	PSSP0882	1B21	20	2	
47	PSSP0883	1B21	20	2	
48	PSSP0884	1B21	20	2	
49	PSSP0886	1B21	20	3	
50	PSSP0887	1B21	20	2	
51	PSSP0888	1B21	20	2	
52	PSSP0889	1B21	20	2	
53	PSSP0977	1B21	20	2	
54	PSSP0990	1B21	14	2	el 78' east
55	PSSP0993	1B21	20	2	
56	PSSP0997	1B21	20	2	
57	PSSP0996	1B21	20	2	
58	PSSP0998	1B21	20	2	
59	PSSP0999	1B21	20	2	
60	PSSP0785	1B21	20	2	
61	PSSP0786	1B21	20	2	
62	PSSP0584	1B21A	13	2	el 78' west
63	PSSP0586	1B21A	14	2	
64	PSSP0803	1B21A	20	2	

PREPARER/DATE | REVIEWER/CHECKER/DATE | INDEPENDENT REVIEWER/DATE

SUBJECT/TITLE | QA CATEGORY/CODE CLASS

1	2	3	4	5	6
65	PSSP0805	1B21A	20	1	
66	PSSP0808	1B21A	20	1	
67	PSSP0809	1B21A	20	1	
68	PSSP0813	1B21A	20	2	POOR ORIGINAL
69	PSSP0815	1B21A	20	2	
70	PSSP0031	1C41	20	1	
71	PSSP0812	E-11	11	3	
72	PSSP0815	E-11	11	3	
73	PSSP0824	E-11	12	2	
74	PSSP0825	E-11	06	2	
75	PSSP0826	E-11	12	2	
76	PSSP0827	E-11	12	2	
77	PSSP0828	E-11	07	3	
78	PSSP0830	E-11	14	2	
79	PSSP0832	E-11	20	2	
80	PSSP0837	E-11	03	2	
81	PSSP0838	E-11	06	2	
82	PSSP0839	E-11	06	2	
83	PSSP0901	E-11	20	2	
84	PSSP0903	E-11	20	2	
85	PSSP0906	E-11	20	2	
86	PSSP0907	E-11	20	2	

PREPARER/DATE

REVIEWER/CHECKER/DATE

INDEPENDENT REVIEWER/DATE

SUBJECT/TITLE

QA CATEGORY/CODE CLASS

1	2	3	4	5	6
87	PSSP 0908	E-11	20	2	
88	PSSP 0909	E-11	20	2	
89	PSSP 0910	E-11	20	2	
90	PSSP 0911	E-11	03	2	POOR ORIGINAL
91	PSSP 0803	E 11A	07	2	
92	PSSP 0805	E 11A	07	2	
93	PSSP 0806	E 11A	11	2	
94	PSSP 0808	E 11B	02	2	
95	PSSP 0809	E 11B	03	2	
96	PSSP 0811	E 11B	09	2	
97	PSSP 0818	E 11B	11	2	
98	PSSP 0819	E 11B	03	2	
99	PSSP 0821	E 11B	12	2	
100	PSSP 0850	E 11B	11	2	
101	PSSP 0806	E 21	20	2	
102	PSSP 0809	E 21	20	2	
103	PSSP 0060	E 41C	11	2	
104	PSSP 0800	E 41C	04	2	
105	PSSP 0801	E 41C	05	1	
106	PSSP 0805	E 41C	10	2	
107	PSSP 0072	E 51A	14	2	
108	PSSP 0073	E 51A	14	2	

ENGINEER/DATE	REVIEWER/CHECKER/DATE	INDEPENDENT REVIEWER/DATE

SUBJECT/TITLE

QA CATEGORY/CODE CLASS

1	2	3	4	5	6
109	PSSP 0807	IE S1B	20	2	
110	PSSP 0808	IE S1B	20	2	
111	PSSP 0810	IE S1B	20	2	
112	PSSP 0811	IE S1B	20	2	
113	PSSP 0803	E S1C	05	2	
114	PSSP 0804	E S1C	05	2	
115	PSSP 0805	E S1C	05	2	
116	PSSP 0226	G 33	20	2	
117	PSSP 0227	G 33	20	2	
118	PSSP 0234	G33	16	3	
119	PSSP 0235	G 33	16	3	
120	PSSP 0237	G 33A	14	2	
121	PSSP 0238	G 33A	14	2	
122	PSSP 0240	G 33A	14	2	
123	PSSP 0241	G 33A	14	2	
124	PSSP 0243	G 33A	14	2	
125	PSSP 0244	G 33A	14	2	
126	PSSP 0200	G 33C	20	2	
127	PSSP 0201	G 33C	16	2	
128	PSSP 0202	G 33C	16	2	
129	PSSP 0205	G 33C	16	2	
130	PSSP 0206	G 33C	16	2	

POOR ORIGINAL

PREPARER/DATE | REVIEWER/CHECKER/DATE | INDEPENDENT REVIEWER/DATE

SUBJECT/TITLE | QA CATEGORY/CODE CLASS

1	2	3	4	5	6
131	PSSP0207	G33C	16	Z	
132	PSSP0208	G33C	16	Z	
133	PSSP0210	G33C	16	Z	
134	PSSP0211	G33C	16	Z	
135	PSSP0214	G33C	16	Z	
136	PSSP0216	G33C	16	Z	
137	PSSP0218	G33C	16	Z	
138	PSSP0219	G33C	16	Z	
139	PSSP0220	G33C	16	Z	
140	PSSP0221	G33C	16	Z	
141	PSSP0223	G33C	16	Z	
142	PSSP0224	G33C	16	Z	
143	PSSP0232	G33C	16	Z	
144	PSSP0178	G41B	15	Z	
145	PSSP0179	G41B	18	Z	
146	PSSP0814	N11	21	Z	
147	PSSP0815	N11	21	Z	
148	PSSP0816	N11	21	Z	
149	PSSP0817	N11	21	Z	
150	PSSP0847	N11	21	Z	
151	PSSP0849	N11	21	Z	
152	PSSP0850	N11	21	Z	

POOR ORIGINAL

PREPARER/DATE | REVIEWER/CHECKER/DATE | INDEPENDENT REVIEWER/DATE

SUBJECT/TITLE | QA CATEGORY/CODE CLASS

1	2	3	4	5	6
153	PSSP0851	N11	21	Z	
154	PSSP0853	N11	21	Z	
155	PSSP0854	N11	21	Z	
156	PSSP0848	N11A	21	Z	
157	PSSP0852	N11A	21	Z	
158	PSSP0373	P41	01	Z	
159	PSSP0801	P41B	01	Z	
160	PSSP0802	P41B	01	Z	
161	PSSP0803	P41B	01	Z	
162	PSSP0804	P41B	01	Z	
163	PSSP0810	P41B	07	Z	
164	PSSP 081[illegible]	P41B	06	Z	
165	PSSP0817	P41B	24	Z	
166	PSSP0818	P41B	24	Z	
167	PSSP0820	P41B	24	Z	
168	PSSP0821	P41B	07	Z	
169	PSSP0832	P41B	07	Z	
170	PSSP0919	P41B	—	Z	
171	PSSP0807	P41B	01	Z	
172	PSSP0801	X60	23	Z	
173	PSSP0802	X60	23	Z	
174	PSSP044	X60			
175	PSSP043	X60			

POOR ORIGINAL

PREPARER/DATE | REVIEWER/CHECKER/DATE | INDEPENDENT REVIEWER/DATE

SUBJECT/TITLE | QA CATEGORY/CODE CLASS

1	2	3	4	5	6
176	PSSP0791	1B21	20	2	
177	PSSP0861	1B21	20	2	
178	PSSP0217	G33C	16	2	

POOR ORIGINAL

Item 5 - LOCA Loadings on Reactor Vessel Supports and Internals

This open item deals with the effects of pressurization of the annular space between the shield wall and the reactor vessel and with the integrity of the reactor vessel support system and internals when subjected to the resultant blowdown reaction forces. SER section 6.2.1.6 identifies the need for sufficient information to assess the effects of annulus pressurization on the reactor vessel support skirt. Shoreham contends that all necessary information has been submitted in a revised response to NRC question 041.1 dated September, 1978, and that this analysis is in fundamental agreement with the requirements of NUREG-0609. A nodalization sensitivity study, designed to maximize the asymmetric load on the reactor vessel "to show that conservative loads are used in assessing the design of the ... vessel support system" is described in FSAR Section 6.2.1.1.4.3 with results presented in FSAR Section 6.2.1.3.5.3 and Figure 6.2.1 - 34. Moreover, the analysis described in FSAR Section 6.2.1.3.5 has already been reviewed by NRC -CSB and the results are in substantial agreement with NRC - CSB predictions for Shoreham. NRC - CSB predicts a pressure in the break node approximately 6 percent greater than that predicted by Shoreham, but the integrated effect on the asymmetric load is less. A 6 percent increase in the break node pressure will increase the maximum asymmetric load by only 1.5 percent. Therefore, the load definition does not appear to be a significant issue.

In addition, Shoreham's response to Staff question 112.21 has been provided in SNRC 566, dated May, 1981.

Item 6 - Downcomer Fatigue Analysis

The fatigue analysis of the ASME class 2 and 3 downcomers and safety relief valve discharge piping in the wetwell is being completed by Shoreham using ASME Class 1 fatigue rules.

In DAR Rev. 4, Shoreham has committed to perform the fatigue analysis and to provide documentation of the results of the analysis by November, 1981 (DAR Rev. 5). We are prepared to review the preliminary results of this effort with the Staff on June 3, 1981 and to provide a letter submittal of the final results in August, 1981 well within SER Supplement 2 time frame.

It is Shoreham's opinion that this effort should be treated as part of the long term confirmatory program since the commitment has been made to perform the analysis. This opinion is consistent with many other requirements called for in the Mark II containment acceptance criteria and is particularly appropriate for a fatigue evaluation where potential problems, if any, would develop only after many years of plant operation.

Item 7 - Piping Functional Capability Criteria

In Section 3.9.2.1 of the Shoreham SER, the concern is raised that the Shoreham basis for assuring functional capability may not be the same as that approved by the NRC staff (NEDE-21985). Shoreham will amend the summary statement in DAR Revision 4, Appendix E (Section E1.0) to read as follows:

E1.0 SUMMARY

This Appendix provides "Functional Capability Criteria" for evaluation of essential piping in Mark II nuclear power plants. The criteria were established so as to be conservative and to assist in assuring maximum reliability of the piping considering all aspects of design, fabrication, in-service inspection, and operation.

The criteria are contained in pages E-5 and 6. The criteria are structured to make maximum use of the equations and definitions contained in the Code [1]. However, the functional capability criteria are not intended to substitute for or supersede any requirement of the Code.

The basis for the criteria is described in pages E-7 through 16.

The criteria are based, in large part, on the conservative approach contained in NUREG/CR-0261 [2]; i.e. on the single-hinge, limit moment concept with little or no consideration of strain hardening or dynamic effects. Recommendations or concepts given in NUREG/CR-0261 for B indices are used. For elbows with α_o <90°, excess conservatism has been avoided by using a right-hand-side limit of $1.5S_y$ or $2.0S_y$ rather than the less applicable factors on S_m or S_h as used in the Code for A, B, C, or D limits.

For Do/t >50, the allowable moments are decreased by increasing the B_2 indices and equivalents of (0.75i). This is based on test data on straight pipe at room temperature with, for ferritic materials, a temperature factor based on ratios of allowable longitudinal compressive stresses from Reference 1.

Dynamic effects may make the criteria very conservative when used for conditions where the loadings are dynamic in nature.

Shoreham hereby specifies that the functional capability criteria outlined in this appendix is equivalent to that presented in NEDE-21985 (Reference 6 approved by Reference 7).

Shoreham will also amend the last sentence of DAR Revision 4, Section 9.1.1.2.4 to remove the word "representative". This correction will reflect the current status that all Shoreham essential systems meet the functional capability requirements.

MARK-UP

number of valves actuated are determined for typical piping components.

These stress values are used to convert the number of stress cycles for various numbers of valves actuated to equivalent all valve actuation stress cycles by using an equivalent fatigue damage formula.

The equivalent fatigue damage formula is constructed such that an identical fatigue usage factor would be obtained from ASME Section III[6], Design Fatigue Curve, with the proper consideration of alternating stresses at a piping component. Furthermore, a conservative number of stress cycles per SRV actuation is used to obtain the appropriate total number of stress cycles for the SRV load.

For the Shoreham plant, this has been determined to be 4,050 stress cycles at the level of an all valve actuation and 45,400 stress cycles at the level of four valve actuation.

9.1.1.2.4 Functional Capability

The piping systems which are required to safely shut down the reactor and maintain its shutdown conditions as well as functioning to prevent or mitigate the consequences of LOCA are classified as essential systems. The other systems which do not have to perform the essential functions are classified as nonessential systems.

The basis for functional capability evaluation of the essential piping systems is "Functional Capability Criteria for Mark-II Piping" (Appendix E). The criteria are also outlined in Section 2 of this report. Evaluations of functional capability of the essential piping systems are to assure the fluid-flow capability of the piping systems. The capability will be maintained if significant reductions in cross-sectional area do not occur under any service condition.

Equation 9 of ASME Section III, NB-3652 is used, with modifications in accordance with Appendix E, for this evaluation. The results indicate that ALL Shoreham ~~representative~~ essential systems meet the functional capability requirements.

9.1.1.3 Results

9.1.1.3.1 Summary of Results

A reevaluation of all Shoreham reactor building piping and supports has been completed, accounting for the effects of the hydrodynamic SRV and LOCA loads. All load combinations and acceptance criteria described in Section 2 were addressed. The following three load combinations were found to be potentially controlling:

Item 15 - Inservice Inspection of Pumps & Valves

As stated in LILCO response to Item MEB-7 dated March 26, 1981 (LILCO letter to H. Denton from J. P. Novarro, SNRC-548), the pump and valve operability program described in that response is currently being reviewed. The ISI plan, as required by 10CFR50, will be submitted by January 1, 1982. Therefore, this request is premature.

Item 16 - Leak Testing of Pressure Isolation Valves

The periodic leak testing of pressure isolation valves will be conducted in conjunction with the Appendix J. Type C test program, the ASME code requirements and the technical specification surveillance requirements. Pressure isolation valves, as described herein, are defined as those redundant valves within the Class 1 piping boundary that form an interface between the reactor pressure vessel and a low pressure system (RHR, CORE SPRAY, LPCI).[1] These low pressure systems are also protected by valve position indication, pressure indication and relief valves, thus long term leakage will be detected and the postulated LOCA is not likely to occur.

Specific valves have been identified as pressure isolation valves and given the appropriate ASME Section XI category of A or AC. This information is provided in Table 1. In each case, the pressure isolation valve is also a containment isolation valve. Leakage tests for each of the pressure isolation valves are presented in Table 2. As described in FSAR response 212.106, these valves were grouped according to their leak tightness capabilities at reactor operating and reduced pressure conditions. Specifically, reduced pressure leak tests (Appendix J. Type C) are proposed for pressure isolation valves for which pressure tends to enhance leak tightness (e.g., globe valve with pressure overseat). For tests at system pressure, the acceptance limit on leakage will be 1 gpm.

The frequency of testing is presented below:

a) At least once per refueling outage approximately every 18 months.
b) Prior to returning the valve to service following maintenance, repair or replacement work on the internals of the valve which requires full valve disassembly and repairs to seat and/or disc.

[1] Note: This situation does not exist on the RCIC system.

TABLE 2

PRESSURE ISOLATION VALVE LEAK TESTS
SHOREHAM NUCLEAR POWER STATION

Valve	Proposed Test
E11 - MOV037 (A&B) E11 - MOV047 E11 - MOV048 E11 - MOV054 E21 - MOV033 (A&B)	Leak test at reduced pressure in conjunction with Appendix J, Type C Test Program.
E11 - MOV053 E11 - AOV-081 (A&B) E11 - MOV-081 (A&B) E21 - AOV081 (A&B) E21 - MOV081 (A&B)	Leak test at Reactor System pressure at frequency indicated. Acceptance Value 1 gpm

TABLE 1

PRESSURE ISOLATION VALVE DESCRIPTION AND CATEGORIZATION
SHOREHAM NUCLEAR POWER STATION

SYSTEM: Core Spray (E21)

Valve	Description	ASME Section XI Category	Class	Size (in)	Valve Type	Actuator Type	Normal Position
AOV081 A	Testable Check on	AC	1	10	C	AO	C
AOV081 B	Core Spray Discharge	AC	1	10	C	AO	C
MOV033 A	Outboard Isolation	A	1	10	GT	MO	C
MOV033 B	Valve on Core Spray Discharge Vessel	A	1	10	GT	MO	C
MOV081 A	Bypass on	A	1	2	GL	MO	C
MOV081 B	Testable Check	A	1	2	GL	MO	C

SYSTEM: Residual Heat Removal & LPCI (E11)

Valve	Description	ASME Section XI Category	Class	Size (in)	Valve Type	Actuator Type	Normal Position
MOV037 A	LPCI Injection	A	1	24	GT	MO	C
MOV037 B		A	1	24	GT	MO	C
MOV047	RHR Suction from RPV	A	1	20	GT	MO	C
MOV048		A	1	20	GT	MO	C
AOV081 A	LPCI Injection	AC	1	24	C	AO	C
AOV081 B	Testable Check	AC	1	24	C	AO	C
MOV081 A	Bypass on LPCI	A	1	1	GL	MO	C
MOV081 B	Injection Testable Check	A	1	1	GL	MO	C
MOV054	Head Spray (Inboard)	A	1	4	GT	MO	C
MOV053	Head Spray (Outboard)	A	2	4	GT	MO	C

Item #17 - SRV Surveillance Program

1.0 Purpose

The purpose of the program is to accumulate information in sufficient detail to allow identification of generic safety/relief valve problems.

