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MEMORANDUM FOR: Darrell G. Eisenhut, Director
Division of Licensing

FROM: Themis P. Speis, Director
Division of Safety Technology

SUBJECT: PRELIMINARY REVIEW OF SHOREHAM PRA STUDY

We have completed our preliminary review of the Shoreham probabilistic risk assessment (PRA) study. The current Shoreham PRA study, performed by Science Applications Inc. (SAI), considers only internal events (including internal flooding, but not including fire) and considers the frequencies of radioactive releases of various magnitudes but does not consider ex-plant consequence. The report on ex-plant consequence analysis, which has not been submitted, will be based on the results of the work performed by Pickard, Lowe, and Garrick (PLG). The Reliability and Risk Assessment Branch, with technical assistance from Brookhaven National Laboratory (BNL), has reviewed the internal event analysis, and the Reactor Systems Branch (RSB) in the Division of Systems Integration (DSI), also with BNL technical assistance, has reviewed the containment failure and radionuclide release analysis.

The Shoreham PRA study includes flooding in the reactor building initiated by an internal event. However, fires and external events such as earthquakes are not considered in the PRA study.

We and our contractors believe that the Shoreham PRA study is a good and comprehensive piece of work within its stated scope. The Long Island Lighting Company (LILCO) estimate of the total core vulnerable frequency at Shoreham is about 5×10^{-5}/reactor-year.

The Shoreham PRA study indicates that loss of coolant makeup following a transient challenge results in about 58% of the total core vulnerable frequency. Loss of containment heat removal following a transient challenge results in about 16% of the total frequency. Anticipated transients without scram (ATWS) sequences with a failure of alternate rod insertion (ARI) result in about 25% of the total frequency. Loss of offsite power (LOOP) events result in about 20% of the total frequency. There are about 20 sequences which contribute to 80% of the total core vulnerable frequency. There appears to be no single risk outlier which, if it is removed, would significantly reduce the total frequency.

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Based on the BNL requantification, we estimate that the total core vulnerable frequency at Shoreham is about 1×10^{-4}/reactor-year. Our review indicates that ATWS events contribute about 40% to the total frequency and LOOP events contribute about 23% to the total frequency.

The comparison between the Shoreham estimates of core vulnerable frequencies and the BNL estimates are given in Table 1 in the BNL report.

Our review does not identify any safety issue that needs immediate action. We note that ATWS events at Shoreham contribute significantly to the total core vulnerable frequency. However, we believe that the implementation of the ATWS rule requirements would reduce the contribution to the total core vulnerable frequency due to ATWS events.

We note that the Shoreham PRA study has been used to address two issues, namely, flooding in the reactor building and reactor water level measurement system. These issues as well as the associated actions are discussed in Enclosure 1 and in our previous memoranda[1,2].

Since the reactor building at Shoreham is an open annulus, a break in the high-pressure coolant injection (HPCI) or reactor core isolation cooling (RCIC) steam lines in the reactor building with a subsequent failure to isolate the break may have damaging effects on the safety equipment in the reactor building. This issue is still under study and will be addressed in our final report.

With respect to the Shoreham containment response and radionuclide release analyses, BNL has completed their preliminary review and submitted to the RSB in the DSI. The evaluation from the RSB will be included in our final report.

Our review of the ATWS events at Shoreham indicates that there is a large discrepancy between the determinstic analyses regarding the magnitude of the reactor power when the reactor water level is maintained at the top of the active fuel (TAF). We believe that the times available for operators to take critical actions are dependent on the magnitude of the reactor power. We request the RSB in the DSI to provide us with feedback on this issue.

In addition, we request that our evaluation be sent to LILCO for comments. We request that all feedback and comments from LILCO as well as other NRR divisions be forwarded to us in three weeks to allow us sufficient time for consideration in our final evaluation.

Enclosure 1 contains a summary of our preliminary findings and discussions of areas that may need further resolution. Enclosure 2 contains the preliminary report from BNL.

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With this evaluation the Phase I (preliminary review) work on the Shoreham PRA study is complete.

