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VICE PRESIDENT - NUCLEAR OPERATIONS

SNRC-1424

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Response to NRC Staff's Additional Questions
Regarding LILCO's Proposed
Supplemental Containment System
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

References:
(1) Summary of Meeting With LILCO on July 21, 1987 Regarding the Design of a Supplemental Containment System (SCS), USNRC, dated July 28, 1987
(2) Request for Concurrence with the Proposed Supplemental Containment (SC) Plan and Response to Staff Concerns, SNRC-1367, dated August 28, 1987
(3) Supplemental Containment System - PRA Status, SNRC-1396, dated December 10, 1987

Gentlemen:

As discussed in our July 21, 1987 presentation to the NRC Staff, LILCO has proposed to install a Supplemental Containment System (SCS) for the Shoreham Nuclear Power Station. The SCS would consist of a gravel filter bed similar to the Swedish FILTRA Unit installed at the Barseback Nuclear Power Station in southern Sweden. During that presentation and in subsequent correspondence, LILCO requested Staff concurrence on the QA and seismic categories LILCO had established for design and construction of the SCS. LILCO proposed that the SCS be designed and constructed as a QA Category I (safety related) installation for those portions of the system within the containment boundary (i.e., to the outermost rupture disc), while the balance of the system would be classified QA Category II (non-safety related). In addition, the entire SCS will be designed in accordance with Seismic Category I design criteria. As part of the assessment of the proposed categorization, the Staff indicated that it required

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responses to those questions raised during the July presentation, plus additional questions forwarded with the Staff's summary of the meeting (Reference 1). In large part, the response to these additional questions required the completion of a full power Probabilistic Risk Assessment (PRA) for Shoreham that included consideration of the proposed SCS.

Significant progress has been made since the July presentation. On August 28, 1987, LILCO provided partial responses to the Staff questions raised during the presentation and, most recently, LILCO finalized the 100% power PRA that considers operation of Shoreham both with and without the SCS. This current PRA update includes both internal and external (i.e., seismic, fire) event initiators of accident sequences. In addition, the SCS design has been revised as reflected by the current flow diagram (SK-SCS-004) provided within the enclosed summary report, entitled, "Shoreham Nuclear Power Station Probabilistic Risk Assessment with Supplemental Containment System". These design changes, made in response to insights gained during the PRA update, improve the performance of the SCS. The primary differences between the revised design and that previously submitted to the Staff are:

- the addition of a second rupture disc in the main wetwell airspace vent line constituting the outer boundary of the primary containment;

- modifications to the FILTRA building outlet configuration to consist of two pressure relief valves and an air operated ball valve, to provide for overpressure protection and to maximize radioactive decay and dispersion of any releases.

- the addition of a second vent line from the drywell section of the primary containment to fulfill the requirements of the BWROG Emergency Procedure Guideline Revision 4 venting recommendations. Although this second vent path has not been included in the current PRA work, it is our conclusion that the additional vent further enhances the conclusions of the 100% power PRA.

The completion of the PRA allows LILCO to fulfill your earlier request, and to provide responses to the six (6) outstanding questions. As part of this response, LILCO has developed the enclosed summary report entitled, "Shoreham Nuclear Power Station Probabilistic Risk Assessment with Supplemental Containment System," which highlights the results of the 100% power PRA update effort. The current PRA update concludes that there are

significant improvements in the plant's ability to cope with severe accidents given the addition of the various hardware and procedural changes made since the 1983 PRA, and an even further improvement with the addition of the SCS filtered vent. The major conclusions of this PRA update are:

- The risks associated with the operation of Shoreham at 100% power with and without the SCS are small. For severe accidents, most core melts at Shoreham without the SCS would not result in doses to populated areas in excess of the EPA plume exposure pathway Protective Action Guide (PAG) doses at distances beyond about 0.7 mile. With SCS, such severe accidents would not exceed plume exposure pathway PAG doses beyond 0.3 mile. For severe accidents without the SCS, the probability given core melt of exceeding a 200 rem whole body dose at 0.9 mile is equal to that in NUREG 0396 at 10 miles. For severe accidents with the SCS, the probability given core melt of exceeding a 200 rem whole body dose is further reduced to an equivalent distance of less than 0.1 mile. Doses to populated areas under the USAR Design Basis Accidents will meet PAG dose standards at a distance of less than one mile, with or without the SCS.