2.0 Scope

This document defines the data records which will be used for monitoring performance of the Main Steam Line Safety/ Relief Valves (S/RVs) throughout the useful service life of each such valve.

2.1 The program is structured to collect sufficient data to allow the identification of safety/relief valve problems to minimize the possibility of a failure of the valve.

2.1.1 Data is also to be collected on inadvertent safety/relief valve operation, and on the failure of safety/relief valves to open or close.

2.2 For each safety/relief valve problem that is found to exist, the following information would be reported:

i) A description of the problem;
ii) The operating conditions;
iii) The failure mode(s) and the reason(s);
iv) The remedial action

3.0 Description of Program

3.1 Introduction

3.1.1 A MSL safety/relief valve data record will be maintained for the Shoreham SRV's.

3.1.2 The information identified in Section 3.2 through 3.5 will be recorded and maintained for each safety/relief valve.

3.1.3 The maintenance records will be updated with entries each time any work is done on the valve(s). This will include information regarding scheduled maintenance, and unscheduled maintenance, as described in Section 3.4.

3.1.4 If subassemblies or components are interchanged for maintenance, the records will provide traceability for the components and the valve assembly.

3.1.5 The SRV maintenance history will be analyzed annually for the purpose of identifying potential trends which could lead to generic safety/relief problems, as described in Section 3.5.

3.2 Records

Maintenance records will be maintained for each S/RV. The record will, as a minimum, identify the following:

i) Manufacturer and model number;
ii) Type and size (including throat bore);
iii) Manufacturer's serial number;
iv) Set pressure stamped on nameplate;
v) Date first installed on main steam line
vi) Vessel hydrotest pressure after the valves have been installed;
vii) Copy of original production test records;
viii) Results of startup tests and any special tests;
ix) Manufacturer's drawing and instruction manual references (GE-VPF).

3.3 Scheduled Maintenance Records

3.3.1 Identification of whether the complete valve or subassembly has been serviced and/or removed from the steam line.

3.3.2 Dates when equipment is:

i) Removed from service;
ii) Reworked;
iii) Retested;
iv) Reinstalled in Service.

3.3.3 Operating history will be recorded for prior service cycle. This will include the following information:

i) The number of power actuations, and date;
ii) The number of pressure actuations, date and cause of actuations;
iii) Leak detection device indications/signatures and history of these;
iv) Other events; such as whether the steam lines were flooded for reactor shutdown;
v) Ambient temperature, air and electrical supply condition.

3.3.4 Results of tests conducted prior to refurbishing, if applicable, will be recorded.

i) Performed where;
ii) Performed by;
iii) References used for cleaning, testing and refurbishing procedures;
iv) Extent of disassembly performed;
v) Details of any machining, rework or nondestructive examinations performed on components
vi) Post reassembly bench test details will refer to test report number and attach summary of results.

3.4 Unscheduled Maintenance Records

3.4.1 The following information will be maintained for unscheduled maintenance.

3.4.1.1 The reason why a valve is being removed from service.

3.4.1.2 The following information will be recorded about the operating history of the valve:

i) Has the valve malfunctioned in the past? If yes, a brief summary of past history will be provided.

ii) Has this valve caused trouble in the past? If yes, a brief summary of past history will be provided.

3.4.1.3 Record the results of diagnostic tests if and when performed prior to disassembly.

3.5 Analysis of Data

3.5.1 The data accumulated on this program may be stored manually or by electronic data processing in a manner which gives flexibility in data retrieval.

3.5.2 A summary of cumulative failure rates will be maintained for predominant failure modes and for overall failures. This summary will be updated annually.

Item 24 - Appendix H - II.C.3.B - Surveillance Capsules

This question 121.37

Paragraph 11.C.3B of Appendix H requires that four (4) surveillance capsules be included in the Surveillance Program. Provide technical justification for the fact that Shoreham Unit 1 Surveillance Program has three (3) rather than four (4) surveillance capsules.

Response

The Shoreham Surveillance Program was designed prior to the requirements under 10CFR50 Appendix H, and three (3) Surveillance capsules were provided. Under 10CFR50 January 1, 1980 Part 50, Appendix H, Section II Paragraph C3 Surveillance Program criteria Revised Withdrawl Schedule four capsules are required with the fourth capsule indicated as standby.

For Shoreham, three (3) capsule supports are available on the reactor vessel and it is no longer advisable to perform additional welding on the reactor vessel. Test coupons are available and it is proposed that a fourth capsule will be installed when the first capsule is removed. This will serve to continue to provide a standby capsule should one be needed. The withdrawl schedule will be in compliance with 10CFR50, Appendix H Section II.

Additional justification for this surveillance program in Shoreham is based on the fact that the weld material, which is limiting, is also used in the LaSalle #2 Surveillance Program. (See Response to Shoreham Question 121.36). Thus, between Shoreham and LaSalle 1, a total of seven (7) capsules (four (4) on Shoreham, three (3) on LaSalle) will be irridated to study the effects on the properties of limiting material.

Open Item 26 - Suppression Pool Bypass

1. With respect to the high pressure bypass leakage test, the way in which the vacuum breakers are to be included is presently under review by Shoreham and will be addressed separately. The acceptance criterion for the high pressure test given in SER Section 6.2.1.7 (10 percent of $A/\sqrt{K}$ = 0.05 ft^2) is unacceptable. Shoreham has demonstrated that the allowable $A/\sqrt{K}$ for breaks capable of producing large differential pressures on the drywell floor is at least an order of magnitude larger than 0.05 ft^2 (refer to FSAR Figure 6.2.1-23B). In SNRC-318 dated September 18, 1978, it was further demonstrated that even with an acceptance criterion of 10 percent of the large break capability, it would be impossible to establish the 35 psi structural acceptance test differential pressure across the drywell floor with the size compressors to be used unless the leakage across the floor were within acceptance values. Shoreham, therefore, considers the high pressure bypass leakage test to be a closed issue except for treatment of the vacuum breakers as noted above.

2. With respect to the actual small break capability, Shoreham has reviewed the possible reasons for the discrepancy between the NRC staff calculation (operator response time <15 minutes) and that done by Shoreham (operator response time = 26 minutes). It is believed that the model used by NRC-CSB is excessively conservative in the following ways:

   a. The NRC model uses a single control volume with an initial pressure of 30 psig. The operator response time is calculated from the time steam addition begins to the time the drywell design pressure is exceeded. The 30 psig used by NRC-CSB is approximately 6 psi higher than the wetwell pressure corresponding to complete air carryover for Shoreham (which, when exceeded, requires spray actuation). The effect of using the higher pressure is to shorten the available operator response time by approximately 5 minutes.

   b. The NRC model uses an 8 percent revaporization fraction which is based on large break, single-chamber containment data where the steam mole fraction exceeded that of the wetwell airspace during steam bypass. Since heat transfer is reduced by a relatively high air mole fraction, the quantity of condensate removed by the wetwell sinks during bypass is small compared to the large break, single-chamber containment data identified above. One would expect, however, that the revaporization fraction of this smaller quantity of condensate would be greater. Preliminary studies have shown virtually no effect on operator response time for revaporization fractions between 20 and 100 percent. Below 20 percent there is a moderate decrease in operator response time with decreasing revaporization fraction. Shoreham's bypass version of LOCTVS uses an equilibrum treatment of condensate which corresponds to 100 percent revaporization. Use of 8 percent revaporization shortens the available operator response time by an estimated 5-6 minutes.

3. With respect to the low pressure test acceptance criterion, Shoreham proposed a criterion of 20 percent of the small break bypass capability, but the NRC staff has remained adamant on a 10 percent criterion. The NRC position has the effect of introducing a 1,000 percent margin between the maximum expected response and the design capability of the plant. In view of this 10 percent acceptance criterion and the NRC position on revaporization fraction, Shoreham has performed a reanalysis of pool bypass which includes:

   - drywell and wetwell heat sinks (8 percent revaporization)
   - downcomer heat addition
   - heat and mass transfer between the airspace and the pool

   The results of this analysis are presented in the following section.

Shoreham Bypass Reanalysis

This analysis supercedes all previous bypass analyses for Shoreham. A revision to FSAR Section 6.2.1.3.6 and the response to NRC question 041.32 will be made consistent with the following information.

The reanalysis of bypass for Shoreham has been performed with the Stone & Webster computer code CONSBA (Containment Small Break Analysis). The results of this reanalysis must be considered preliminary at this time since documention and qualification of CONSBA will not be complete until July, 1981. However, the reactor model is that of CONTORT, a fully qualified computer code developed by Stone & Webster for analysis of pool temperature transients due to safety/relief valve (SRV) discharge. It includes models for SRV operation, high and low pressure ECCS operation, and feedwater. It does not include a level swell or bubble rise model and is, therefore, not suitable for prediction of maximum containment pressure response to large breaks. Modeling of high pressure ECCS and SRV operation permits calculation of long term depressurization/pool heatup effects which could previously be approximated in LOCTVS only thrugh the use of a suppression pool heat addition curve.

A relatively simple containment model has been combined with the CONTORT code to create CONSBA. The vent clearing and vent flow models are appropriate for the relatively small drywell floor differential pressures which charactrize small (and even moderately large steam) breaks. Drywell and wetwell heat sinks are available with revaporization fraction an input variable.

Drywell and wetwell heat sink data are the same as that given in Table 6.2.1-1 of the Shoreham FSAR. Heat and mass transfer between the airspace and the pool are modeled in the same manner as that described for horizontal vent containments (Mark III) in Section 7.3.2 of SWECO 8101 submittal by R.B. Bradbury (S&W) to J.R. Miller (NRC) on March 6, 1981 with the exception that the emissivity for radiant heat transfer from the airspace to the pool is set at 0.8 rather than 1.0. No explicit model has been included for downcomer heat addition, but the effect has been calculated iteratively and included by means of a net wetwell airspace heat addition curve.

For consistency with SBA pool temperature transients presented in Section 10 of the Shoreham Plant Design Assessment for Hydrodynamic Loads (DAR), feedwater is used as the makeup source rather than ECCS. This represents a significant conservatism for three reasons:

1. More energy in the reactor/containment system.

2. Higher pool level increasing vent submergence and drywell floor differential pressure.

3. Higher pool level compressing the wetwell airspace.

The objective of the reanalysis is to determine the maximum value of $A/\sqrt{K}$ that will permit a spray delay time of 30 minutes (1,800 sec) when considering the worst case break size. In determining the critical break size, it is necessary to assume a value of $A/\sqrt{K}$. If the value assumed is reasonably close to the final result, it is not necessary to repeat the iteration. A value of $A/\sqrt{K}$=0.16 ft^2 was chosen to study the effects of break size since it approximates the final expected result of including heat and mass transfer from the airspace to the pool. In performing this critical break size study, heat transfer from the downcomers to the airspace and the effect of miscellaneous steel heat sinks in the wetwell were ignored. Neither is a large effect and both must be included in the analysis by manual calcultion and application of heat addition (or subtraction) curves to the wetwell aispace state calculation.

Figure 26-1 provides the results of the break size study. Note that there is very little variation in drywell pressure at 1,800 seconds with break size for a wide range of breaks. This is primarily a consequence of hot feedwater addition which tends to limit depressurization of the reactor at the end of the transient. (For example, for the 1.0 ft^2 break, the feedwater temperature at the end of the run is approximtely 317°F corresponding to a saturation pressure of approximtely 86 psia). The 1.0 ft^2 break is considered the limiting case.

Figure 26-2 provides the reactor, drywell, and wetwell pressure for the 1.0 ft^2 steam break described above. In this figure, the effects of downcomer heat addition and wetwell miscellaneous steel heat sinks have been included. As noted previously, the Shoreham containment heat sinks are described in detail in FSAR Table 6.2.1-1. It is assumed that the heat transfer rates to the wetwell airspace from the downcomer and from the wetwell airspace to the miscellaneous steel heat sinks both decrease from t=0 to 1,800 sec. Also it is assumed that the miscellaneous steel is in equilibrium with the wetwell atmosphere at t=1,800 sec. The net result is uniform heat rate addition to the wetwell airspace.

In calculting heat transfer rates from the downcomer to the wetwell airspace a convective heat transfer coefficient of 0.8 Btu/hr-ft^2-°F was used and an emissivity of 0.9 was employed for radiation. An average wetwell temperature of 225°F was initially assumed and was verified in the final analysis.

Note that no thermal stratfication in the pool is considered. Intermittant condensation at the downcomer vent during the relatiely low mass flow characteristic of small breaks has been shown in full scale tests to be an excellent mixing mechanism. Temperatures at the pool surface are generally less than the mass average temperature of the pool.

Tables 26-1 through 26-6 provide mass and energy balance information for the Reactor Coolant, Suppression Pool, Drywell Atmosphere, Wetwell Atmosphere, Liquid on the Drywell Floor and the overall containment, respectively.

Shoreham considers the information presented above to be adequate for a complete review of steam bypass for Shoreham. Although the large conservatism of ignoring heat and mass transfer from the aispace to the pool has been extracted from previous analyses with the effect now considered explicity, there remains the following sources of conservatism:

1. Feedwater addition
2. All-steam bypass
3. Revaporization of wetwell heat sink condensate limited to 8 percent.

These conservatisms seem more than adequate in view of the 1,000 percent margin applied to the results.

In summary, Shoreham's analysis of pool bypass has shown the following:

1. Performing the drywell floor structural acceptance test constitutes an acceptable high pressure steam bypass test as long as the total compressor flow to be used is approximately 5000 SCFM or less. Treatment of the vacuum breakers (exposure to test pressure) is to be addressed separately.
2. The NRC acceptance criterion for the low pressure test (10 percent of the largest $A/\sqrt{K}$ that will permit at least 30 minutes operator delay for manual spray actuation at the worst case break size) is acceptable to Shoreham as long as all relevant effects, including heat and mass transfer from the air space to the pool, are included in the analysis. For Shoreham, the low pressure test acceptance criteria will be 10 percent of $A/\sqrt{K}$ = 0.16 ft^2.

S?022FB? +STEAM BYPASS BREAK AREA= 1.0 FT2, A/K=.16, CONT. SPRAYS DELAYED — CONSBA 15 MAY 1991 14:22:59 PAGE 1
B & W ENG CORP NU-169 VERSION 0T, LEVEL 0T CONTAINMENT SMALL BREAK ACCIDENT CODE — CREATED 01.125 03:03:04 UNIT 10

******** MASS AND HEAT BALANCE ********

(UNITS: TIME IS IN SEC, MASS IS IN LBM, ENERGY IS IN BTU)

TIME=	0.0	450.0	900.0	1350.0	1800.0
CONTROL VOLUME: REACTOR COOLANT					
TOTAL MASS=	494107.8	575923.0	594062.2	594139.1	598032.8
TOTAL INTERNAL ENERGY=	271612220.0	182333592.6	163173453.9	157066262.3	158983691.1
INTEGRATED FLOW AND ENERGY INTO RV:					
COASTDOWN HEAT	0.0	2494592.3	2494592.3	2494592.3	2484592.3
DECAY HEAT	0.0	42996007.4	73361298.7	90280008.8	120328048.0
METAL HEAT	0.0	39732478.2	59675596.5	67329366.7	68425908.9
FW-MASS	0.0	342346.9	444341.9	487776.9	516765.9
FW-ENERGY	0.0	126755781.1	156056186.1	160527731.1	176842121.1
CRD-MASS	0.0	0.0	0.0	0.0	0.0
CRD-ENERGY	0.0	0.0	0.0	0.0	0.0
HPCS MASS FROM CST AND PL	0.0	0.0	0.0	0.0	0.0
HPCS-ENERGY FROM CST AND PL	0.0	0.0	0.0	0.0	0.0
HPCS PUMP HEAT	0.0	0.0	0.0	0.0	0.0
HPCI-MASS FROM CST AND PL	0.0	0.0	0.0	0.0	0.0
HPCI-ENERGY FROM CST AND PL	0.0	0.0	0.0	0.0	0.0
HPCI PUMP HEAT	0.0	0.0	0.0	0.0	0.0
LPCS MASS	0.0	0.0	0.0	0.0	0.0
LPCS-ENERGY	0.0	0.0	0.0	0.0	0.0
LPCS PUMP HEAT	0.0	0.0	0.0	0.0	0.0
LPCI-MASS	0.0	0.0	0.0	0.0	0.0
LPCI-ENERGY	0.0	0.0	0.0	0.0	0.0
LPCI PUMP HEAT	0.0	0.0	0.0	0.0	0.0
RCIC-MASS FROM CST AND PL	0.0	0.0	0.0	0.0	0.0
RCIC-ENERGY FROM CST AND PL	0.0	0.0	0.0	0.0	0.0
RCIC PUMP HEAT	0.0	0.0	0.0	0.0	0.0
RHR PUMP HEAT SHUTDOWN COOLING MODE	0.0	0.0	0.0	0.0	0.0
INPUT SPECIFIED FLOW CURVE	0.0	0.0	0.0	0.0	0.0
INPUT SPECIFIED ENERGY CURVE	0.0	0.0	0.0	0.0	0.0
TOTAL MASS INTO RV	0.0	342346.9	444341.9	487796.7	516766.9
TOTAL ENERGY INTO RV	0.0	212169357.1	291577633.7	336621690.9	368080571.2
INTEGRATED FLOW AND ENERGY OUT OF RV:					
RHR HX SHUTDOWN COOLING MODE	0.0	0.0	0.0	0.0	0.0
MAIN STEAM MASS	0.0	4191.2	4191.2	4191.2	4191.2
MAIN STEAM ENERGY	0.0	4999181.6	4999181.6	4999181.6	4999181.6
MANUAL-SRV MASS	0.0	0.0	0.0	0.0	0.0
MANUAL-SRV-ENERGY	0.0	0.0	0.0	0.0	0.0
AUTO-SRV-MASS	0.0	0.0	0.0	0.0	0.0
AUTO-SRV-ENERGY	0.0	0.0	0.0	0.0	0.0
HPCI-MASS	0.0	0.0	0.0	0.0	0.0
HPCI-ENERGY	0.0	0.0	0.0	0.0	0.0
RCIC MASS	0.0	0.0	0.0	0.0	0.0
RCIC-ENERGY	0.0	0.0	0.0	0.0	0.0
STEAM BLOWDOWN MASS	0.0	247119.7	330256.3	373654.4	390730.7
STEAM BLOWDOWN ENERGY	0.0	296440313.6	395017224.9	446160493.8	475709927.3