Original signed by
Frank H. Rowsome

for Themis P. Speis, Director
Division of Safety Technology

Enclosure:

1. Preliminary Review of Shoreham PRA Study
2. "A Review of the Shoreham Nuclear Power Station Probabilistic Risk Assessment," BNL, November 1984

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ENCLOSURE 1

PURPOSES OF PRA

The Shoreham nuclear power station under the ownership and management of LILCO was built by Stone & Webster using a BWR-4 reactor and a Mark II containment. LILCO performed the Shoreham PRA study on its own initiative in order to evaluate plant response to hypothetical accident sequences and to assess their risk. LILCO intends to maintain the event tree/fault tree models consistent with the current design of Shoreham.

INTERNAL EVENT REVIEW

The core damage sequences at Shoreham are divided into five classes, depending on the timing between onset of core damage and onset of containment failure:

(1) Class I sequences involve the early core damage sequences due to loss of makeup to the core prior to containment failure.

(2) Class II sequences involve the sequences in which the core is intact but the containment fails later due to loss of containment heat removal.

(3) Class III sequences are core damage sequences due to loss of coolant accident (LOCA) in the containment.

(4) Class IV sequences are ATWS sequences with core damage prior to containment failure.

(5) Class V sequences are due to LOCA outside the containment.

LILCO estimates that the total frequency of core vulnerable accident sequences at Shoreham is about 5.5×10^{-5}/reactor-year. The Shoreham PRA study defines the core vulnerable state as an end state of the plant in which the reactor core or containment integrity is challenged. Certain operator actions, including operator actions "in extremis" can be used in a core vulnerable state to prevent core melt.

The Shoreham PRA study indicates that loss of coolant makeup following a transient challenge results in about 58% of the total core vulnerable frequency. Loss of containment heat removal following a transient challenge results in about 16% of the total frequency. ATWS sequences with a failure of ARI result in about 25% of the total frequency. LOOP events result in about 20% of the total frequency. There are about 20 sequences which contribute to 80% of the total core vulnerable frequency. There appears to be no single risk outlier which, if it is removed, would significantly reduce the total frequency.

Based on the BNL requantification, we estimate that the total core vulnerable frequency at Shoreham is about 1.5×10^{-4}/reactor-year. Our review indicates that ATWS events contribute about 40% to the total frequency and LOOP events contribute about 23% to the total frequency.

ATWS EVENTS

The Shoreham estimate of the core vulnerable frequency due to ATWS events is 1.8×10^{-5}/reactor-year and the BNL estimate is 5.9×10^{-5}/reactor-year. The discrepancy between the Shoreham and the BNL estimates is mainly due to the difference in the data for transient initiator frequency.

The transient initiator frequencies given in the Shoreham PRA study are mainly based on the data in NP-801[3]. The updated data given in NP-2230[4] is used by BNL to estimate the transient initiator frequencies which are higher than the Shoreham estimates. However, we believe that the data on isolation transient frequency in NP-2230 may be conservative. The reasoning is as follows: Most BWR-4 plants use the water level setpoint at level 2 (L2) to initiate main steam isolation valve (MSIV) closure. However, at Shoreham the MSIV closure setpoint is lowered from L2 to level 1 (L1). It appears that if other BWRs had used L1 for MSIV closure, the frequencies of isolation transients should have been less than what are in NP-2230 because there is close to 8 feet of water between L1 and L2 which allows operators more time to prevent an isolation event. The BNL analysis of the transient initiator frequencies does not take credit for the conservatism in the data.

The probability of scram failure used in the Shoreham PRA study is 3×10^{-5}/demand which is taken directly from NUREG-0460[5] as an appropriate point estimate. The probabilities of scram failures due to mechanical and electrical common-mode failures are assumed to be 10-5/demand and

2×10^{-5}/demand respectively. Shoreham has incorporated two design features for preventing or mitigating an ATWS event:

(1) Shoreham has incorporated ARI which is a diverse and redundant backup method of signaling control rod insertion. This design feature substantially reduces the probability of an electrical common-mode failure to insert control rods.