- Changes in the Shoreham plant[1/] have resulted in a reduction in the frequency of potential core damage from the 1983 PRA results by approximately 29% to 4.2×10^{-5} per reactor year. By comparison, installation of the SCS further reduces the frequency of potential core damage by an additional 10%, to 3.6×10^{-5} per reactor year.

- No new types of internally initiated accident sequences which could lead to core melt, were identified beyond those originally discussed in the 1983 PRA.

- Eighty-two percent of all core melt conditions (with SCS) lead to containment integrity being maintained. Release rates for these cases are very small, on the order of those anticipated for the plant design basis accident.

1/ In addition to the proposed SCS, LILCO has committed to various procedural and hardware modifications since the completion of the 1983 PRA. These modifications have been accounted for in the current PRA udpate. A complete discussion of these modifications is included in the enclosed summary report entitled, "Shoreham Nuclear Power Station Probabilistic Risk Assessment With The Supplemental Containment System."

- Approximately 92% of all potential core melt conditions (with and without SCS) are characterized by a slow accident progression which results in a release to the environment that does not begin until at least 24 hours following the initiating event. For these same conditions, the frequency weighted average release duration is 18 hours with the SCS, and 16 hours without SCS.

  The remaining 8% of the potential core melt conditions are characterized as early releases. For these core melt conditions, the frequency weighted average release start time is about 9 hours with the SCS, and about 4 hours without the SCS. The frequency weighted average release duration is about 10 hours with the SCS and about 4 hours without the SCS.

- The SCS reduces the severity of land contamination given a severe accident. In general, the gravel bed overall decontamination factor associated with particulate releases is greater than 500. This leads to a reduction in land contamination by an average factor of about ten.

The results of this PRA update clearly show that there has been significant improvement in the ability of the Shoreham plant to mitigate severe accidents since the original 1983 PRA was performed. Using current unfiltered venting capabilities, overall core melt frequencies have been reduced, with the majority of these severe accidents characterized by slow progression. The inclusion of the SCS further improves the Shoreham plant by enhancing its containment capabilities and significantly filtering any potential releases to the environment. Significant is the fact that even with the SCS, no new types of internally initiated accident sequences leading to core melt have been identified.

We trust that the enclosed information is responsive to the Staff's request and will allow you to concur with the quality group categorization which LILCO has established for the SCS. It

is our desire to receive your concurrence by April 8, 1988 so as not to impact the overall project schedule. Should you need to discuss our request further, or require any additional information, please do not hesitate to contact my office.

Very truly yours,

John D. Leonard, Jr.

John D. Leonard, Jr.
Vice President - Nuclear Operations

BEG:ck

Enclosures

cc: W. T. Russell - Region I Administrator
S. Brown
F. Crescenzo

ATTACHMENT TO
SNRC-1424

Question #1: How will the activation of filtered venting be considered in conjunction with the unfiltered venting procedure in the EOP?

Response #1:

LILCO is currently in the final phase of implementing revisions to Emergency Operating Procedures (EOPs) by incorporating Revision 4 of the BWROG Emergency Procedure Guidelines. The Containment Pressure Control procedure will require remote manual venting through the existing 6 inch wetwell vent line (IT46*AOV-079A/B) in order to maintain containment pressure less than 60 psig. If containment pressure continues to rise, or the wetwell cannot be vented, operators will be directed to open an existing 6 inch drywell vent line (IT46*AOV-078A/B). These lines discharge into the Secondary Containment and have a combined capability to vent steam generated equivalent to approximately 1% of plant rated power at a containment pressure of 60 psig.