LIQUID BLOWDOWN MASS	0.0	0.0	0.0	0.0	0.0
LIQUID BLOWDOWN ENERGY	0.0	0.0	0.0	0.0	0.0
TOTAL MASS OUT OF RV	0.0	251310.9	339947.5	377095.6	402921.9
TOTAL ENERGY OUT OF RV	0.0	301047455.2	400016400.5	451167665.4	400709100.9
TOTAL OFFSET IN RV MASS BALANCE EVALUATED FROM 0 SEC		-0.0000	-0.0000	-0.0000	-0.0000
TOTAL OFFSET IN RV ENERGY BALANCE EVALUATED FROM 0 SEC		-0.0000	-0.0000	-0.0000	-0.0000

SCSSTF82 STEAM BYPASS BREAK AREA= 1.0 FT2, A/K=.16, CONT. SPRAYS DELAYED CONSEA 15 MAY 1981 19:22:59 PAGE 2
S & W ENG CCMP NU-169 VERSION 07, LEVEL 07 CONTAINMENT SMALL BREAK ACCIDENT CODE CREATED 81.125 08:03:04 UNIT 10

*** MASS AND HEAT BALANCE ***

(UNITS: TIME IS IN SEC, MASS IS IN LBM, ENERGY IS IN BTU)

TIME=	0.0	450.0	900.0	1350.0	1800.0
CONTROL VOLUME: SUPPRESSION POOL					
TOTAL MASS=	5055443.7	5223163.9	5287831.9	5323062.7	5343458.3
TOTAL INTERNAL ENERGY=	2932933410.9	409635945.3	559246010.3	593651674.7	622375654.7
INTEGRATED FLOW AND ENERGY INTO POOL:					
MANUAL SRV MASS	0.0	0.0	0.0	0.0	0.0
MANUAL SRV ENERGY	0.0	0.0	0.0	0.0	0.0
AUTO SRV MASS	0.0	0.0	0.0	0.0	0.0
AUTO SRV ENERGY	0.0	0.0	0.0	0.0	0.0
HPCS-MASS FROM CST	0.0	0.0	0.0	0.0	0.0
HPCS-ENERGY FROM CST	0.0	0.0	0.0	0.0	0.0
HPCS-PUMP HEAT	0.0	0.0	0.0	0.0	0.0
HPCI-MASS	0.0	0.0	0.0	0.0	0.0
HPCI-ENERGY	0.0	0.0	0.0	0.0	0.0
LPCS-PUMP HEAT	0.0	0.0	0.0	0.0	0.0
LPCI-PUMP HEAT	0.0	0.0	0.0	0.0	0.0
RCIC-MASS	0.0	0.0	0.0	0.0	0.0
RCIC-ENERGY	0.0	0.0	0.0	0.0	0.0
RHR PUMP HEAT PL COOLING MODE	0.0	0.0	0.0	0.0	0.0
CONDENSATE FROM HH ATM	0.0	0.0	0.0	0.0	0.0
ENERGY OF CONDENSATE FROM HH ATM	0.0	0.0	0.0	0.0	0.0
DH LTD OVERFLOW THRU VENT	0.0	0.0	0.0	0.0	0.0
ENERGY OF DH LTD OVERFLOW THRU VENT	0.0	0.0	0.0	0.0	0.0
VENT FLOW-VAPOR	0.0	165092.6	225077.0	255257.3	270506.8
ENERGY OF VENT FLOW-VAPOR	0.0	196240900.4	266054502.4	302394702.1	320359309.7
CONDENSATE FROM HH SINK	0.0	2136.5	5064.9	9792.9	13641.7
ENERGY OF CONDENSATE FROM HH SINK	0.0	302230.0	895189.0	1551499.0	2215533.2
CONDENSATE FROM HH COOLER	0.0	0.0	0.0	0.0	0.0
ENERGY OF CONDENSATE FROM HH COOLER	0.0	0.0	0.0	0.0	0.0
CONDENSATION AT PL SURFACE	0.0	491.2	1445.5	2571.0	3066.2
ENERGY OF CONDENSATE AT PL SURFACE	0.0	572659.4	1601065.9	2902701.2	4475741.3
UNCOOLED SPRAY FLOW	0.0	0.0	0.0	0.0	0.0
ENERGY OF HH UNCOOLED SPRAY FLOW	0.0	0.0	0.0	0.0	0.0
INPUT SPECIFIED FLOW CURVE	0.0	0.0	0.0	0.0	0.0
INPUT SPECIFIED ENERGY CURVE	0.0	0.0	0.0	0.0	0.0
HEAT TRANSFER FROM VENT (AIR) FLOW	0.0	650204.9	650302.4	650309.3	650310.7
TOTAL MASS INTO POOL	0.0	167720.2	232390.3	267619.0	288014.6
TOTAL ENERGY INTO POOL	0.0	197774003.7	270090059.7	307507292.5	327708095.0
INTEGRATED FLOW AND ENERGY OUT OF POOL:					
RHR HX POOL COOLING MODE	0.0	0.0	0.0	0.0	0.0
HPCS-MASS	0.0	0.0	0.0	0.0	0.0
HPCS-ENERGY	0.0	0.0	0.0	0.0	0.0
HPCI-MASS	0.0	0.0	0.0	0.0	0.0
HPCI-ENERGY	0.0	0.0	0.0	0.0	0.0
LPCS-MASS	0.0	0.0	0.0	0.0	0.0
LPCS-ENERGY	0.0	0.0	0.0	0.0	0.0

POOR ORIGINAL

TABLE 26-2

LPCI-MASS	0.0	0.0	0.0	0.0	0.0
LPCI-ENERGY	0.0	0.0	0.0	0.0	0.0
RCIC-MASS	0.0	0.0	0.0	0.0	0.0
RCIC-ENERGY	0.0	0.0	0.0	0.0	0.0
EVAP AT POOL SURFACE	0.0	0.0	0.0	0.0	0.0
ENERGY OF EVAP AT POOL SURFACE	0.0	0.0	0.0	0.0	0.0
DW SPRAY FLOW	0.0	0.0	0.0	0.0	0.0
ENERGY OF DW SPRAY FLOW	0.0	0.0	0.0	0.0	0.0
WW SPRAY FLOW	0.0	0.0	0.0	0.0	0.0
ENERGY OF WW SPRAY FLOW	0.0	0.0	0.0	0.0	0.0
NATURAL CONVECTION FORM PL TO WW	0.0	-66704.0	-123003.2	-160050.7	-200965.3
RADIATION FROM PL TO WW	0.0	-92320.7	-167633.0	-229064.4	-205627.1
PL HEAT SINKS	0.0	2582920.0	5430104.4	7620951.7	9122751.5
TOTAL MASS OUT OF POOL	0.0	0.0	0.0	0.0	0.0
TOTAL ENERGY OUT OF POOL	0.0	2833457.3	5139468.2	7231036.6	8620659.2
TOTAL OFFSET IN PL MASS BALANCE EVALUATED FROM 0 SEC		-0.0000	-0.0000	-0.0000	-0.0000
TOTAL OFFSET IN PL ENERGY BALANCE EVALUATED FROM 0 SEC		-0.0000	-0.0001	-0.0001	-0.0001

POOR ORIGINAL

** MASS AND HEAT BALANCE **

(UNITS: TIME IS IN SEC, MASS IS IN LBM, ENERGY IS IN BTU)

TIME=	0.0	450.0	900.0	1350.0	1800.0
CONTROL VOLUME: DRYWELL ATMOSPHERE					
TOTAL MASS=	13156.6	25755.4	26491.7	26947.5	27548.9
TOTAL INTERNAL ENERGY=	1731782.9	28260474.2	29082704.0	29590440.2	30263255.1
INTEGRATED FLOW AND ENERGY INTO DRYWELL ATM:					
STEAM BLOWDOWN	0.0	247119.7	330256.3	373654.4	398730.7
ENERGY OF STEAM BLOWDOWN	0.0	296448313.6	395017226.9	446168403.8	475709927.3
LIQUID BLOWDOWN	0.0	0.0	0.0	0.0	0.0
ENERGY OF LIQUID BLOWDOWN	0.0	0.0	0.0	0.0	0.0
VACUUM BREAKER FLOW-VAPOR	0.0	0.0	0.0	0.0	0.0
ENERGY OF VACUUM BREAKER FLOW-VAPOR	0.0	0.0	0.0	0.0	0.0
VACUUM BREAKER FLOW-AIR	0.0	0.0	0.0	0.0	0.0
ENERGY OF VACUUM BREAKER FLOW-AIR	0.0	0.0	0.0	0.0	0.0
SPRAY FLOW MIXED WITH ATM	0.0	0.0	0.0	0.0	0.0
ENERGY OF SPRAY FLOW MIXED WITH ATM	0.0	0.0	0.0	0.0	0.0
INPUT SPECIFIED FLOW CURVE	0.0	0.0	0.0	0.0	0.0
INPUT SPECIFIED ENERGY CURVE	0.0	0.0	0.0	0.0	0.0
TOTAL MASS INTO DRYWELL ATM	0.0	247119.7	330256.3	373654.4	398730.7
TOTAL ENERGY INTO DRYWELL ATM	0.0	296448313.6	395017226.9	446168403.8	475709927.3
INTEGRATED FLOW AND ENERGY OUT OF DRYWELL ATM:					
CONDENSATE FROM ATM	0.0	67.7	676.2	1102.1	1437.7
ENERGY OF CONDENSATE FROM ATM	0.0	17556.0	176283.0	300276.3	376374.2
VENT FLOW-VAPOR	0.0	165092.6	225077.8	255257.3	270506.8
ENERGY OF VENT FLOW-VAPOR	0.0	196240900.4	266059502.4	302374782.1	320359309.7
VENT FLOW-AIR	0.0	12782.9	12785.4	12785.6	12785.7
ENERGY OF VENT FLOW-AIR	0.0	2353922.0	2354375.5	2354407.0	2354416.0
CONDENSATE FROM DW HT SINK	0.0	51591.6	67977.5	74715.1	78105.9
ENERGY OF CONDENSATE FROM DW HT SINK	0.0	12900605.0	17262116.0	19029473.5	19923680.7
CONDENSTAE FROM DW COOLER	0.0	0.0	0.0	0.0	0.0
ENERGY OF CONDENSTAE FROM DW COOLER	0.0	0.0	0.0	0.0	0.0
BYPASS FLOW-VAPOR	0.0	4996.2	10404.2	15923.4	21502.3
ENERGY OF BYPASS FLOW-VAPOR	0.0	5912766.9	12279102.5	18779010.0	25351476.5
BYPASS FLOW-AIR	0.0	0.0	0.0	0.0	0.0
ENERGY OF BYPASS FLOW-AIR	0.0	0.0	0.0	0.0	0.0
ENERGY REMOVED BY DW COOLER	0.0	0.0	0.0	0.0	0.0
DW HEAT SINKS	0.0	52405672.0	68739765.6	75443174.8	78813189.1
TOTAL MASS OUT OF DRYWELL ATM	0.0	234520.9	316921.2	359063.5	384338.4
TOTAL ENERGY OUT OF DRYWELL ATM	0.0	269919422.3	367666225.1	418309026.5	447170455.1
TOTAL OFFSET IN DW MASS BALANCE EVALUATED FROM 0 SEC		-0.0000	-0.0000	-0.0000	-0.0000
TOTAL OFFSET IN DW ENERGY BALANCE EVALUATED FROM 0 SEC		0.0000	0.0000	0.0000	0.0000

POOR ORIGINAL

SUBCASE: STEAM BYPASS BREAK AREA= 1.0 FT2, A/K=.16, CONT. SPRAYS DELAYED CONSBA 15 MAY 1981 10:22:59 PAGE 9
S & W ENGR CORP NU-169 VERSION 01, LEVEL 01 CONTAINMENT SMALL BREAK ACCIDENT CODE CREATED 81.125 08:03:04 UNIT 18

********************** MASS AND HEAT BALANCE **********************

(UNITS: TIME IS IN SEC, MASS IS IN LBM, ENERGY IS IN BTU)

TIME=	0.0	450.0	900.0	1350.0	1800.0
CONTROL VOLUME: WETWELL ATMOSPHERE					
TOTAL MASS=	10155.4	25306.0	26034.6	26502.8	26935.6
TOTAL INTERNAL ENERGY=	1229250.3	5610094.2	6320023.6	6793097.1	7254280.9
INTEGRATED FLOW AND ENERGY INTO WETWELL ATM:					
EVAP FROM PL SURFACE	0.0	0.0	0.0	0.0	0.0
ENERGY OF EVAP FROM PL SURFACE	0.0	0.0	0.0	0.0	0.0
BYPASS FLOW-VAPOR	0.0	4996.2	10404.2	15923.4	21502.3
ENERGY OF BYPASS FLOW-VAPOR	0.0	5912766.9	12279102.5	18779010.0	25351476.5
BYPASS FLOW-AIR	0.0	0.0	0.0	0.0	0.0
ENERGY OF BYPASS FLOW-AIR	0.0	0.0	0.0	0.0	0.0
VENT FLOW-AIR	0.0	12702.9	12705.4	12705.6	12705.7
ENERGY OF VENT FLOW-AIR	0.0	1695717.0	1696073.1	1696100.5	1696106.1
SPRAY FLOW MIXED WITH ATM	0.0	0.0	0.0	0.0	0.0
ENERGY OF SPRAY FLOW MIXED WITH ATM	0.0	0.0	0.0	0.0	0.0
INPUT SPECIFIED FLOW CURVE	0.0	0.0	0.0	0.0	0.0
INPUT SPECIFIED ENERGY CURVE	0.0	249405.5	498390.6	747375.6	995940.1
NATURAL CONVECTION FROM PL SURFACE	0.0	-66704.0	-123003.2	-160050.7	-208465.3
RADIATION FROM PL SURFACE	0.0	-82328.7	-167633.0	-229864.4	-285627.1
TOTAL MASS INTO WETWELL ATM	0.0	17779.1	23109.6	28709.0	34209.0
TOTAL ENERGY INTO WETWELL ATM	0.0	7701056.7	14183010.1	20824570.9	27549430.3
INTEGRATED FLOW AND ENERGY OUT OF WETWELL ATM:					
CONDENSATE FROM ATM	0.0	0.0	0.0	0.0	0.0
ENERGY OF CONDENSATE FROM ATM	0.0	0.0	0.0	0.0	0.0
VACUUM BREAKER FLOW-VAPOR	0.0	0.0	0.0	0.0	0.0
ENERGY OF VACUUM BREAKER FLOW-VAPOR	0.0	0.0	0.0	0.0	0.0
VACUUM BREAKER FLOW-AIR	0.0	0.0	0.0	0.0	0.0
ENERGY OF VACUUM BREAKER FLOW-AIR	0.0	0.0	0.0	0.0	0.0
CONDENSATE FROM WW HT SINKS	0.0	2136.5	5964.9	9789.9	13641.7
ENERGY OF CONDENSATE FROM WW HT SINKS	0.0	302230.0	896169.0	1551499.8	2215533.2
CONDENSATE FROM WW COOLER	0.0	0.0	0.0	0.0	0.0
ENERGY OF CONDENSATE FROM WW COOLER	0.0	0.0	0.0	0.0	0.0
CONDENSATION AT PL SURFACE	0.0	491.2	1445.5	2571.8	3866.2
ENERGY OF CONDENSATION AT PL SURFACE	0.0	572660.4	1681065.9	2982701.2	4475741.3
ENERGY REMOVED BY WW COOLER	0.0	0.0	0.0	0.0	0.0
WW HEAT SINKS	0.0	2445332.5	6506709.8	10725731.1	14833133.1
TOTAL MASS OUT OF WETWELL ATM	0.0	2627.7	7310.5	12361.7	17507.8
TOTAL ENERGY OUT OF WETWELL ATM	0.0	3320230.9	9084244.8	15259932.1	21524407.6
TOTAL OFFSET IN WW MASS BALANCE EVALUATED FROM 0 SEC		0.0000	0.0000	0.0000	-0.0000
TOTAL OFFSET IN WW ENERGY BALANCE EVALUATED FROM 0 SEC		0.0000	-0.0000	0.0000	0.0000

POOR ORIGINAL

********** MASS AND HEAT BALANCE **********

(UNITS: TIME IS IN SEC, MASS IS IN LBM, ENERGY IS IN BTU)