(2) Shoreham has incorporated the recirculation pump trip (RPT) which introduces voids and negative reactivity to the core and therefore reduces the reactor power.

The estimates of the frequencies of ATWS core vulnerable sequences, given in the Shoreham PRA study and by BNL, include the benefits of these design features.

The standby liquid control system (SLCS) at Shoreham consists of an unpressurized tank for storing sodium pentaborate solution, two positive displacement pumps each with a capacity of 43 gpm. The SLCS is manually initiated from the control room and only one SLCS pump can operate at one time. We note that the Shoreham and the BNL analyses use 43 gpm as the capacity of the SLCS at Shoreham. We believe that, in order to meet the regulations stated in the ATWS rulemaking analysis, LILCO may be required to upgrade the SLCS either by increasing the SLCS capacity from 43 gpm to 86 gpm or by increasing boron concentration in the solution and upgrading the heat tracing to prevent the more concentrated boron from precipitating. Either measure would definitely decrease the ATWS core vulnerable frequency.

Our review of the Shoreham ATWS events indicates that operator actions are most crucial for mitigation. For example, the staff task force on ATWS rulemaking determines that if an ATWS isolation event occurs and the SLCS capacity is assumed to be 86 gpm, the operator must initiate SLCS within two minutes after the beginning of the transient in order not to exceed the suppression pool limit of 200°F. For all ATWS isolation events with the current SLCS capacity of 43 gpm, the suppression pool limit of 200°F will be exceeded slightly even if the operator immediately follows the emergency procedure guidelines (EPG) and actuates the SLCS.

In an ATWS isolation event, the times available for operator actions are dependent on the reactor power which also depends on when the poison is injected and on how close to the TAF the reactor water is maintained at. There is some discrepancy between the BNL estimate of the reactor power in an ATWS event and the estimate used in the regulatory analysis for ATWS rulemaking. For example, BNL believes that when the water is at the TAF, the reactor power is between 15% to 20%. The regulatory analysis uses an estimate of 8% power. We believe these considerations which can be only obtained from a thorough deterministic analysis have significant impact on the success of operator actions.

LOSS OF OFFSITE POWER (LOOP)

The Shoreham PRA study uses 0.08/reactor-year as the initiator frequency of a LOOP event. This estimate is based primarily on the fossil plant data

from 1965 to 1981 in which four LOOP events occurred in 61.5 plant-years. The estimate of 0.08/reactor-year is obtained by dividing five LOOP events (four occurrences plus one hypothesized incipient occurrence) by 61.5 plant-years. We believe that instead of the fossil plant data, it is more appropriate to use the nuclear plant data on LOOP events as recently published in NSAC-80[6]. Using this data base and a Bayesian approach, the BNL estimate of the LOOP frequency at Shoreham is 0.15/reactor-year. The BNL approach takes the data from NSAC-80 which corresponds to the NPCC and uses this data to obtain a distribution for the LOOP frequency. Implicit in this process is the assumption that a plant taken randomly from the Northeast Power Coordinating Council is representative of the Shoreham plant, for estimating the LOOP frequency.

The Shoreham PRA study assumes the battery will sustain its load for 24 hours during blackout events. BNL assumes that the depletion rate of battery is higher and the battery will be lost in 10 hours. The Shoreham PRA study does not take into account that during blackout events the level instrumentation readings in the control room are lost because they are not powered from DC. This may make it difficult for the operators to follow procedures and to control HPCI/RCIC systems. Based on the BNL analysis, these events contribute about 1×10^{-5}/reactor-year to the total core vulnerable frequency.

We note that the diesel generator data used in the Shoreham and BNL analyses, for example, failure to start probability of 2×10^{-2}, is based

on generic data originating from LERs given in NUREG/CR-1362[7].