The SCS will provide the capability to vent the containment through a self-actuated 24 inch line, which vents the containment wetwell airspace to achieve maximum fission product scrubbing. In addition, two 6 inch lines will be added to provide the capability to vent the drywell to the SCS. These 6 inch vent paths are intended to mitigate the potential for drywell failure due to elevated temperatures, and to facilitate containment flooding to preserve the reactor core as part of the long term recovery measures. The SCS will be capable of handling approximately 8% of plant rated power through the self-actuated vent path, and approximately 1% of plant rated power through the new 6 inch drywell vent paths at 60 psig (see SCS flow diagram SK-SCS-004). The existing 6 inch lines that vent the containment to the reactor building will remain.

When the SCS is available for operation, the EOP on Containment Pressure Control will be revised to identify the SCS filtered vent as the primary means of containment pressure control. Use of the existing unfiltered manual containment venting paths will be restricted such that, to the maximum extent possible, containment venting will make use of the SCS. The current PRA update results are based on the assumption that no containment venting to the reactor building is used in conjunction with the SCS.

Question #2: Provide analyses to quantify the reduction of public risk.

Response #2:

The results of the 100% power Probabilistic Risk Assessment are provided in the enclosed summary report entitled, "Shoreham Nuclear Power Station Probabilistic Risk Assessment with Supplemental Containment System". The assessment considered the Shoreham plant both with the SCS (filtered venting) and without the SCS (non-filtered venting).

The major conclusions of this PRA update are:

- The risks associated with the operation of Shoreham at 100% power with and without the SCS are small. For severe accidents, most core melts at Shoreham without the SCS would not result in doses to populated areas in excess of the EPA plume exposure pathway Protective Action Guide (PAG) doses at distances beyond about 0.7 mile. With SCS, such severe accidents would not exceed plume exposure pathway PAG doses beyond 0.3 mile. For severe accidents without the SCS, the probability given core melt of exceeding a 200 rem whole body dose at 0.9 mile is equal to that in NUREG 0396 at 10 miles. For severe accidents with the SCS, the probability given core melt of exceeding a 200 rem whole body dose is further reduced to an equivalent distance of less than 0.1 mile. Doses to populated areas under the USAR Design Basis Accidents will meet PAG dose standards at a distance of less than one mile, with or without the SCS.

- Changes in the Shoreham plant[1/] have resulted in a reduction in the frequency of potential core damage from the 1983 PRA results by approximately 29% to 4.2×10^{-5} per reactor year. By comparison, installation of the SCS further reduces the frequency of potential core damage by an additional 10%, to 3.6×10^{-5} per reactor year.

1/ In addition to the proposed SCS, LILCO has committed to various procedural and hardware modifications since the completion of the 1983 PRA. These modifications have been accounted for in the current PRA update. A complete discussion of these modifications is included in the enclosed summary report entitled, "Shoreham Nuclear Power Station Probabilistic Risk Assessment with Supplemental Containment System."

- No new types of internally initiated accident sequences which could lead to core melt were identified beyond those originally discussed in the 1983 PRA.

- Eighty-two percent of all core melt conditions (with SCS) lead to containment integrity being maintained. Release rates for these cases are very small, on the order of those anticipated for the plant design basis accident.

- Approximately 92% of all potential core melt conditions (with and without SCS) are characterized by a slow accident progression which results in a release to the environment that does not begin until at least 24 hours following the initiating event. For these same conditions, the frequency weighted average release duration is 18 hours with the SCS, and 16 hours without SCS.

  The remaining 8% of the potential core melt conditions are characterized as early releases. For these core melt conditions, the frequency weighted average release start time is about 9 hours with the SCS, and about 4 hours without the SCS. The frequency weighted average release duration is about 10 hours with the SCS and about 4 hours without the SCS.