TIME=	0.0	450.0	900.0	1350.0	1800.0
CONTROL VOLUME: LIQUID ON DRYWELL FLOOR					
TOTAL MASS=	0.0	51649.3	68653.9	75897.2	79543.6
TOTAL INTERNAL ENERGY=	0.0	13006161.0	17430399.0	19338449.8	20300062.9
INTEGRATED FLOW AND ENERGY INTO CONTROL VOLUME:					
CONDENSATE FROM DW ATM	0.0	67.7	676.2	1182.1	1437.7
ENERGY OF CONDENSATE FROM ATM	0.0	17556.0	176283.0	308976.3	376374.2
CONDENSATE FROM DW HT SINK	0.0	51531.6	67977.5	74715.1	78105.9
ENERGY OF CONDENSATE FROM DW HT SINK	0.0	12988605.0	17262116.0	19029473.5	19923688.7
CONDENSATE FROM DW COOLER	0.0	0.0	0.0	0.0	0.0
ENERGY OF CONDENSATE FROM DW COOLER	0.0	0.0	0.0	0.0	0.0
DW UNMIXED SPRAY FLOW	0.0	0.0	0.0	0.0	0.0
ENERGY OF DW UNMIXED SPRAY FLOW	0.0	0.0	0.0	0.0	0.0
TOTAL MASS INTO DRYWELL LIQ	0.0	51649.3	68653.9	75897.2	79543.6
TOTAL ENERGY INTO DRYWELL LIQ	0.0	13006161.0	17430399.0	19338447.8	20300062.9
INTEGRATED FLOW AND ENERGY OUT OF CONTROL VOLUME:					
OVERFLOW THRU VENT	0.0	0.0	0.0	0.0	0.0
ENERGY OF OVERFLOW THRU VENT	0.0	0.0	0.0	0.0	0.0
TOTAL MASS OUT OF DRYWELL LIQ	0.0	0.0	0.0	0.0	0.0
TOTAL ENERGY OUT OF DRYWELL LIQ	0.0	0.0	0.0	0.0	0.0
TOTAL OFFSET IN CV MASS BALANCE EVALUATED FROM 0 SEC		-0.0000	-0.0000	0.0000	0.0000
TOTAL OFFSET IN CV ENERGY BALANCE EVALUATED FROM 0 SEC		-0.0000	-0.0000	-0.0000	-0.0000

POOR ORIGINAL

TABLE 26-6

SP0CCF02 *STEAM BYPASS BREAK AREA= 1.0 FT2, A/K=.16, CONT. SPRAYS DELAYED   CONSBA 15 MAY 1981 14:22:57 PAGE 6
S & W ENGR CORP NU-169 VERSION 0T, LEVEL 0T CONTAINMENT SMALL BREAK ACCIDENT CODE   CREATED 81.125 00:03:04 UNIT 10

** MASS AND HEAT BALANCE **

(UNITS: TIME IS IN SEC, MASS IS IN LBM, ENERGY IS IN BTU)

TIME=	0.0	450.0	900.0	1350.0	1800.0
CONTROL VOLUME: SUPPRESSION POOL, REACTOR VESSEL, DRYWELL ATM, WETWELL ATM AND LIQUID ON DRYWELL FLOOR					
TOTAL MASS=	5562943.5	5901697.2	6003024.2	6046549.2	6075519.2
TOTAL INTERNAL ENERGY=	567060680.0	717047077.3	774260671.7	806440724.2	829176944.7
INTEGRATED FLOW AND ENERGY INTO CONTROL VOLUME:					
CONSTDOWN HEAT	0.0	2404592.3	2404592.3	2404592.3	2404592.3
DECAY HEAT	0.0	42976097.4	73341290.7	90280003.0	120320040.0
METAL HEAT	0.0	39732470.2	59675596.5	67329366.7	68425003.9
FW-MASS	0.0	342946.9	444341.9	487796.9	516766.9
FW-ENERGY	0.0	126955701.1	156056146.1	168527731.1	176842121.1
CRD-MASS	0.0	0.0	0.0	0.0	0.0
CRD-ENERGY	0.0	0.0	0.0	0.0	0.0
HPCS-MASS FROM CST	0.0	0.0	0.0	0.0	0.0
HPCS-ENERGY FROM CST	0.0	0.0	0.0	0.0	0.0
HPCI-MASS FROM CST	0.0	0.0	0.0	0.0	0.0
HPCI-ENERGY FROM CST	0.0	0.0	0.0	0.0	0.0
RCIC-MASS FROM CST	0.0	0.0	0.0	0.0	0.0
RCIC-ENERGY FROM CST	0.0	0.0	0.0	0.0	0.0
HPCS-PUMP HEAT	0.0	0.0	0.0	0.0	0.0
HPCI-PUMP HEAT	0.0	0.0	0.0	0.0	0.0
LPCS-PUMP HEAT	0.0	0.0	0.0	0.0	0.0
LPCI-PUMP HEAT	0.0	0.0	0.0	0.0	0.0
RCIC-PUMP HEAT	0.0	0.0	0.0	0.0	0.0
RHR PUMP HEAT SHUTDOWN COOLING MODE	0.0	0.0	0.0	0.0	0.0
RHR PUMP HEAT POOL COOLING MODE	0.0	0.0	0.0	0.0	0.0
FLOW CURVE INTO RV	0.0	0.0	0.0	0.0	0.0
ENERGY CURVE INTO RV	0.0	0.0	0.0	0.0	0.0
FLOW CURVE INTO PL	0.0	0.0	0.0	0.0	0.0
ENERGY CURVE INTO PL	0.0	0.0	0.0	0.0	0.0
FLOW CURVE INTO DW ATM	0.0	0.0	0.0	0.0	0.0
ENERGY CURVE INTO DW ATM	0.0	0.0	0.0	0.0	0.0
FLOW CURVE INTO WW ATM	0.0	0.0	0.0	0.0	0.0
ENERGY CURVE INTO WW ATM	0.0	249405.5	420390.6	747375.6	995940.1
DW SPRAY PUMP HEAT	0.0	0.0	0.0	0.0	0.0
WW SPRAY PUMP HEAT	0.0	0.0	0.0	0.0	0.0
PUMP HEAT WHEN DW WW SPRAY OVERLAP	0.0	0.0	0.0	0.0	0.0
TOTAL MASS INTO CONTROL VOL	0.0	342946.9	444341.9	487796.9	516766.9
TOTAL ENERGY INTO CONTROL VOL	0.0	212410264.6	292076024.3	337369074.5	369076511.3
INTEGRATED FLOW AND ENERGY OUT OF CONTROL VOL:					
RHR HX POOL COOLING MODE	0.0	0.0	0.0	0.0	0.0
RHR HX RV SHUTDOWN COOLING MODE	0.0	0.0	0.0	0.0	0.0
MAIN STEAM MASS	0.0	4191.2	4191.2	4191.2	4191.2
MAIN STEAM ENERGY	0.0	4999101.6	4999101.6	4999101.6	4999101.6
HPCI TURBINE LOSS	0.0	0.0	0.0	0.0	0.0
RCIC TURBINE LOSS	0.0	0.0	0.0	0.0	0.0

POOR ORIGINAL

HEAT REMOVED BY DH AIR COOLER	0.0	0.0	0.0	0.0	0.0
HEAT REMOVED BY IH AIR COOLER	0.0	0.0	0.0	0.0	0.0
DH SPRAY HEAT EXCHANGER	0.0	0.0	0.0	0.0	0.0
FH SPRAY HEAT EXCHANGER	0.0	0.0	0.0	0.0	0.0
DH I IH OVERLAP SPRAY HEAT EXCHANGER	0.0	0.0	0.0	0.0	0.0
DH HEAT SINKS	0.0	52405972.9	60739765.6	75943174.8	78013107.1
IH HEAT SINKS	0.0	2495352.5	6564982.8	10725731.1	14033133.1
FH HEAT SINKS	0.0	2509658.0	5530109.4	7628951.7	9122751.5
TOTAL MASS OUT OF CONTROL VOLUME	0.0	4191.2	4191.2	4191.2	4191.2
TOTAL ENERGY OUT OF CONTROL VOLUME	0.0	62439876.1	85376091.4	98797039.2	107768255.4
TOTAL OFFSET IN CV MASS BALANCE EVALUATED FROM 0 SEC		-0.0000	-0.0000	-0.0000	-0.0000
TOTAL OFFSET IN CV ENERGY BALANCE EVALUATED FROM 0 SEC		-0.0000	-0.0001	-0.0001	-0.0001

POOR ORIGINAL

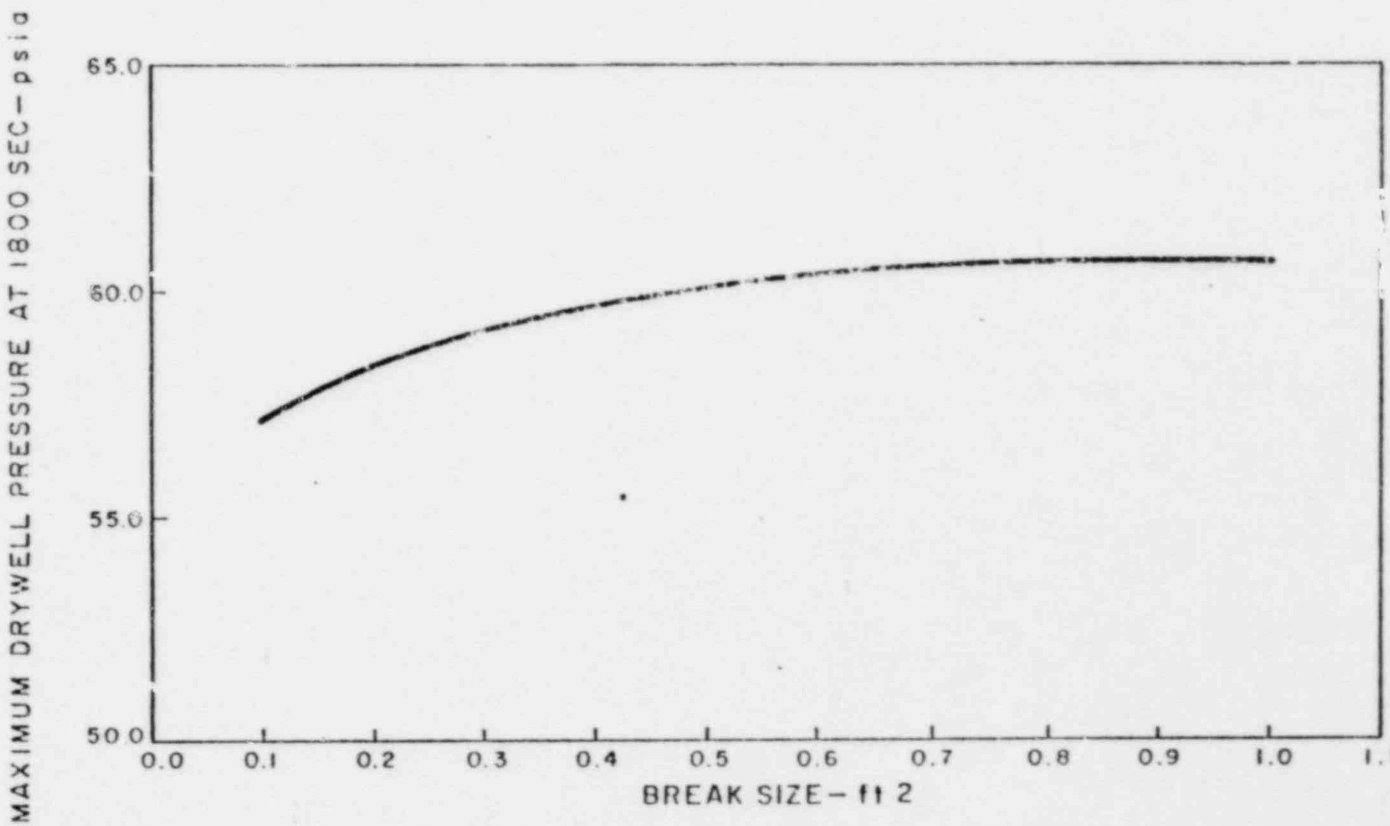
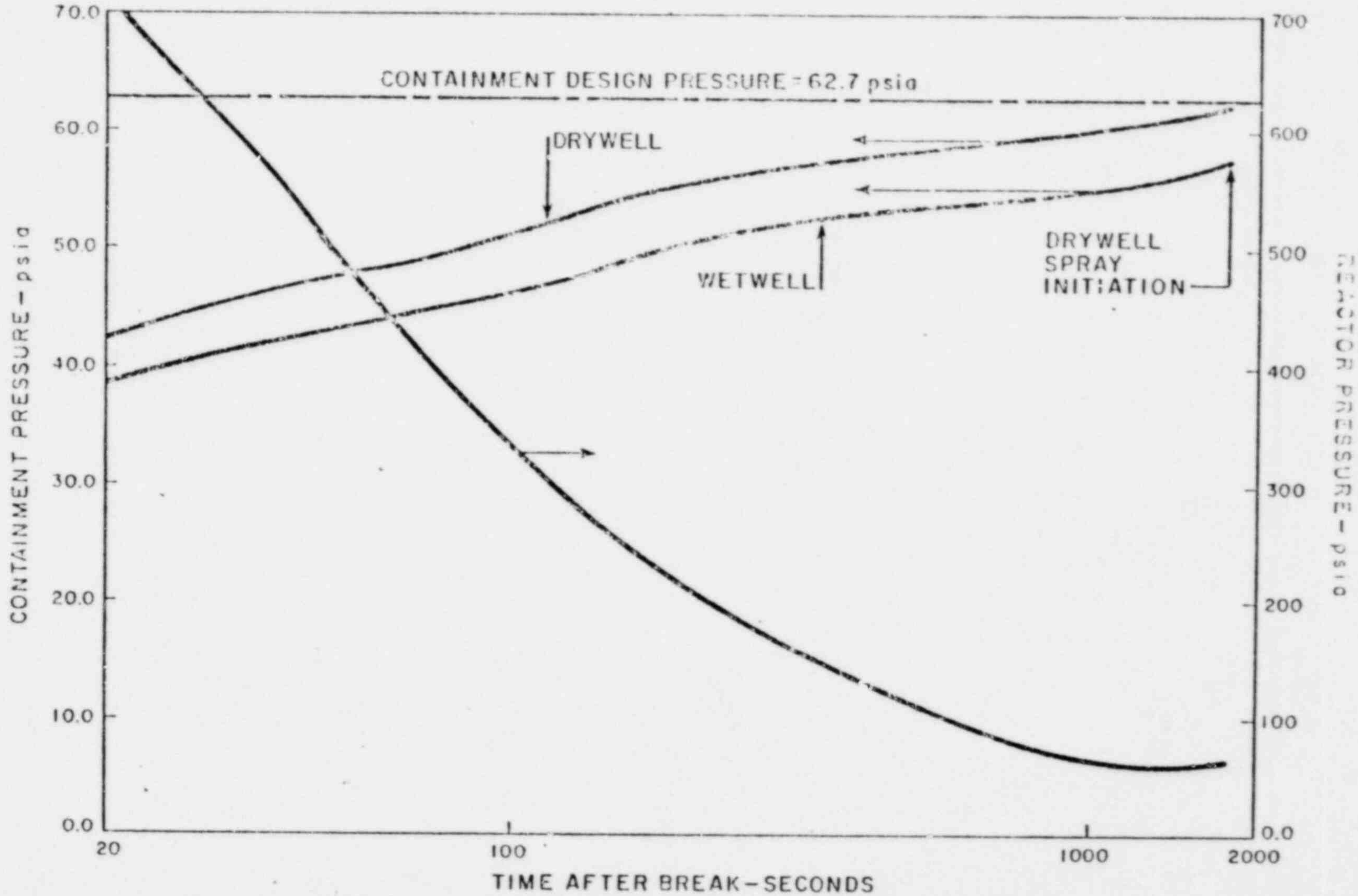


FIGURE 26-1
MAXIMUM DRYWELL PRESSURE
AT 1800 SEC VS. BREAK SIZE
$A/\sqrt{k}$ = 0.16 ft 2-NO WETWELL THIN
STEEL OR DOWNCOMER HEAT TRANSFER



NOTE:

WETWELL THIN STEEL AND DOWNCOMER HEAT TRANSFER INCLUDED

FIGURE 26-2
CONTAINMENT AND REACTOR PRESSURE VS. TIME
BREAK AREA = 1.0 ft 2, $A/\sqrt{k}$ = 0.16 ft 2

Item 27 - Steam Condensation Downcomer Lateral Loads

As identified in DAR Section 4.2.5.1, Shoreham has performed a static single vent, a dynamic single vent and a dynamic multivent analysis of downcomer lateral loads due to steam condensation with the bracing located at Elevation 27'9". In all cases the results are acceptable. NRC staff has not yet completed the review of the dynamic methodology proposed by the Mk II Owners Group and used in the Shoreham dynamic analyses. Until a satisfactory review is complete, the only acceptable basis for downcomer lateral load assessment is that presented in NUREG-0487, the static load definition. Shoreham hereby commits to perform a static multivent analysis in accordance with NUREG-0487 with the results to be submitted prior to fuel load in the event that a NRC approved dynamic multivent method is not available by January 1, 1982.

Item 28 - Steam Condensation Oscillation and Chugging Loads

The condensation oscillation (CO) load definition used by Shoreham in its confirmatory program is the Mk II program generic load. Based on PSD comparisons, this load bounds the interim CO load accepted by NRC in NUREG-0487, Supplement #2.