Since Shoreham has TDI diesel generators which experienced problems, Shoreham is committed to install other permanent diesel generators which will be qualified. At this point, we are not sure when the permanent diesel generators would be operational (probably in a year or so). However, in the interm, there is a 20 MW Pratt and Whitney gas turbine on site. In addition, there are four temporary mobile 2.5 MW EMD diesel generators on site. On a given LOOP event, the gas turbine will automatically start. If the gas turbine fails to start or cannot be loaded, the temporary diesels would start and synchronize to provide the emergency power. After the permanent diesel generators are installed and shown to be operable, it is reasonable to use industry-averaged data for assessing the reliability of the diesel generators at Shoreham. In fact, if both the TDI diesel generators and the other permanent diesel generators are present, then the analysis of LOOP sequences in the BNL anaylsis would be conseravtive.

LOCA OUTSIDE CONTAINMENT

The Shoreham PRA study considers unisolated LOCAs outside the containment and into the reactor building. The two important contributions to LOCA outside the containment are interfacing LOCAs and HPCI/RCIC line breaks.

INTERFACING LOCA

The low-pressure systems such as residual heat removal (RHR)/low pressure coolant injection (LPCI) system are usually separated from the high-pressure

primary system by testable check valve and motor-operated valve (MOV) in series. If the valves are open due to valve failures and/or human errors, the primary fluid would enter the low-pressure system, possibly rupturing the system and producing an interfacing LOCA outside the containment.

The Shoreham PRA study estimates that the core vulnerable frequency due to an interfacing LOCA is 3.7 x 10^{-8}/reactor-year. We note that the Shoreham PRA study uses fault trees to quantify the interfacing LOCA frequencies for the low-pressure systems. However, we find that the fault trees are not properly modelled. For example, spurious openings of the MOV due to false signals are not considered.

Based on the licensee event reports (LERs) related to interfacing LOCA events in BWRs as discussed in the BNL report, we estimate that the unavailability of a testable check valve is 2 x 10^{-2}/reactor-year. With respect to the MOV failure, we believe that the dominant failure mode of a MOV is due to spurious opening. We assume 10-3 as the conditional probability of spurious opening of a MOV. Finally we assume that the conditional probability of a rupture in a low-pressure system with intrusion of the primary fluid is 10^{-1}. We arrive at 2 x 10^{-6}/reactor-year for an estimate of the core vulnerable frequency due to interfacing LOCAs.

We recognize that the consequences due to an interfacing LOCA at Shoreham are

severe because emergency core cooling system (ECCS) pumps, motor control centers and electrical equipment may be inoperable as a result of flooding or environmental conditions. However, there is an ongoing effort in the Division of Safety Technology (DST) to prioritize the generic issue of an interfacing LOCA in a BWR. The operability of the testable isolation check valve and MOV will be included in the generic issue.

HPCI/RCIC LINE BREAK

A release of the primary fluid outside the primary containment into the reactor building due to a non-isolable high energy line break (HELB) may impact the operability of the safety systems and components in the reactor building. This is because the reactor building at Shoreham is an open annulus and there is lack of separation between the safety components.

The Shoreham PRA study has considered the HPCI system in the analysis of HELB becuase the HPCI steam line is of 10" diameter. The PRA study does not consider the RCIC system because the RCIC steam line is of 3" diameter and there would be a large amount of time for recovery action given a break in the RCIC steam line.

We note that the inboard isolation valves on the HPCI/RCIC steam lines are open and the outboard isolation valves are closed, unlike many other BWRs. A break downstream of the outboard isolation valves would not produce a LOCA unless the outboard isolation valves are not in their normally closed

position, which is a low probability event. Therefore the frequency of LOCAs caused by breaks downstream of the outboard isolation valves has negligibly low frequency.

BNL estimates that the contribution to the total core vulnerable frequency from HPCI/RCIC steam line breaks are very small, about 10^{-8}/reactor-year. We note that the BNL analysis assumes the inboard MOV will close upon a demand signal generated due to sensing the conditions of a steam line break. The BNL analysis does not address the adequacy of the MOV qualification under the steam line break conditions.

REACTOR WATER LEVEL INSTRUMENTATION ANALYSIS

BNL estimates that the core vulnerable frequency due to reactor water level instrumentation failures is 1.2×10^{-5}/reactor-year which is about 10% of the frequency of the Class 1 core vulnerable sequences. The LILCO estimate is 3.1×10^{-6}/reactor-year. The discrepancy between the LILCO and the BNL estimates is mainly due to the following:

(1) BNL has examined the LER data and determined that there were two events in which operator error during maintenance resulted in the failures of the second reference leg. BNL estimates that the operator error is 1.9×10^{-4} and used this value to quantify the core vulnerable frequency.