- The SCS reduces the severity of land contamination given a severe accident. In general, the gravel bed overall decontamination factor associated with particulate releases is greater than 500. This leads to a reduction in land contamination by an average factor of about ten.

Question #3: Provide analyses on the releases from the SCS in new accident sequences where the containment is recoverable.

Response #3:

As discussed in the response to question 1, the revised Shoreham EOPs require the operator to vent the containment (unfiltered venting) in order to maintain pressure below 60 psig, which is well below the ultimate containment pressure of 130 psig. This pressure is identical to the set pressure of the rupture discs which will actuate the SCS (see Section III of the enclosed summary report). Since venting will occur if containment pressure reaches 60 psig with or without SCS, it can be concluded that the SCS does not introduce new sequences in which venting would occur, even though the containment might otherwise be recoverable or not lost at all. It is also important to note that this current PRA update has not identified any new types of internally initiated accident sequences leading to core melt.

A comparison of the current PRA update results for the plant, both with and without SCS, show a shift in the accident frequency towards less severe accidents, and a similar positive shift in the release category frequency towards delayed and less severe releases. With regard to accident severity, containment venting through either the unfiltered vent flow path or SCS, primarily impacts the Class II and IV[2/] accident sequences. Class II accident sequences represent those accidents in which the buildup of decay heat in the containment induces high containment pressure and high suppression pool temperatures. Class IV accident sequences represent those accidents in which there is a failure to insert negative reactivity. The following table, which was developed as part of this current update, shows the contribution of each accident subclass to the total core melt frequency for the FILTRA (SCS) and containment vent (without SCS) cases. With the SCS, there is a shift in the relative significance of the accident subclasses from Class IIA to IIF and from Class IVA to IVF and IVG. This is significant because accident sequences in which there is a loss of containment decay heat removal which ultimately leads to containment failure, become a small part of the total core melt frequency, while accident sequences in which the containment remains intact and the releases are filtered become the major contributors to the total core melt frequency.

With regard to shifts in release category [3/] frequency, a similar comparison can be made. The following figure shows that there is a shift in the frequency of the early release categories (RC1-RC4) to categories RC-11 and RC-12, which are respectively characterized as filtered releases and events in which there is recovery of core cooling.

2/ A description of each of the plant damage states considered in the current PRA update effort is included in Section IV of the enclosed summary report entitled, "Shoreham Nuclear Power Station Probabilistic Risk Assessment With The Supplemental Containment System".

3/ A description of each of the release categories is also included in Section IV of the enclosed summary report entitled, "Shoreham Nuclear Power Station Probabilistic Risk Assessment With The Supplemental Containment System".

TABLE 1

COMPARISON OF VENTING SENSITIVITY ANALYSES

Accident Subclass Category	Plant System Configuration			
	FILTRA		6" CONTAINMENT VENT	
	Subclass Frequency (Per Reactor Year)	% of Total Class Frequency	Frequency (Per Reactor Year)	% of Total Class Frequency
TW SEQUENCES				
II A	6.9E-8	2%	8.2E-6	94%
II B	5.7E-9	<1%	8.0E-8	1%
II F	3.4E-6	98%	4.4E-7	5%
Total	3.4E[illegible]		8.7E-6	
ATWS SEQUENCES				
IV A	4.4E-7	31%	2.1E-6 (IV)	100%
IV F	6.9E-7	48%		
IV G	3.0E-7	21%		
Total	1. E-6		2.1E-6	