The chugging load definition used by Shoreham in its confirmatory program is the Mk II program generic load without averaging. This non-averaged generic load provides additional conservatism as discussed below. A comparison between this load definition and the interim chugging load accepted by NRC in NUREG-0487, Supplement #2 is not as straightforward as that made for CO. The primary reason for this is that generic chugging is a source load definition and wall pressures may vary somewhat from facility to facility or even from location to location within the same facility. Wall pressure time histories for Shoreham are just now being completed and PSDs are not available. Some ARS data is available for both the interim and the Shoreham confirmatory load definitions, but in both cases, the data was generated for internal evaluations only and is incomplete. Comparisons of the limited ARS data available show the two loads to be comparable at high frequency, but the Shoreham confirmatory load bounding at low frequency. In the reactor building (secondary containment) where the majority of the piping and equipment is located, the Shoreham confirmatory load is generally bounding across the frequency range, and where it is exceeded, both loads are bounded by the design basis (DFFR 20 to 30 hz).

Because of the somewhat inconclusive nature of the limited ARS comparisons, Shoreham is submitting the following additional information. Figure 28-1 shows a comparison of the PSD of the Shoreham confirmatory load definition in the JAERI-CRT facility with that of the accepted interim load. Two elevations are presented for the Shoreham confirmatory load: 3600 mm (vent exit plane) and 1800 mm. The Shoreham confirmatory PSD was generated with the same dephasing window as that used for plant application (50 msec), but instead of choosing the worst variance in 1,000 trials for individual chug start times, an "averaging" procedure was used (described in detail in Section 6.2 of the generic load definition report (NEDE-24302-P) to deliberately decrease the predicted wall pressures. The effect of this deliberate decrease in the predicted wall pressure loads is most pronounced at high frequencies.

Because of the decreased degree of conservatism in the method of application of the Shoreham confirmatory chugging load to the JAERI-CRT facility as compared to that used in the Shoreham evaluation, Shoreham considers the PSD comparison of JAERI-CRT shown in Figure 28-1 to be more than adequate for assessing the interim load effect on the Shoreham plant. An inspection of Figure 28-1 shows an exceedence of the Shoreham confirmatory by the interim only at approximately 2 hz. The reason for this exceedence is that the condensation event at t = 25.3 sec in Run 26 was considered in the interim load definition to be a chug and was included in the chugging data base. The Mark II Owners Group considers this event to be part of condensation oscillation because water did not enter the vent. The Mark II OG did include this event in the CO data base, where it is completely bounded by other CO events as shown in Figure 28-1.

A second concern expressed by the NRC staff is that when the final generic chugging load definition with averaging is approved, the Shoreham confirmatory load (without averaging) may not be completely bounding. This is because the averaging procedure brings seven additional chugs into the load definition. Figure 28-2 shows that the PSD envelope of the 7 key chugs bounds the PSD of the 7 adjacent chugs except for slight exceedences at approximately 14 hz and 28 hz. It is evident that a load definition based solely on the key chugs is clearly conservative, even though averaging will result in some minor shifts in frequency content. The above demonstrates the conservatism of the Shoreham confirmatory load definition without averaging.

In summary, in the interest of expediting closure of this open item, Shoreham will commit to a two phase approach. In Phase I, Shoreham will use the confirmatory basis described above (generic chugging without averaging) as an interim load in place of the interim load accepted by NRC-GIB in NUREG-0487, Supplement #2 to demonstrate design basis adequacy. In Phase II, Shoreham will evaluate the generic load definition (once accepted by NRC-GIB) against the load used in the interim evaluation. This is consistent with the position stated in NUREG-0487 that final loads will be used to confirm those used in the interim.

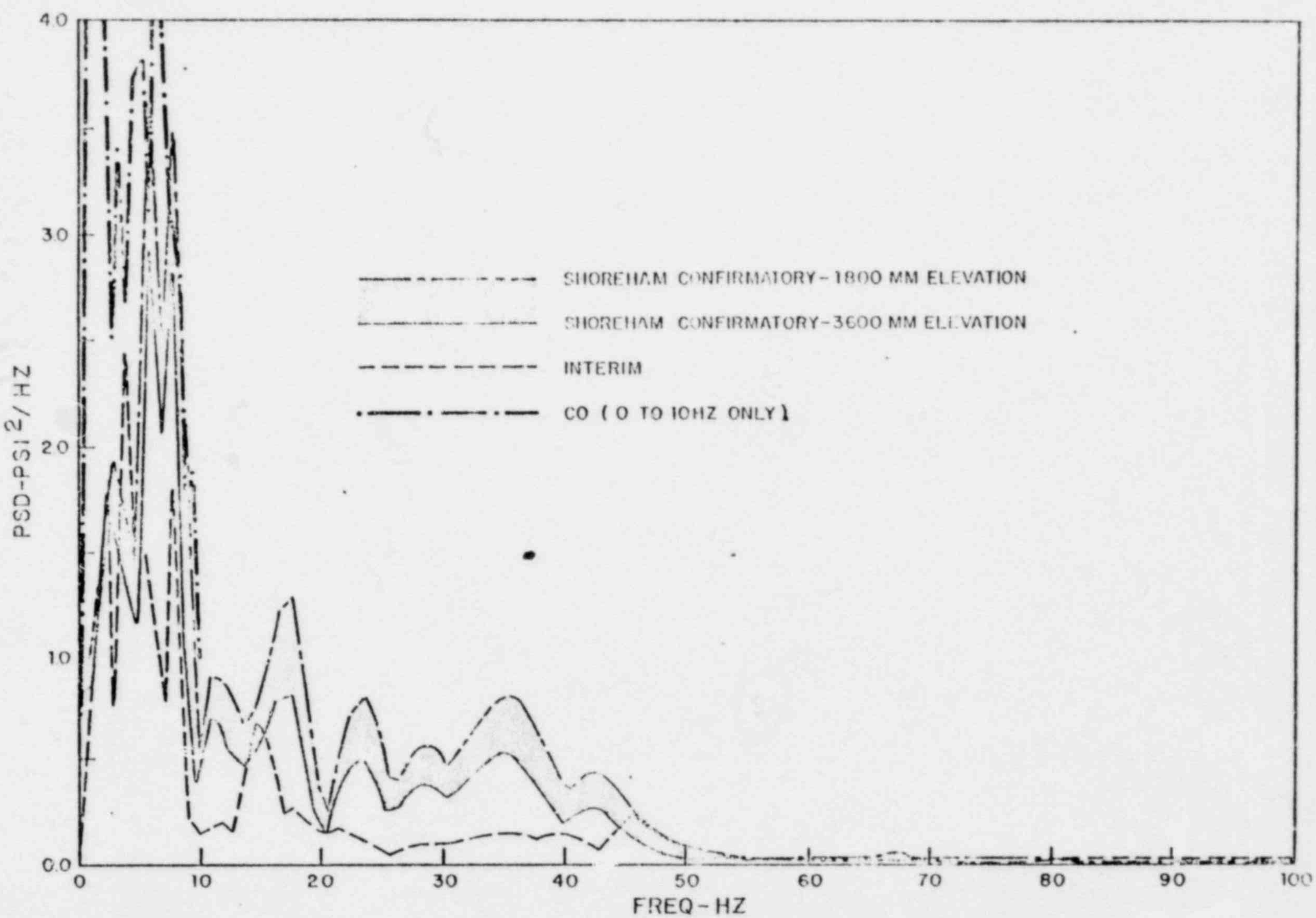


FIGURE 28-1
PSD COMPARISON - INTERIM VS.
SHOREHAM CONFIRMATORY IN JAERI-CRT

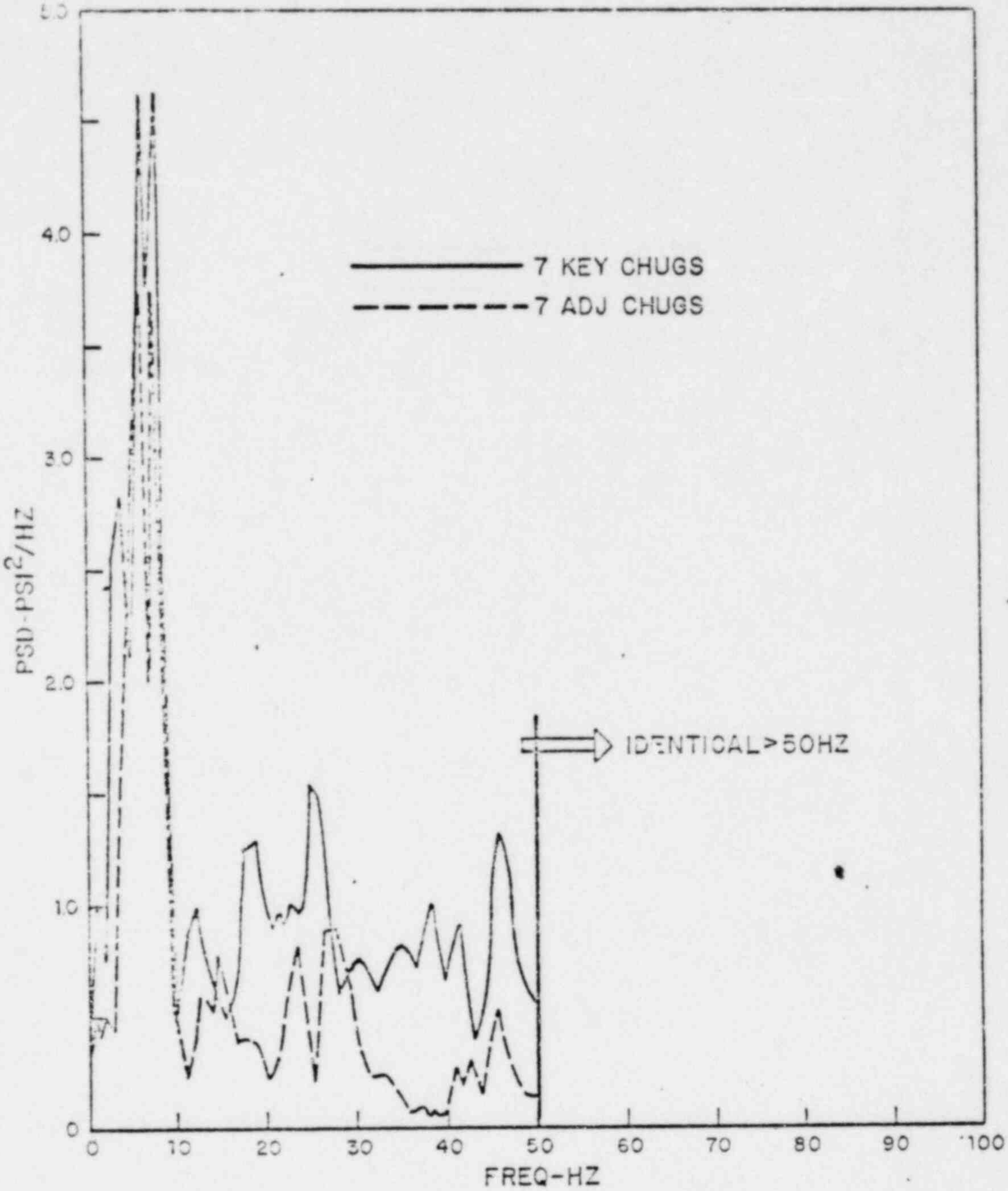


FIGURE 28-2
PSD COMPARISON - 7 KEY CHUGS
VS. 7 CHUGS USED FOR AVERAGING

Item 29 - Quencher Air Clearing Loads

The Shoreham quencher air clearing load is that described in KWU Report R141/141/79/E "Application of the SSES - Test Measurement Results to the Overall Loading of the Suppression Chamber of the Shoreham Plant by Depressurization Processes - Revision 1" dated October 23, 1979 with two exceptions. The first exception is that the ADS trace (Test 11.1) will employ frequency multiplier of 2.0 to 2.86 instead of 2.0 to 2.2. The second exception is that Test 4.1.6 will employ an amplitude multiplier of 1.12 instead of 1.07.

The opening characteristics of the Shoreham safety/relief valves (SRV's) have been verified to be essentially the same as those tested at Karlstein (opening time $\approx$ 20-30 msec). The configuration of the Shoreham SRV discharge line vacuum breakers is the same as that of Karlstein: two 6 inch valves in parallel. Therefore, the data taken at Karlstein is applicable to Shoreham when corrected for pool geometry effects.

Actuation of ADS at Shoreham will not occur at a reactor pressure greater than 78 bar. In order to actuate ADS, low reactor coolant level condition must exist. Low reactor coolant level also causes a reactor SCRAM which immediately reduces reactor power to decay heat levels (e.g., 5 percent power approximately one minute after SCRAM from full power). Therefore, by the time sufficient coolant inventory could be lost from a small break to actuate ADS, decay heat would have decreased to a point where the first set of SRV's would be maintaining pressure at approximately 1115 psig. This is less than the accumulator pressure for ADS Test 11.1. Therefore, the wall pressure loads for Test 11.1 need not be extrapolated for reactor pressure.

It is Shoreham's understanding that, subject to the above exceptions and qualifications, the NRC staff has found this load definition acceptable and that official approval is imminent.

Item 30 - Drywell Pressure History (for pool swell)

Shoreham plant-unique drywell pressure history vs. generic (NEDM-10320 with Moody Slip - flow treatment of subcooled inventory) comparisons were provided to NRC-CSB at the ACRS meeting on April 28, 1981. These comparisons demonstrate that the Shoreham drywell pressure history is bounding and, therefore, acceptable.

Item 31 - Impact Loads on Grating (supports)

The NRC impact load criterion provided in NUREG-0487 covers only flat and cylindrical targets. Shoreham employs wedges on certain platform supports in the pool swell zone to divert flow and reduce impact loads on the supports. The method used by Shoreham to calculate impact loads on wedges is identified in Appendix D of the Shoreham DAR Revision 4 (the response to NRC question 020.72). Shoreham is awaiting NRC review of this material.

Item 32 - Steam Condensation Submerged Drag Loads

NRC-CSB has not yet reviewed the steam condensation submerged structure load methodology described in LILCO letter SNRC-445 dated November 7, 1979 and reiterated in Appendix K to Shoreham DAR Revision 4. Shoreham is awaiting NRC review of this material.

Item 33 Pool Temperature Limit (Review of suppression pool temperatures transients involving SRV discharge)

In Section 6.2.1.8(f) of the Shoreham SER, the NRC staff states that the review of the Shoreham suppression pool temperature transients involving SRV discharge is still in progress. A meeting between representatives of Containment Systems Branch, Generic Issues Branch and the Mark II OG Mass/Energy Subcommittee on March 17, 1981 clarified the generic versus plant-unique aspects of the NRC staff review of the pool temperature transients. Based on the results of that meeting, Shoreham is providing the following additional information to CSB to facilitate review of the plant-unique aspects of the analysis.

1. The transients are analyzed in accordance with "Assumptions for Use in Analyzing Mark II BWR Suppression Pool Temperature Response to Plant Transients involving Safety/Relief Valve Discharge" dated March 24, 1980 with one exception: loss of offsite power is not assumed for SBA and isolation/SCRAM cases as described in IV.A.1. This assumption is conservative as discussed below.

2. Feedwater is added non-mechanistically for all transients. In all cases except SORV at power with main condenser available, steam line isolation is assumed to occur 3.5 seconds after the start of the transient to maximize heat addition to the pool. In reality, MSIV closure would rapidly terminate steam flow to the turbine driven feed pumps and feedwater flow to the vessel. In such a sequence of events, feedwater could again begin to enter the vessel once reactor pressure fell below the shutoff head of the condensate pumps. This assumption is the basis for continuing to add feedwater. However, the condensate pumps require availability of offsite power to continue operation, and therefore, availability of offsite power is the conservative assumption.

3. Availability of offsite power permits continued delivery of CRD return flow to the vessel. CRD flow was <u>not</u> assumed in the analysis of SBA with failure of one RHR Hx for Shoreham and provides additional conservatism for this case.

4. Availability of drywell coolers is not assumed for SBA cases. For SBA cases, pool cooling is suspended for 10 minutes when reactor pressure decreases to the permissive value for LPCI operation. The 10 minutes suspension allows for realignment of the RHR system to pool cooling mode.

5. For Shoreham, HPCI operation is not included in the transient analysis because of the FW assumption discussed above. In reality, HPCI would be expected to operate. There is no pool temperature cutoff for HPCI operation incorporated in the Shoreham design. Adequate HPCI pump NPSH has been verified for operation at the maximum pool temperature observed in the analyses for reactor pressure greater than 150 psia.