(2) In the event tree for sequences of loss of a DC bus following reactor water level instrument line leak, BNL places the operator action for recognizing the need for injection before the injection of HPCI/RCIC systems. If the operator fails to recognize the situation in a timely manner, BNL assumes that core will be uncovered.

(3). The consequences due to miscalibration of water level instrumentation on the alternate leg are similar to that due to loss of the DC bus. This sequence is not explicitly modelled in the Shoreham PRA study. BNL has included this sequence in the analysis.

We note that the Shoreham reactor water level measurement system has been previously reviewed under an EG&G technical assistance contract. This is in response to the TMI requirements stated in NUREG-0737[8] Item II.F.2, "Instrumentation for Detection of Inadequate Core Cooling." As discussed in the memorandum dated October 27, 1983 to L. Rubenstein, DSI from F. Rowsome, Division of Safety Technology (DST) on "Review of BWR Water Level Measurement System," the results of the review indicate that after implementation of improvements as recommended in SLI-8211[9], the core damage frequency due to water level measurement failures at Shoreham is small when compared to the total core damage frequency in recent PRAs for BWRs. Furthermore, the risk of offsite consequences due to water level measurement failures is judged to be even smaller because most of the related accident sequences do not involve gross failure of containment during the core damage events (These sequences belong to Class I).

In a memorandum dated March 30, 1984 on "Shoreham Flooding" from A. Thadani, DST to A. Schwencer, Division of LIcensing (DL), we have provided our findings on the Shoreham flooding analysis. We estimate that the mean value of the core vulnerable frequency of accidents initiated by flooding in the

reactor building at Shoreham is 2×10^{-5}/reactor-year which is higher than the LILCO estimate of 4×10^{-6}/reactor-year.

Our conclusion is that the flooding sequences do not contribute significantly to the total core vulnerable frequency. However, we identify some potential deficiencies in the Shoreham alarm response procedures for mitigating a flood. We note that the human error probability used by BNL assumes good alarm-response procedures. The core vulnerable frequency may be higher than that estimated unless the procedures are corrected. We believe that Region I personnel would be involved with verifying the needed procedure revisions, as stated in memorandum dated May 4, 1984 on "Shoreham Reactor Building Interval Flooding Protection" from D. Eisenhut, DL to R. Starostecki, Division of Project and Resident Program.

CONTAINMENT RESPONSE AND RADIONUCLIDE RELEASE ANALYSIS

With respect to the containment response and radionuclide release analyses in the Shoreham PRA study, there is an ongoing effort in the RSB in the DSI. In addition, RSB is assisted by technical review effort under a BNL contract (FIN A-3772). BNL has completed their preliminary review of the Shoreham containment response and radionuclide release analyses. The evaluation from the RSB will be included in our final report.

REFERENCES

1. Memorandum dated October 27, 1983 to L. Rubenstein, DSI from F. Rowsome, DST, "Review of BWR Water Level Measurement System."

2. Memorandum dated March 30, 1984 to A. Schwencer, DL from A. Thadani, DST, "Shoreham Flooding."

3. EPRI NP-801, "Anticipated Trnasients - A Reappraisal," July, 1978.

4. EPRI NP-2230, "Anticipated Transients - A Reappraisal," January, 1982.

5. NUREG-0460, "Anticipated Transients Without Scram For Light Water Reactors," December, 1978.

6. NSAC-80, "Loss of Offsite Power at U.S. Nuclear Power Plants Through 1983," July, 1984.

7. NUREG/CR-1362, "Data Summaries of Licensee Event Reports of Diesel Generators at U.S. Commercial Nuclear Power Plants," March, 1980.

8. NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

9. SLI-8211, "Review of Shoreham Water Level Measurement Systems," S. Levy Inc., September, 1982.