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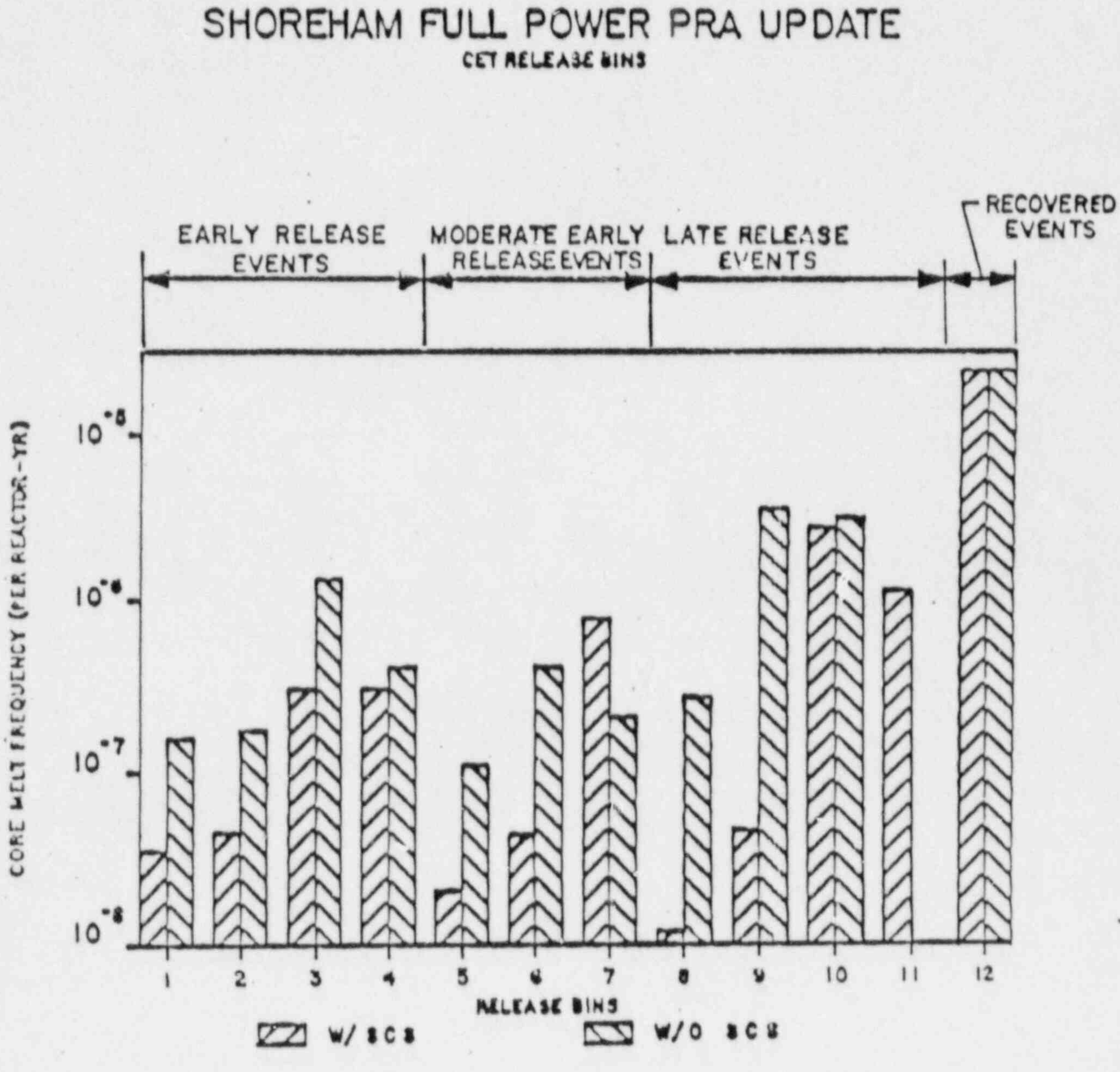

FIGURE 1

Question #4: Response of the SCS to a short duration pressure pulse in the containment.

Response #4:

There are two short duration pressure pulse events to consider: the first resulting from hydrogen detonation and the second resulting from the rapid pressure rise in the containment following melt through of the lower vessel head. During normal plant operation, the Shoreham containment is inerted in accordance with the requirements of the Technical Specifications. Considering this factor, MAAP analyses performed during the current PRA update confirm that no oxygen is produced in the containment for the representative sequences of each release category with or without the SCS. Consequently, hydrogen detonation leading to a short duration pressure spike is not anticipated.