6. For Shoreham, no single active failure can result in the permanent loss of one loop of pool cooling and the simultaneous loss of shutdown cooling mode. A power supply

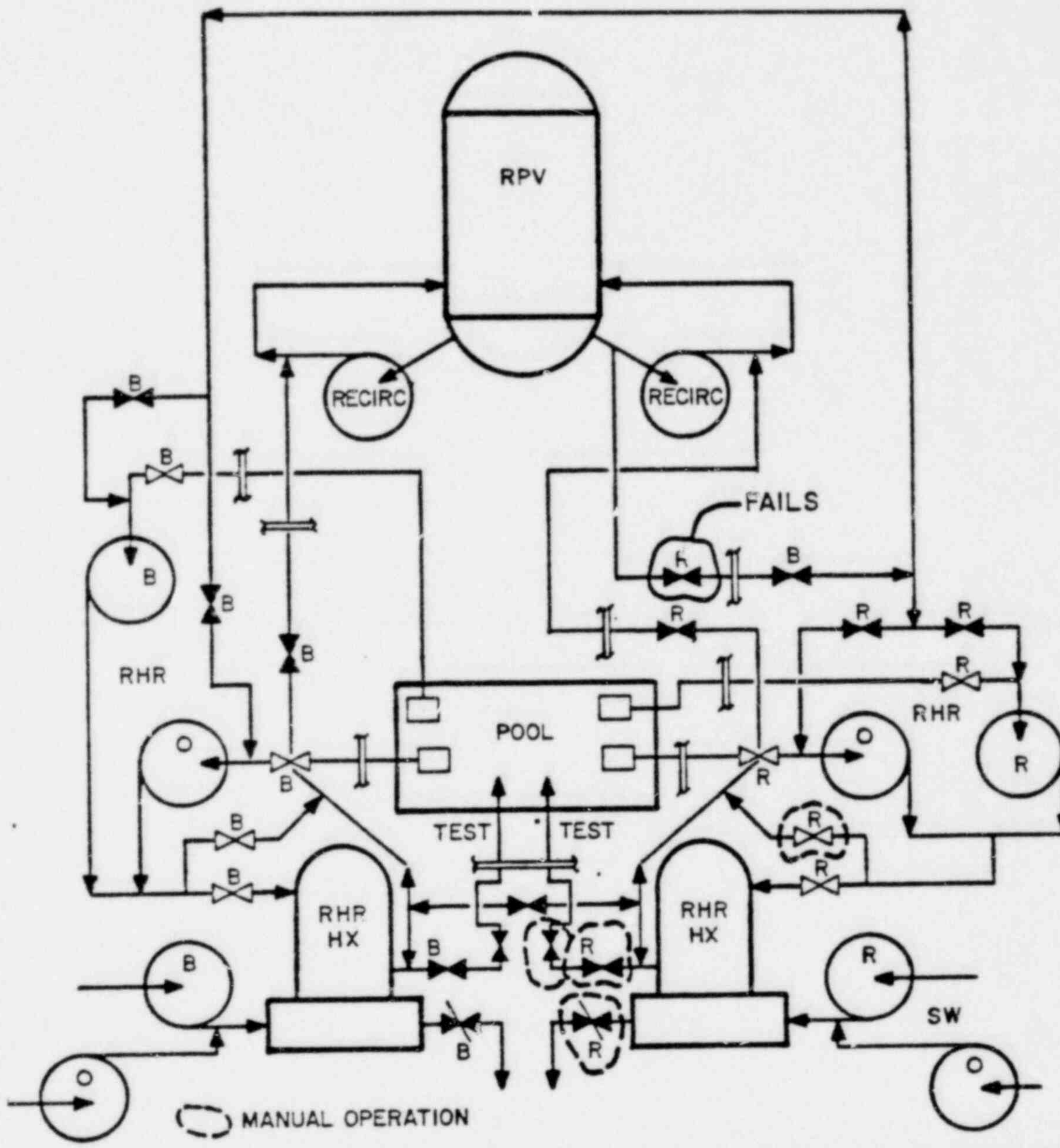
failure disabling the inboard letdown valve for pool cooling and one RHR loop can be overcome by manual realignment of four valves in the "faulted loop" (see Figure 33-1). These valves are accessible for manual operation, and realignment can be accomplished within one hour. A one hour delay in bringing the second loop into operation would have a negligible effect on the final pool temperature. The seal cooler on the "swing" bus pump would not be operable in this configuration, but seal cooling is not required for pump flow temperatures less than 212°F which is satisfactory for pool cooling mode. Passive failures (e.g. of the heat exchanger) are not postulated in the short term, and two heat exchanger operation is not required in the long term.

7. In the Shoreham DAR Revision 4, a pool temperature alarm at TS1 (90°F) requiring operation of the RHR pool cooling mode is described. A second alarm is now being added at TS3 (110°F) requiring SCRAM. Technical specification require the mode switch to be placed in "Shutdown" if TS3 is exceeded. Shoreham has also implemented pressure switches in the SRV tailpipes to provide positive indication of an SRV lift.

8. The Shoreham main condenser is described in detail in FSAR Section 10.4.1. Operating procedures will be reviewed to ensure that the main condenser is identified as the preferred heat sink for any transient not leading to MSIV closure.

9. The Shoreham Main Condenser [illegible]r Removal System, Steam Seal System, Turbine Bypass System [illegible] Circulating Water System are described in detail in FSAR Section 10.4.2 through 10.4.5 respectively.

The above information covers the topics identified in the March 17, 1981 meeting described above and together with DAR Section 10, Appendix I and Appendix J provides the necessary information for a review of the Shoreham suppression pool temperature transients due to SRV discharge.



NOTE: SHOREHAM RHR SYSTEM SHOWING THE EFFECT OF INBOARD RHR VALVE FAILURE. MANUAL OPERATION OF FOUR VALVES REQUIRED TO PLACE "FAULTED LOOP" INTO POOL COOLING MODE - DELAY ESTIMATED TO BE ~1 HOUR.

FIGURE 33-1
SER OPEN ITEM NO. 33 -- POOL TEMPERATURE LIMIT (SECTION 6.2.1.8)

Item 34 - Quencher Arm and Tie-down Loads

This open item deals with the option exercised by Shoreham to use the T-quencher arm and tie-down load specification described in the PP&L Design Assessment report instead of the X-quencher method outlined in the DFFR. In actuality, Shoreham has used Karlstein test data in lieu of the generic PP&L specification where the test data was greater than the specified load. It is Shoreham's understanding that the NRC staff and its consultants have found the generic PP&L load specification acceptable with the exception of the quencher support and arm bending moments where the test data exceeded the load specified. Shoreham has used the measured quencher support and arm bending moments in its design, but to account for possible system variations, Shoreham will commit to increasing the measured quencher support bending moment by a factor of 1.25 for design assessment.

Item # 39 - Emergency Procedures

Emergency procedures for ATWS events will be developed and submitted to the NRC for review. Likewise, operating procedures for the primary to secondary containment leakage detection and return system will also be submitted to the NRC for review after their development.

Item # 47 - Control System Failure

A review has been conducted in which it was determined that only Class 1E systems are necessary to achieve cold shutdown. Failure of the power source to the reactor manual control system and other nonessential systems and components will not affect any essential equipment nor the ability to safely shutdown the plant. It has been further determined that the loss of any one power source serving essential instrument and control systems will not affect the ability to achieve a cold shutdown since diverse equipment and systems are supplied from independent busses. A more detailed discussion of the above is included in our response to SER Open Item No. 46.

While the ability to achieve cold shutdown is not affected by the events described above, a failure of certain control systems may either directly or indirectly impact the characteristics and severity of anticipated transients. These control systems are categorized as follows:

Category A

Directly involved in initial identification or detection of event.

- Reactor water level 8 trip

Category B

Directly and actively involved in event or its effects.

- Relief valve operation
- Bypass system operation
- Rod block monitor

Category C

Indirectly or passively involved in event or its effects.

- Reactor feedwater system
- Reactor turbine pressure regulator
- Recirculation flow controller

Item # 47 (cont'd)

Category D

Not involved directly or indirectly in event or its effects.

- Instrumentation or power electric busses
- Environmental control systems
- Component service water systems

The impact of Category A and B failures is discussed below. Category C & D equipment performance does not significantly impact event severity.

Reactor Water Level Trip (L 8)

If this trip system, an anticipator of level, pressure and heat sink problems fails then the turbine generator moisture/vibration monitor would initiate a turbine trip. In addition, the Reactor Protection System is a fully safety grade backup to the Level 8 trip. General Electric has generically analyzed the impact of Level 8 trip failure upon transient event severity.

The results of these studies have been discussed in meetings with the NRC both generically and on plant specific dockets. It has been concluded that the delta MCPR impact consequence of the L8 trip failure is sufficiently small to justify its continued use in transient analyses.

In order to provide further assurance of L8 trip operability an addition will be made to the Shoreham Technical Specification to provide for formal surveillance.

Relief Valve Operation

Should the relief function of the SRVs fail to operate, the valves would open automatically in the safety mode (fully safety grade). There is no difference in event impact between the relief and safety functions since the MCPR reaches its lowest values before opening of the SRVs.

Main Turbine Bypass System

The main turbine steam bypass system provides a momentary relief function for certain events. The most limiting transient event which takes credit for the turbine bypass system is the feedwater controller failure. Analysis indicates that a delta MCPR increase of approximately 0.08 applies to the transient without a functioning main turbine bypass system. In light of bypass system reliability and the very low probability of feedwater controller failure, this delta MCPR increase is not considered to be large enough to justify a change in the present Chapter 15 transient analyses.

Item # 47 (cont'd)

In order to provide further assurance of turbine bypass operability, an addition will be made to the Shoreham Technical Specifications to provide for formal surveillance.

Rod Block Monitor

The Reactor Manual Control System implements a rod block if an erroneous rod withdrawal is attempted. The rod withdrawal error transient is evaluated utilizing the mitigating effect of the Rod Block Monitor (RBM).

General Electric met with the NRC on January 22, 1981 to demonstrate that the RBM is highly reliable having many redundant and self-testing features and that credit for its operation should be allowed in transient analyses.

The NRC indicated tentative approval of the design and transient analysis with the addition of periodic Technical Specification testing to assure system operability.

The Shoreham Technical Specifications will be amended to include this requirement.

Item # 48 - High Energy Line Breaks

The Environmental Qualification Program is currently being implemented at Shoreham. It describes the program by which Class 1E equipment will be qualified in accordance with NUREG-0588 to certain defined environmental limits. Those limits are established on the basis of analysis of worst case events, including high energy line breaks. However, nonsafety related equipment does not require environmental qualification. The failure of these components in an accident environment will not affect the ability to safely shut down the plant and do not result in consequences more severe than those of Chapter 15 analyses or beyond the capability of operators or safety systems. A more detailed discussion of the significance of control system failures is provided in our response to SER Open Items numbers 46 and 47.

SECTION A

1. Biological shielding within Primary Containment, Reactor Building, Control Building.

Response: Table 3.2.1-1, Section XLII, Structures, has been modified to incorporate item 1.

2. Missile Barriers within Primary Containment, Reactor Building and Control Building.

Response: Table 3.2.1-1, Section XLIII, Structures, has been modified to incorporate the missile barriers within the Reactor Building and Control Building. There are no missile barriers located within the Primary Containment.

3. Combustible Gas Control System

Response: Item 3 is part of the Primary Containment Atmosphere Control System. Item XXVIII of Table 3.2.1-1.

4. Engineered Safety Feature Actuation System

Response: The engineered safety systems and their actuation signals are previously called out in Table 3.2.1-1 as part of the system(s) in which they are located.

5. Sampling System

   5a Containment isolation valves
   5b Piping within containment isolation valves

Response: For the Post-Accident Sampling System, the corresponding isolation valves (5a above) and the piping within the valves (5b above) are designated as part of the system(s) being sampled, thus they are previously discussed in Table 3.2.1-1.

6. Containment Spary

Response: Containment spray is part of the RHR system which is addressed in Section IX of Table 3.2.1-1.

7. Onsite Power Systems (Class 1E)

   7a Diesel generator package including auxiliaries...

Response: The diesel generator package, including auxiliaries, are addressed in Section XXVII, Onsite Power Systems, Subsection (a) Diesel Emergency Power System, item 6, diesel generators.

7b 4160V switchgear
7c 480V load centers
7d 480V motor control centers
7e Instrument, control and power cables...
7g Transformers
7i Protective relays and control panels
7j AC control power inverters
7k 120V AC vitial bus distribution equipment
7l Containment electrical penetration assemblies
7m Other cable penetrations (fire stops)

Response: 7,b,c,d,e,g,i,j,k,l,m were previously covered in Table 3.2.1-1. Further clarification of items 7,b,c,d,e,g,i,j, k,l,m is provided in Table 3.2.1-1, in revised Section XXVII, Onsite Power Systems.

7f Conduit and cable trays and their supports...

Response: Item 7f are off the shelf hardware items, thus they will not be included in Table 3.2.1-1.

7h Valve operators have system designation, and are classified as part of, and along with, the system in which they fall.

8. DC Power Systems (Class 1E)

8a 125V batteries, battery charges, and distribution equipment
8b Cables
8d Battery racks
8e Protective relays and control panels

Response: Items 8,a,b,d,e are covered in the revised Section XXVII, Onsite Power Systems, Subsection (e) DC Power Systems.

8c Conduit and cable trays and their supports...

Response: Item 8c are commercial grade hardware items, thus they will not be included in Table 3.2.1-1.

9. Main Steam Isolation Valves Leakage Control Systems

Response: Item 9 above has been incorporated in Table 3.2.1-1 see Section XXXIV.

10. Radiation Monitoring (fixed and portable)

11. Radioactivity Monitoring (fixed and portable)

12. Radioactivity Sampling (air surface and liquid)

Response: Items 10,11,12 were previously covered in Section VIII, Process Radions Monitors, and Section XXXIX, Area Radiation Monitoring System of Table 3.2.1-1.

13. Radioactive Contamination Measurement and Analysis

14. Personnel Monitoring Internal and External

15. Instrument Storage, Calibration and Maintenance

16. Decontamination

17. Respiratory Protection, Including Testing

18. Contamination Control

Response: Items 13-18 are administrative requirements and will not be included in Table 3.2.1-1.

19. Radiation Shielding

Response: Item 19 is similar to item 1 above and is incorporated in Table 3.2.1-1 Section XLII Structures.

20. Waterproof Doors to Safety-Related Buildings

Response: Section XLII of Table 3.2.1-1, Structures, has been modified to include the waterproof doors of the Control Building, Screenwell and the Diesel Fuel Pump House.

21. Site Grading

Response: Section XLII of Table 3.2.1-1, Structures, has been modified to include that area adjacent to the intake canal called out in the Safety Evaluation Report pg. 2-24.

22. Sediment Measurements in the Intake Canal

Response: Item 22 will not be included in Table 3.2.1.-1 since it is an administrative requirement. However, this item is a Technical Specification Requirement.

23. Meteorological Data Colllection Program

Response: Item 23 will not be included in Table 3.2.1-1 since this is an administrative requirement.

24. Expendable and Consumable Items Necessary for the Functional Performance of Safety-Related Structures, Systems and Components.

Response: Item 24 will not be included in Table 3.2.1-1 since this is an administrative requirement.

25. Safety-Related Masonry Walls

Response: Section XLII of Table 3.2.1-1, Structures, has been modified to incorporate this item.

26. Measuring and Test Equipment Used for Safety-Related Structures, Systems and Components.

Response: Item 26 is an administrative requirement and will not be incorporated in Table 3.2.1-1.

SECTION B

1. Primary containment atmospheric control system - The hydrogen recombiners and associated containment isolation valves and piping within containment isolation valves should be under the controls of the operational QA program.

Response: This system has been QA qualified in Table 3.2.1-1, Section XXVIII, and all administrative QA responsibilities have been initiated.

2. Reactor building closed loop cooling water system - The piping within the containment isolation valves should be under the controls of the operational QA program.

Response: This system has been QA qualified in Table 3.2.1-1, Section XXVIII, and all administrative QA responsibilities have been initiated.

3. Identify the safety-related instrumentation and control systems and components to the same scope and level of detail provided in Chapter 7 of the FSAR.

Response: The instrumentation and control system components have been classified in Table 3.2.1-1 as part of the systems in which they fall.

4. Clarify that charcoal filters are included in the building standby ventilation system and the control room ventilation system.

Response: Section XXIX, Reactor Building Standby Ventilation Systems and Section XXXVIII, Miscellaneous Ventilating Systems of Table 3.2.1-1 have been modified to include item 4.

5. Clarify that the floodproofing of the seismic category I civil structures listed in item XLII of Table 3.2.1-1 meets the QA requirements of 10 CFR 50, Appendix B.

Response: Section XLII, Structures, has been modified to incorporate all items in 5 above.

6. Provide a Section in Table 3.2.1-1 for Effluent Radiation Monitors.

Response: Section VIII, Process Radiation Monitors includes all airborne and effluent monitors.

7. Radwaste System tank, atmospheric, must meet Regulatory Guide 1.143.

Response: Section XVIII Radwaste System of Table 3.2.1-1 has been modified to incorporate item 7.

8. Ducting and Isolation Valves should be classified under ASME III-2.

Response: ASME III-2 does not cover ventilation ducting, thus, there is no impact on Table 3.2.1-1.

9. Provide a Section in Table 3.2.1-1 for ESF Filtration Systems.

Response. Table 3.2.1-1 already classifies ESF filtration systems in the Reactor Building Standby Ventilation Systems, Section XXIX.

SECTION C

1. Plant Safety-Parameter Display Console

Response: Table 3.2.1-1 modified to incorporate this item, see Section XLIV.

2. Reactor Coolant System Vents

Response: Not applicable to Shoreham. No plant related change required.

3. Plant Shielding

Response: Not applicable to Shoreham. No plant related changed required.

4. Post Accident Sampling

Response: Table 3.2.1-1 modified to incorporate post accident sampling system - Section XLV.

5. Valve Position Indication

Response: Table 3.2.1-1 modified to incorporate valve position indication, Section XLVI.

6. Dedicated Hydrogen Penetration

Response: Item 6 is previously covered in Section XLIII, Primary Containment Structure, item 3, Penetrations, in Table 3.2.1-1.

7. Containment Isolation Dependability

Response: Containment isolation dependability is a system by system responsibility and the related systems are already classified and included in Table 3.2.1-1.

8. Accident Monitoring Instrumentation

Response: Incorporated into Table 3.2.1-1, Section XLVII, Accident Monitoring Instrumentation System.

9. Instrumentation for Detection for Inadequate Core-Cooling

Response: Implementation of changes will be accomplished on a system by system basis, hence any modifications will be in Table 3.2.1-1 on a system by system basis.

10. HPCI and RCIC Initiation Levels

Response: All changes to the HPCI and RCIC have the same classification as the systems in which they are located.

11. Isolation of HPCI and RCIC

Response: All changes to the HPCI and RCIC have the same classification as the systems in which they are located.

12. Challenges to and Failure of Relief Valves

Response: There is no hardware changes related to item 12, thus no modification to Table 3.2.1-1 is required.

13. ADS Actuation

Response: There is no hardware change related to this item, thus, no modification to Table 3.2.1-1 is required.

14. Restart of Core Spray and LPCI

Response: There is no hardware change related to this item, thus, no modification to Table 3.2.1-1 is required.

15. RCIC Suction

Response: There is no hardware change related to this item, thus, no modification to Table 3.2.1-1 is required.

16. Space Cooling for HPCI and RCIC

Response: There is no hardware change related to this item, thus, no modification to Table 3.2.1-1 is required.

17. Power on Pump Seals

Response: There is no hardware change related to this item, thus, mo modification to Table 3.2.1-1 is required.

18. Common Reference Levels

Response: There is no hardware change related to this item, thus, no modification to Table 3.2.1-1 is required.

19. ADS VAlves, Accumulators and Associated Equipment and Instrumentation.

Response: There is no hardware change related to this item, thus, no modification to Table 3.2.1-1 is required.

20. Emergency Plans

Response: Item 20 is an administrative requirement and has no impact on Table 3.2.1-1.

21. Emergency Support Facilities

Response: Item 21 incorporated in Table 3.2.1-1, Section XXXIII, Permanent Emergency Support Facilities.

22. Inplant I_2 Radiation Monitoring

Response: Components of the Inplant I_2 Radiation Monitoring System that are used in conjuntion with the Process Radiation Monitoring System are classified as part of the Process Radiation Monitoring System, Section VIII, of Table 3.2.1-1.

23. Control Room Habitability

Response: Systems responsible for monitoring control room habitability are identified in Table 3.2.1-1.

TABLE 3.2.1-1

EQUIPMENT CLASSIFICATION

	Principal Component(1)	Scope(2) of Supply	Location(3)	Quality(4a) Group Classification	LILCO(4b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
I.	Reactor System								
	1. Reactor vessel	GE	PC	A	I	I	2/67	ASME III A, 1965 Winter 66	
	2. Reactor vessel support skirt	GE	PC	-	I	I	2/67	ASME III A, 1965 Winter 66	
	3. Reactor vessel appurtenances, pressure retaining portions	GE	PC	A	I	I	12/69	ASME III A, 1965 Winter 66	
	4. CRD housing supports	GE	PC	-	I	I		X	
	5. Reactor internal structures, engineered safety features	GE	PC	A	I	I		X	
	6. Core support structures	GE	PC	-	I	I	2/67	X	
	7. Other internal structures								
	a. Shroud head & separator assembly	GE	PC	-	I	I		X	
	b. Dryers	GE	PC	-	II	I		X	(9)
	8. Control rods	GE	PC	-	I	I		X	
	9. Control rod drives	GE	PC	-	I	I		X	
	10. Power range detector hardware	GE	PC	B	I	I		ASME III-2	
	11. Fuel assemblies	GE	PC	-	I	I		X	
	12. Reactor vessel stabilizer	GE	PC	-	I	I		X	
	13. Reactor vessel star truss	P	PC	-	I	I		X	
	14. Reactor vessel insulation	GE	PC	-	II	NA		X	
II.	Nuclear Boiler System								
	1. Vessels, instrumentation condensing chambers	GE	PC	A	I	I		ASME III-1	
	2. Vessels, air accumulators	P	PC	B	I	I		ASME III-2	

TABLE 3.2.1-1 (CONT'D)

Principal Component(1)	Scope of Supply(2)	Location(3)	Quality(4a) Group Classification	LILCO(4b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
3. Piping, relief valve discharge (including ramshead and supports)	P	PC	C	I	I		ASME III-3	
4. Piping, main steam within outer isolation valve	GE	PC	A	I	I	11/69	B31.1.0	
5. Pipe supports, main steam within outer isolation valve	GE	PC	A	I	I	1/75	ASME III	
6. Pipe whip restraints, main steam	P	PC	-	I	I		X	
7. Piping feedwater, within outermost isolation valves	P	PC, RB	A	I	I		ASME III-1	
8. Other primary coolant pressure boundary piping within isolation valves	P	PC	A	I	I		ASME III-1	
9. Piping, instrumentation beyond outermost isolation valves	P	RB	D	II	NA		See Note 8	
10. Safety/Relief Valves	GE	PC	A	I	I	12/69	ASME I, III, & I. 1968 Winter	
11. Valves, main steam isolation valves	GE	PC, RB	A	I	I	10/69	B31.1.0/ASME VIII	
12. Valves, feedwater isolation valves and within	P	PC, RB	A	I	I		ASME III-1	(10)
13. Valves, other, isolation valves and within	P	PC, RB	A	I	I		ASME III-1	
14. Valves, instrumentation beyond outermost isolation valves	P	RB	D	II	NA		See Note 8	
15. Electrical modules with safety function	GE	PC	-	I	I		X	
16. Cable, with safety function	P	-	-	I	NA		X	

III. Recirculation System

Principal Component(1)	Scope of Supply(2)	Location(3)	Quality(4a) Group Classification	LILCO(4b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
1. Piping	GE	PC	A	I	I	10/69	B31.1.0	
2. Piping suspension, recirculation line	GE	PC	-	I	I	12/74	ASME III	

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
	3. Pipe restraints, recirculation line	GE	PC	–	I	I			
	4. Pumps	GE	PC	A	I	I	11/69	ASME IIIC	
	5. Valves	GE	PC	A	I	I	10/69	ASME VIII/MSS Sp66	
	6. Pump motors	GE	PC	–	II	I	10/69	X	
	7. Electrical modules, with safety function	GE	RB,R	–	I	I		X	
	8. Cable with safety function	P	–	–	I	NA		X	
IV.	CRD Hydraulic System								
	1. Valves, isolation, water return line	P	PC, RB	A	I	I		ASME III-1	
	2. Valves, scram discharge volume lines	GE	RB	B	I	I	12/69	B31.1.0	
	3. Valves insert and withdraw lines	P	RB	B	I	I		ASME III-2	(11)
	4. Valves, other	P	RB	D	II	NA		B31.1.0	
	5. Piping, water return line within isolation valves	P	PC	A	I	I		ASME III-1	
	6. Piping, scram discharge volume lines	P	RB	B	I	I		ASME III-2	
	7. Piping, insert and withdraw lines	P	PC, RB	B	I	I		ASME III-2	
	8. Piping other	P	RB	D	II	NA		B31.1.0	
	9. CRD pumps, filters, and strainers	GE	RB	D	II	NA	1970	X	
	10. Hydraulic control unit	GE	RB	–	I	I		Special	(12)
	11. Cable, with safety function	P	–	–	I	NA		X	
V.	Standby Liquid Control System								
	1. Standby liquid control tank	GE	RB	B	I	I	2/74	ASME II, III, IX & API 620/650	
	2. Pump	GE	RB	B	I	I	12/69	B31.1.0 & HIS	
	3. Pump motor	GE	RB	–	I	I	12/69	X	
	4. Valves, explosive	GE	RB	B	I	I	12/67	ASME VIII-1	

TABLE 3.2.1-1 (CONT'D)

	Principal Component(1)	Scope(2) of Supply	Location(3)	Quality(4a) Group Classification	LILCO(4b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
	5. Valves, isolation and within	P	RB	A	I	I		ASME III-1	
	6. Valves, beyond isolation valves	P	RB	B	I	I		ASME III-2	
	7. Piping, within isolation valves	P	PC, RB	A	I	I		ASME III-1	
	8. Piping, beyond isolation valves	P	RB	B	I	I		ASME III-2	
	9. Electrical modules, with safety function	GE	RB, R	-	I	I		X	
	10. Cable, with safety function	P	-	-	I	NA		X	
	11. Accumulators	GE	RB	C	I	I	8/71	See Note 13 (13)	
	12. Valves and piping other	P	[illegible]	D	NA	NA		B31.1.0	
	13. Drain tank	P	[illegible]	D	NA	NA		ASME VIII	
VI.	Neutron Monitoring System								
	1. Piping, TIP	GE	PC			I		B31.1	
	2. Valves, isolation, TIP subsystem	GE	RB	B D	I II	I		[illegible] ASME III-2	
	3. Electrical modules, IRM and APRM	GE	RB, R	-	I	I		X	
	4. Cable, IRM and APRM	P		-	I	NA		X	
VII.	Reactor Protection System								
	1. Electrical modules	GE	PC, RB, R, T	-	I	I		X	
	2. Cable	P		-	I	NA		X	
VIII.	Process Radiation Monitors								
	1. Electrical modules, main steam line and reactor building ventilation monitors	GE	RB, R, T	-	I	I		X	
	2. Cable, main steam line and reactor building ventilation monitors	P		-	I	NA		X	

POOR ORIGINAL

TABLE 3.2.1-1 (CONT'D)

	Principal Component(1)	Scope(2) of Supply	Location(3)	Quality(4a) Group Classification	LILCO(4b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
	3. Electrical modules for process liquid, process ventilation, air ejector offgas, and offgas treatment radiation monitoring systems	GE	R,T,RW,RB	-	II	NA		X	
		P	R,T,RWW,RB	-	II	NA		X	
	4. Electrical modules for effluent radiation monitoring system	P	R,T,RW,RB	-	I & II	I & NA	9/76		
IX.	RHR System								
	1. Heat exchangers, primary side	GE	RB	B	I	I	8/69	ASME III-C & TEMA C	
	2. Heat exchangers, secondary side	GE	RB	C	I	I	8/69	ASME VIII & TEMA C	
	3. Piping, connected to RCPB within outermost isolation valves	P	PC	A	I	I		ASME III-1	
	4. Piping, beyond outermost isolation valves	P	RB	B	I	I		ASME III-2	
	5. Pumps	GE	RB	B	I	I	8/69	B31.1.0/ ASME III-C	
	6. Pump motors	GE	RB	-	I	I	10/69	X	
	7. Valves, isolation, LPCI and shutdown lines	P	PC, RB	A	I	I		ASME III-1	
	8. Valves, isolation, other	P	PC, RB	B	I	I		ASME III-2	
	9. Valves, beyond isolation valves	P	RB	B	I	I		ASME III-2	
	10. Electrical modules, with safety function	GE	RB,R	-	I	I		X	
	11. Cable, with safety function	P	-	-	I	NA		X	
X.	Core Spray System								
	1. Piping, within outermost isolation valves	P	PC	A	I	I		ASME III-1	
	2. Piping, beyond outermost isolation valves	P	RB	B	I	I		ASME III-2	

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TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
	3. Pumps	GE	RB	B	I	I	9/69	B31.1.0/ ASME III-C	
	4. Pump motors	GE	RB	-	I	I	7/69	X	
	5. Valves, isolation and within	P	PC, RB	A	I	I		ASME III-1	
	6. Valves, beyond outermost isolation valves	P	RB	B	I	I		ASME III-2	
	7. Electrical modules with safety function	GE	RB, R	-	I	I		X	
	8. Cable, with safety function	P	-	-	I	NA		X	
XI.	HPCI System								
	1. Piping, within outermost isolation valves	P	PC	A	I	I		ASME III-1	
	2. Piping beyond outermost isolation valves	P	RB	B	I	I		ASME III-2	
	3. Piping return test line to condensate storage tank beyond reactor building	P	O	D	II	NA		B31.1.0	
	4. Vacuum pump discharge line	P	RB	B	I	I		ASME III-2	
	5. Pump	GE	RB	B	I	I	6/69	B31.1.0/ ASME III-C	
	6. Valves, isolation and within	P	PC, RB	A	I	I		ASME III-1	
	7. Valves, return test line to condensate storage	P	RB	B	I	I		ASME III-2	
	8. Valves, other	P	RB	B	I	I		ASME III-2	
	9. Turbine	GE	RB	-	I	I	6/69		(14)
	10. Electrical modules with safety function	GE	RB, R	-	I	I		X	
	11. Cable, with safety function	P	-	-	I	NA		X	

TABLE 3.2.1-1 (CONT'D)

	Principal Component(1)	Scope(2) of Supply	Location(3)	Quality(4a) Group Classification	LILCO(4b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
XII.	RCIC System								
	1. Piping, within outermost isolation valves	P	PC	A	I	I		ASME III-1	
	2. Piping, beyond outermost isolation valves	P	RB	B	I	I		ASME III-2	
	3. Piping, return test line to condensate storage tank beyond reactor building	P	O	D	II	NA		B31.1.0	
	4. Vacuum pump discharge line from vacuum pump to containment isolation valves	P	RB	B	I	I		ASME III-2	
	5. Pump	GE	RB	B	I	I	6/69	B31.1.0/ ASME III-C	
	6. Valves, isolation and within	P	PC, RB	A	I	I		ASME III-1	
	7. Valve, return test line to condensate storage	P	RB	B	I	I		ASME III-2	
	8. Valves, other	P	RB	B	I	I		ASME III-2	
	9. Turbine	GE	RB, R	-	I	I	6/69	X	(14)
	10. Electrical modules, with safety function	GE	RB, R	-	I	I		X	
	11. Cable, with safety function	P	-	-	I	NA		X	
XIII.	Fuel Service Equipment								
	1. Fuel preparation machine	GE	RB	-	I	I		X	
	2. General purpose grapple	GE	RB	-	I	I		X	

TABLE 3.2.1-1 (CONT'D)

	Principal Component(1)	Scope of Supply(2)	Location(3)	Quality Group Classification(4a)	LILCO(4b) Quality Assurance Category	Seismic Category(5)	Purchase Order Date(6)	Principal Code(7)	Comments(8)
XIV.	Reactor Vessel Service Equipment								
	1. Steam line plugs	GE	RB	-	I	I		X	
	2. Dryer and separator sling and head strongback	GE	RB	-	I	I		X	
	3. Drywell head lifting rig	P	RB	-	I	I		X	
XV.	In-Vessel Service Equipment								
	1. Control rod grapple	GE	RB	-	I	I		X	
XVI.	Refueling Equipment								
	1. Refueling platform	GE	RB	-	I	I	4/71	AISC	
	2. Refueling bellows, drywell	GE	PC	-	II	NA		X	
	3. Refueling bellows, reactor cavity	P	RB	-	II	See Note (15)		X	(15)
	4. New fuel inspection stand	GE	RB	-	II	NA		X	
XVII.	Storage Equipment								
	1. Fuel storage racks	GE	RB	-	I	I		X	
	2. Defective fuel storage container	GE	RB	-	I	I		X	
	3. Spent fuel pool, dryer/sep. pool, Rx cavity liners	P	RB	-	I	I		X	
XVIII.	Radwaste System								
	1. Tanks, atmospheric	P	RW	D	II	NA		X	Reg Guide 1.143
	2. Heat exchangers	P	RW	D	II	NA		ASME VIII	
	3. Piping, containment isolation	P	PC	B	I	I		ASME III-2	
	4. Valves, containment isolation	P	PC, RB	B	I	I		ASME III-2	
	5. Piping, other	P	PB,O,T,RW	D	II	NA		B31.1.0	
	6. Pumps	P	RB, RW	D	II	NA		X	
	7. Valves, flow control and filter system	P	RW	D	II	NA		B31.1.0	
	8. Valves, other	P	RB, RW	D	II	NA		B31.1.0	(26)

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TABLE 3.2.1-1 (CONT'D)

	Principal Component(1)	Scope(2) of Supply	Location(3)	Quality(4a) Group Classification	LILCO(4b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
XIX.	Reactor Water Cleanup System								
	1. Vessels: filter/ demineralizer	GE	RB	C	II	NA	3/70	ASME III-C	
	2. Heat exchangers	GE	RB	C	I	I	12/69	ASME III-C/ TEMA R	(16)
	3. Piping within outermost isolation valves	P	RB	A	I	I		ASME III-1	
	4. Piping, beyond outermost isolation valves	P	RB	C	I	I		ASME III-3	
	5. Pumps	GE	RB	C	II	NA	12/69	ASME III-C/ B31.1.0	(16)
	6. Valves, isolation valves and within	P	RB	A	I	I		ASME III-1	
	7. Valves, beyond outermost isolation valves	a. GE	RB	C	II	I	12/69	B31.1.0	
		b. P	RB	C	I	I		ASME III-3	
XX.	Fuel Pool Cleanup Subsystem								
	1. Demineralizer vessel	P	RW	D	II	NA		ASME VIII	
	2. Filters	P	RW	D	II	NA		ASME VIII	
	3. Pumps, purification	P	RB	D	II	NA		X	
	4. Piping	P	RB, RW	D	II	NA		B31.1.0	
	5. Valves	P	RB, RW	D	II	NA		B31.1.0	
XXI.	Fuel Pool Cooling Subsystem								
	1. Pumps, cooling	P	RB	C	I	I		ASME III-3	
	2. Heat exchangers	P	RB	C	I	I		ASME III-3	
	3. Piping	P	RB	C	I	I		ASME III-3	
	4. Valves	P	RB	C	I	I		ASME III-3	
XXII.	Control Room Panels								
	1. Electrical modules, with safety function	a. P	R	-	I	I		X	
		b. GE	R	-	I	I		X	
	2. Cable, with safety function	a. P	-	-	I	NA		X	
		b. GE	-	-	I	NA		X	