In considering the response of the primary containment rupture discs, discussions with manufacturers of reverse buckling type rupture discs similar to those that will be used at Shoreham, established that typical response times to a pressure rise for a disc of this type are on the order of 10 milliseconds. The pressure rise following a vessel melt through, as calculated by the MAAP code, was analyzed to determine the duration of the containment pressure spike for each of the representative sequences. The results of the analyses revealed that the length of the pressure spike was much longer than the response time of the rupture discs.

Thus LILCO concludes that for events producing short duration pressure pulses that result in wetwell pressures that exceed 60 psig, the rupture discs will open such that SCS will prevent overpressurization of the containment.

Question #5: Analyses of accident sequences which could lead to pump cavitation and core damage because of the pressure relief, although containment integrity is maintained.

Response #5:

Accident sequences which lead to core damage as a result of pump cavitation, involve those sequences in which the ECCS pumps could fail due to the loss of NPSH. Loss of pump NPSH as a result of pressure relief in the wetwell airspace following actuation of the SCS, have been accounted for in the Class IIF and IVF accident sequences.[4/] However, since NPSH is a function of a number of parameters which may vary in each accident sequence, specific analysis to quantify the explicit contribution of pump cavitation on the overall core damage frequencies, was not performed. Rather, MAAP sensitivity calculations of available LPCI pump NPSH were performed, and used as input to develop an overall enveloping frequency of ECCS pump failure.

Question #6: Potential for the containment to approach a negative pressure in the event that the containment spray is activated.

Response #6:

LILCO has not begun its development of the plant operating procedures for use with the SCS. Nevertheless, important insights concerning containment spraying with the SCS installed were gained from the development of the Shoreham specific unfiltered venting procedures, along with analytical work performed in support of the current PRA update. These insights will be used as input in developing the final operating procedures for the containment spray system when used in conjunction with the SCS.

4/ The results of the current PRA update reveal that accident subclass IIF represents 9% of the total core melt frequency. Accident subclass IVF represents 1% of the total core melt frequency. As a result, loss of ECCS pumps due to NPSH concerns represent only a small percentage of the overall core melt frequency.

The Shoreham specific unfiltered venting procedures (i.e., without the SCS) conform with Revision 4 of the Emergency Procedure Guidelines (EPGs) compiled by the BWROG. They include an exclusion plot of containment pressure versus drywell temperature. This plot provides the operator with information to be used in deciding whether or not to activate containment sprays. Implicit in the development of this plot, was the conservative assumption that the wetwell to drywell vacuum breakers do not operate, thereby maximizing the drywell floor differential pressure. The exclusion plot ensures that the design limits for the containment vacuum and upward drywell floor differential pressure will not be exceeded.[5/] A similiar exclusion plot will become part of the procedures for operation of the containment spray system in conjunction with the SCS.

In addition, as part of the current PRA update, an analysis was performed to determine an acceptable range for containment spray flow, to ensure that the integrity of the containment and the wetwell to drywell vacuum breakers would be maintained. The analysis considered a number of factors, including:

- The rate of depressurization in the drywell and its effect on the integrity of the wetwell to drywell vacuum breakers.

- The rate of depressurization in the drywell due to cooling of nitrogen and its effect on the integrity of the wetwell to drywell vacuum breakers.

- The final drywell pressure based on the aforementioned depressurization rates.

The results of the analysis show that for Shoreham, a containment spray flow range of 250 to 2000 gpm would maintain the containment and wetwell to drywell vacuum breakers within the design basis, while still providing adequate debris cooling and fission product removal. This insight will also become part of the input used to finalize the operating procedures for the containment spray system with SCS installed.

5/ For Shoreham, the containment vacuum design limit is 10 psia. The upward drywell floor differential pressure limit is 5.5 psid.