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
XXIII.	Local Panels								
	1. Electrical modules, with safety function	a. P	RB	–	I	I		X	
		b. GE	RB	–	I	I		X	
	2. Cable, with safety function	P	–	–	I	NA		X	
	3. Remote shutdown panel	GE	RB	–	I	I		X	
XXIV.	Offgas System								
	1. Atmospheric glycol tanks	P	T	D	II	NA		X	
	2. Heat exchangers	P	T	D	II	NA		ASME VIII	
	3. Piping	P	T, RW	D	II	NA		B31.1.0	
	4. Valves, flow control	P	T, RW	D	II	NA		B31.1.0	
	5. Valves, other	P	T, RW	D	II	NA		B31.1.0	
	6. Steam jet air ejectors	P	T	D	II	NA		ASME VIII	
	7. Charcoal vessels	P	RW	D	II	NA		ASME VIII	
	8. Recombiners	P	T	D	II	NA		ASME VIII	
	9. Filters	P	RW	D	II	NA		ASME VIII	
XXV.	Service Water System								
	1. Piping, Safety related	P	RB,O,P,R	C	I	I		ASME III-3	
	2. Piping, other	P	–	D	II	NA		B31.1.0	
	3. Pumps	P	P	C	I	I		ASME III-3	
	4. Pump motors	P	P	–	I	I		X	
	5. Valves, isolation	P	P, R	C	I	I		ASME III-3	
	6. Valves, other	P	T,O,P	D	II	NA		B31.1.0	
	7. Electrical modules, with safety function	P	R, P	–	I	I		X	
	8. Cable, with safety function	P	–	–	I	NA		X	
XXVI.	Compressed Air System								
	1. Vessels, accumulators, supporting safety-related systems	P	PC, RB	C	I	I		ASME III-3	

TABLE 3.2.1-1 (CONT'D)

Principal Component(1)	Scope(2) of Supply	Location(3)	Quality(4a) Group Classification	LILCO(4b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
2. Piping in lines between accumulators and safety-related systems	P	PC, RB	C	I	I		ASME III-3	
3. Valves in lines between accumulators and safety-related systems	P	PC, RB	C	I	I		ASME III-3	
4. Piping, containment isolation	P	PC, RB	B	I	I		ASME III-2	
5. Valves, containment isolation	P	PC, RB	B	I	I		ASME III-2	
6. Electrical modules with safety function	P	PC, RB, R	-	I	I		X	
7. Cables with safety function	P	-	-	I	NA		X	
8. Valves and piping, other	P	-	D	II	NA		B31.1.0	
XXVII Onsite Power Systems								
a. Diesel Emergency Power Systems								
1. Day tanks	P	R, O	-	I	I		ASME III-3	
2. Piping, fuel oil system	P	R, O	-	I	I		ASME III-3	
3. Valves, fuel oil system	P	R, O	-	I	I		ASME III-3	
4. Pumps, fuel oil system	P	R, O	-	I	I		ASME III-3	
5. Pump motors, fuel oil system	P	R, O	-	I	I		X	
6. Diesel-generators	P	R, O	-	I	I		X	
7. Electrical modules with safety functions	P	R, O	-	I	I		X	
8. Cable, with safety functions	P	R, O	-	I	NA		X	
9. Diesel fuel storage tanks	P	-	-	I	I		ASME III-3	
10. Diesel air compressors	P	R	-	I	I		X	

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TABLE 3.2.1-1 (CONT'D)

Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
b. A/C Power Systems								
1. 4160 V Switchgear	P	R	—	I	I	3/73	X	
2. 480 V Switchgear	P	R	—	I	I	1/74	X	
3. 480 V MCC	P	R, RB, P	—	I	I	4/74	X	
4. Cables (instrumentation, control & power)	P	—		I	NA	—	X	
5. Transformers	P	R, RB, P, RW	—	I	I	1/76	X	
c. Containment electrical penetration & assemblies	P	RB	—	I	I	1/75	ASME III R.G.1.63	
d. Fire Stops	P	—		I	NA	2/80	X	
e. D.C. Power Systems								
1. 125 V batteries & battery chargers	P	R	—	I	I	11/74 3/75	X X	
2. Cables	P	—	—	I	NA	—	X	
3. Battery racks	P	R	—	I	I	11/74	X	
4. Protective relays and control panels	P	R	—	I	I	4/75	X	

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TABLE 3.2.1-1 (CONT'D)

	Principal Component(1)	Scope(2) of Supply	Location(3)	Quality(4a) Group Classification	LILCO(4b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
XXVIII.	Primary Containment Atmospheric Control System								
	1. Piping	P	RB	-	I	I		ASME III-2	
	2. Valves	P	RB	-	I	I		ASME III-2	
	3. Fans	P	RB	-	I	I		X	
	4. Hydrogen recombiners	P	RB	-	I	I		X	
	5. Electrical modules with safety functions	P	RB	-	I	I		X	
	6. Cables with safety function	P	-	-	I	NA		X	
XXIX.	Reactor Building Standby Ventilation System								
	1. Ducting and isolation valves with safety function, including filter trains	P	RB	-	I	I		X	
	2. Blowers	P	RB	-	I	I		X	
	3. Unit coolers	P	RB	-	I	I		X	(17)
	4. Chilled water system	P	RB, R	-	I	I		X	(18)
	5. Electrical modules with a safety function	P	RB	-	I	I		X	
	6. Cable with a safety function	P	-	-	I	NA		X	
XXX.	Primary Containment Purge System								
	1. Containment isolation valves and associated piping	P	PC, RB	B	I	I		ASME III-2	
	2. All other components	P	RB	-	II	NA		X	
XXXI.	Power Conversion System								
	1. Main steam piping between outermost isolation valves up to turbine stop valves	P	RO, T	B	I	I		ASME III-2	

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TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
	2. Main steam branch piping to 1st valve capable of timely actuation	P	T	B	I	I		ASME III-2	
	3. Main turbine bypass piping up to bypass valve	P	T	B	I	I		ASME III-2	
	4. First valve that is either normally closed or capable of automatic closure in branch piping connected to main steam and turbine bypass piping	P	T	B	I	I		ASME III-2	
	5. Turbine stop valves, turbine control valves and turbine bypass valves	P	T	D	II	NA		Special	(19)
	6. Main steam leads from turbine control valve to turbine casing	P	T	D	II	NA		Special	(19)
	7. Feedwater and condensate system beyond 3rd isolation valve	P	RB, T	D	II	NA		B31.1.0	(10)
XXXII.	Condensate Storage and Transfer System								
	1. Condensate storage tank	P	O	D	II	NA		API-650	(20)
	2. Piping, suction line to HPCI, RCIC	P	O, RB	B	I	I		ASME III-2	
	3. Piping & Valves other	P	O	D	II	NA		B31.1.0	
	4. Other components	P	O	D	II	NA		X	
XXXIII.	[illegible] Emergency Support Facilities (See Supplement Page 2 attached)								

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TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
XXXIV	Main Steam Isolation Valves Leakage Control System (See Supplement page 2 attached)								
XXXV.	Miscellaneous Components								
	1. Reactor building polar crane	P	RB	-	I	I		X	
	2. ECCS loop level pumps	P	RB	B	I	I		ASME III-2	
XXXVI.	Reactor Building Closed Loop Cooling Water System								
	1. Pumps and heat exchangers	P	RB	C	I	I		ASME III-3	
	2. Valves, containment isolation	P	C	B	I	I		ASME III-2	
	3. Piping and valves for spent fuel pool HX; Reactor Recirc. pump cooler, ECCS pump coolers	P	RB	C	I	I		ASME III-3	
	4. Pumps and piping for motor generator MG set coolers	P	R, C	D	II	NA		B31.1.0	
	5. Piping, other	P	PC, RB	D	II	NA		B31.1.0	
	6. Valves, other	P	PC, RB	D	II	NA		B31.1.0	
XXXVII.	Equipment and Floor Drainage Systems								
	1. Sumps	P	RB,T,RW	D	II	NA		X	
	2. Pumps	P	RB,T,RW	D	II	NA		X	
	3. Piping, containment isolation	P	RB	B	I	I		ASME III-2	
	4. Valves, containment isolation	P	RB	B	I	I		ASME III-2	
	5. Cable, with a safety function	P	-	-	I	NA		X	
	6. Piping, other	P	RB,T,RW	D	II	NA		B31.1.0	
	7. Valves, other	P	RB,T,RW	D	II	NA		B31.1.0	

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TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
XXXVIII.	Miscellaneous Ventilation Systems								
	1. Battery room H & V	P	R	-	I	I		X	
	2. Screenwell pumphouse H & V	P	P	-	I	I		X	
	3. Relay and emergency switchgear H & V	P	R	-	I	I		X	
	4. Control room air conditioning including filter trains	P	R	-	I	I		X	
	5. Diesel generator room ventilation	P	R	-	I	I		X	
XXXIX.	Area Radiation Monitoring System								
	1. All components	GE	RW, T, R, RB	-	II	NA		X	
	2. High range area	P	RB	-	I	I		X	
XL.	Leak Detection System								
	1. Temperature element	GE	PC, RB	-	I	I		X	
	2. Temperature switch	GE	PC, RB	-	I	I		X	
	3. Differential temperature switch	GE	PC, RB	-	I	I		X	
	4. Differential flow switch	GE	PC, RB	-	I	I		X	
	5. Pressure switch	GE	PC, RB	-	I	I		X	
	6. Differential pressure switch	GE	PC, RB	-	I	I		X	
	7. Differential flow summer	GE	PC, RB	-	I	I		X	
	8. Reactor building floor drain sumps	P	RB	-	II	NA		X	(21)
	9. Reactor building floor drain pumps and piping	P	RB	-	II	See Note (22)		X	(22)

TABLE 3.2.1-1 (CONT'D)

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
XLI.	Fire Protection System								
	1. Water spray deluge systems	P	-	-	II	NA		X	
	2. Sprinklers, carbon dioxide systems	P	-	-	II	NA		X	
	3. Portable and wheeled extinguishers	P	-	-	II	NA		X	
XLII.	Civil Structures								
	1. Reactor building	P	RB	-	I	I			
	2. Office and service building	P	-	-	II	NA			
	3. Screenwell	P	P	-	I	I			
	4. Control building	P	C	-	I	I			
	5. Turbine building	P	T	-	II	NA		ACI-318-71	(23)
	6. Intake Canal	P	-	-	NA	I		ACI-301-66&72	
	7. Discharge tunnel	P	-	-	II	NA		AISC-70	
	8. Discharge pipe and diffuser	P	-	-	II	NA			
	9. Radwaste building	P	RW	-	I	I			
	10. Auxiliary boiler and MG set building	P	-	-	II	NA			
	Insert 1 attached								
XLIII.	Primary Containment Structure								
	1. Reinforced concrete	P	PC	-	I	I		ACI-301	
	2. Liner	P	PC	-	I	I	8/70	-	(24)
	3. Penetrations	P	PC	-	I	I	8/70	B31.7, 1969	
	4. Drywell head and drywell equipment, CRD removal and suppression chamber access hatches	P	PC	-	I	I	8/70	ASME III-B Summer 1969	
	5. Drywell personnel hatch	P	PC	-	I	I	8/70	ASME III-B Winter 1969	
	6. Personnel hatch for drywell equipment hatch (Emergency air lock)	P	PC	-	I	I		ASME III-MC Winter 1972	
	7. Downcomers	P	PC	B	I	I		ASME III-2 Winter 1972	(25)

Insert 1

TABLE 3.2.1-1

EQUIPMENT CLASSIFICATION

	Principal Component[1]	Scope[2] of Supply	Location[3]	Quality[4a] Group Classification	LILCO[4b] Quality Assurance Category	Seismic[5] Category	Purchase[6] Order Date	Principal[7] Code	Comments[8]
XLII	Civil Structures								
	11. Biological Shielding	P	P C, R, RW, T, TSC	—	I	I		ACI-318-71 ACI-301-66 & 72	
	12. Missle Barriers	P	RB, R	—	I	I		ACI-318-71 ACI-301-66 & 72	
	13. Waterproof doors	P	P, R Diesel Fuel Pump House	—	I	NA		X	
	14. Site grading	P	O	—	III	NA		X	
	15. Safety related masonary walls	P	RB, R	—	I	I		NA	

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TABLE 3.2.1-1 (SUPPLEMENT)

EQUIPMENT CLASSIFICATION

	Principal Component(1)	Scope(2) of Supply	Location(3)	Quality(4,a) Group Classification	LILCO(4,b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
XLIV	PLANT SAFETY PARAMETER DISPLAY CONSOLE								
	1. SPDS SUBSYSTEM OF ERF	P	TSC BLDG (R)	—	II	I	△	x	
	2. ELECTRICAL MODULES WITH SAFETY FUNCTION	P	R	—	I	I	△	x	
	3. SPDS DISPLAY	P	R	—	II	I	△	x	
XLV	POST ACCIDENT SAMPLE SYSTEM								
	1 POST ACCIDENT SAMPLE SYSTEM CONTROL PANEL	P	SAMPLE BLDG	—	II	NA	△	x	
	2 POST ACCIDENT SAMPLE SKID	P	SAMPLE BLDG	D	II	NA	△		
XLVI	VALVE POSITION INDICATION								
	1. VALVE POSITION SWITCH	P, G-E	RC, RB	—	I	I			
	2. DATA ACQUISITION MULTIPLEXER	P	SAMPLE BLDG	—	II	II □	△	x	
	3. SPDS DISPLAY	P	R	—	II	I	△	x	
	4. TSC/EOF DISPLAY	P	TSC BLDG EOF BLDG	—	II	I	△	x	
	5. SPDS SUBSYSTEM OF ERF	P	TSC BLDG (E)	—	II	II	△	x	
	6. TSC/EOF SUBSYSTEM OF ERF	P	TSC BLDG	—	II	I	△	x	
			EOF BLDG	—	II	II	△	x	

△ NOT YET PURCHASED

□ Actually bought to Cat. 1 / Seismic 1 standards and specifications, credit is taken for Class II only

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TABLE 3.2.1-1 (SUPPLEMENT)

EQUIPMENT CLASSIFICATION

	Principal Component(1)	Scope(2) of Supply	Location(3)	Quality(4a) Group Classification	LILCO(4b) Quality Assurance Category	Seismic(5) Category	Purchase(6) Order Date	Principal(7) Code	Comments(8)
XLVII	ACCIDENT MONITORING INSTRUMENTATION (NUREG 0578)	GE, P	RB, R, A	?	I	I	△	X	
XXXIII	Permanent EMERGENCY SUPPORT FACILITIES								
(Input page 13 of 21)	1. TECHNICAL SUPPORT CENTER (TSC)	P	TSC BLDG.	–	II	I	△	X	BUILDING SEIS I, INSTRUMENTATION NON SEISMIC
	2. EMERGENCY OPERATIONS FACILITY (EOF)	P	*	–	II	N/A	△	X	
	3. OPERATIONAL SUPPORT CENTER (OSC)	P	*	–	II	N/A	△	X	
	4. SAFETY PARAMETER DISPLAY SYSTEM (SPDS)	P	R, TSC BLDG.	–	II	I	△	X	
XXXIV (Input page 14 of 21)	Main Steam Isolation Valve Leakage Control System								
	1. Piping & valves up to first isolation valve of the inboard subsystem	P	RB	A	–	I		ASME III-1	
	2. Piping & valves to second isolation valves	P	RB	B		I		ASME III-2	
	3. Blowers	G.E.	RB	NA		I		X	
	4. Heaters	G.E.	RB	NA		I		X	
	6. Cable with safety function	P	–	NA		I		X	
	6. Electrical modules with safety function	G.E.	RB	NA		I		X	

△ NOT YET PURCHASED
* NOT FINALIZED

Item 59 - Control of Heavy Loads

The information requested to satisfy SER Open Item No. 57, "Control of Heavy Loads" has been specifically delineated in NRC letters October 22, 1980 and March 2, 1981 from D. G. Eisenhut. In accordance with the schedule provided in the referenced generic letters, the information requested in paragraphs 1 and 2 will be submitted by June 22, 1981 for Sections 2.2 and 2.3. Section 2.4 is not applicable to SER's as outlined in the March 2, 1981 letter. We do not believe that resolution of this generic issue should be carried as an open item on the Shoreham docket.

SNPS

Item # 60 - Station Blackout

In the very unlikely event that both offsite and onsite alternating current (AC) power is lost, boiling water reactors may use a combination of safety/relief valves and the Reactor Core Isolation Cooling (RCIC) system to remove core decay heat without reliance on AC power. Emergency procedures will be developed to ensure safe operation of the plant and restoration of AC power. In addition, operators will be trained to effectively deal with this event. In this light, LILCO intends to perform, as part of its low power test program, several tests verifying RCIC operability upon loss of AC power or other degraded electrical conditions. For more details, refer to our response to NUREG-0737 item I.G.1, "Training During Low Power Testing".

The procedures and most of the training described above will be completed prior to fuel load. Completion of training (low power testing) can not be accomplished until after fuel load (but prior to commercial operation).

A complete assessment of LILCO's planned facility procedures and training programs with respect to this matter will be forwarded by June 5, 1981. This is in accordance with the letter from Darrel G. Eisenhut to all Licensees of Operating Nuclear Power Reactors and Applicants for Operating Licenses dated February 25, 1981 and received by LILCO on March 6, 1981.