I request that this issue be evaluated by DST and a priority for the investigation assigned. If you have any questions concerning additional details, please contact W. Hodges at X29410.

OCT 18 1983

MEMORANDUM FOR: Themis P. Speis, Director, Division of Safety Technology

FROM: Roger J. Mattson, Director, Division of Systems Integration

SUBJECT: REQUEST FOR PRIORITIZATION OF GENERIC SAFETY ISSUE - FAILURE OF HPCI STEAM LINE WITHOUT ISOLATION

cc: W. Minners
F. Rowsome
J. Ebersole, ACRS

Jesse Ebersole recently expressed, informally, a concern regarding a break in a high pressure coolant injection (HPCI) steam line with subsequent failure to close of redundant, but untestable, isolation valves. The steam from the unisolatable break could discharge into personnel spaces or spaces containing critical machinery. The Reactor Systems Branch made a preliminary evaluation of the concern and concluded that the concern is valid. The bases for the concern are outlined below.

The HPCI system for a typical BWR/4 consists of a steam turbine driving a main pump and a booster pump though a reduction gear, valves, high-pressure piping, water sources, and instrumentation designed to pump water into the reactor vessel under certain LOCA conditions associated with high reactor pressure. The HPCI system turbine is driven by steam extracted from the main steam lines at a point upstream of the main steam isolation valves (MSIVs). Two isolation valves, one inside containment and one outside containment, provide isolation in case of a break in the HPCI steamline.

The isolation valves are specified to open or close within 15 seconds. Because of steam supply limitations in test facilities, the opening characteristics are tested by valve vendors but not closing characteristics. Therefore, the capability of the valves to close under HPCI steam line break conditions has not been tested. Some valve experts believe that more force is required to open the valves than to close the valves but this has not yet been verified directly Hence, the primary basis for this concern is the potential high probability that both of these isolation valves will fail to close on demand.

With regard to the consequences of an unisolatable break, the HPCI steam line typically traverses the torus room. There are generally compartments in the torus room which house RCIC and ECC pumps and associated equipment. At many BWR/4 plants, the doors to these compartments are left open to prevent floating of the torus under extreme flooding conditions. Therefore, if the HPCI isolation valves failed to close, these systems would be subject to an environment more severe than the design environment.

One possible solution to this problem is to require that the outboard isolation valve be kept normally closed. This would impact the availability of HPCI slightly but would reduce the probability of a HPCI steam line break.

8311020209
3pp

I request that this issue be evaluated by DST and a priority for the investigation assigned. If you have any questions concerning additional details, please contact W. Hodges at X29410.

Original signed by:

Roger J. Mattson, Director
Division of Systems Integration

cc: W. Minners
F. Rowsome
J. Ebersole, ACRS

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

2/6/85

Note to: R. Caruso

From: E. Chow

Subject: Arrange a meeting between BNL and LILCO contractors on PRA comments

Please arrange a working-level meeting at BNL between BNL & LILCO contractors on LILCO comments regarding BNL draft report on PRA review. Telecon BNL indicated to me that March 18, 19, 21, 22, and 25 of 1985 are preferred dates.

P.S. When are we getting additional info on
(i) testing procedures for isolation valves and
(ii) power supply verification for RCIC valve?

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 3 1 1984

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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3pp.

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.

DEC 3 1 1984

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

Frank Rowsome for TPS

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

cc: H. Denton
E. Case
R. Vollmer
R. Bernero
H. Thompson
A. Schwencer
G. Burdick
F. Rosa
B. Sheron
D. Ziemann
G. Thomas
M. Wigdor
D. Silver
R. Caruso
M. Campagnone

Contact: 492-4727

FEB 19 1985

NOTE FOR: Themis P. Speis

FROM: Arthur Buslik

SUBJECT: INTERIM REPORT ON HPCI/RCIC/RWCU LINE BREAK IN SHOREHAM REACTOR BUILDING

This note gives a summary of the status of work related to HPCI/RCIC/RWCU line breaks in Shoreham reactor building, and gives our plan for addressing this issue. We understand that Harold Denton may wish to be briefed on this progress. The concern is that the isolation valves in these lines may not be able to close under blowdown conditions. Generally speaking, they have not been qualified for these conditions.

(1) Current Knowledge

(i) We estimate that the core-vulnerable frequency due to HPCI/RCIC line breaks is about 2×10^{-7}/reactor-year, even if the inboard isolation valve fails to isolate. The contribution of the HPCI line break dominates; the RCIC line is of only 3" diameter; the operator has more time to depressurize the reactor and recovery is more likely.

(ii) The outboard isolation valves are normally closed at Shoreham, while in most BWRs the outboard isolation valves are open. Moreover, the piping between the two isolation valves at Shoreham is of "break-exclusion" type, and is assumed to have an order of magnitude lower failure probability than other primary piping. The estimate of the core-vulnerable frequency at Shoreham due to a HPCI line break takes into account these two considerations.

(iii) We note that the BNL analysis of the HPCI line break gives credit for use of the condensate system. The condensate system is estimated to have a 20% chance of failure for these sequences. The possibility of the break causing failure of the ability to use the condensate system was investigated. The only identified dependency was the effect of the steam environment on the motor control centers (MCCs) for valves in the feedwater line; these MCCs are located in the reactor building annulus. However, it was found that these MCCs would very likely not be affected. They are at a higher elevation than the HPCI line; they are on two opposite sides of the containment; and they are in enclosed cubicles, and are protected from the environmental conditions in the reactor building, according to information obtained by BNL and verified in an informal conference call between the staff and LILCO on January 31, 1985.

Contact:
Ed Chow, RRAB
49-24727

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(4)

(iv) Except for MSIVs, qualification of an isolation valve to determine its ability to close under blowdown conditions is generally not done by test, as far as we know. In response to ACRS questions on valve qualification, Hope Creek recently submitted a data sheet from a valve closure test conducted by Wyle Laboratories. The test data indicated that the valve was capable of closing in about 2 seconds against a differential pressure of 1370 psig. The test demonstrated that the valve differential pressure was never as large as the initial upstream pressure during the closing cycle. This introduced further margin. However, no tests or analysis under blowdown conditions have been performed for these valves. For Limerick, the valves were shown by analysis to be capable of closing during blowdown conditions.

For Shoreham, we have obtained the test data for closing the HPCI and RCIC isolation valves. The tests were performed by the valve vendor, Velan Engineering Company. The test data for the RCIC isolation valves indicated that these valves closed in about 16 seconds under a differential pressure of 1135 psig. The test data for the HPCI isolation valves indicated that, under a differential pressure of 1135 psig, the HPCI inboard isolation valve closed in about 17 seconds and the HPCI outboard isolation valve closed in 44 seconds. LILCO is still trying to retrieve test data on RWCU isolation valves and the procedures for testing HPCI/RCIC/RWCU isolation valves from Velan. We have received no schedule from LILCO as to when they will submit this test data and procedures to us. We will ask Equipment Qualification Branch (EQB) to review this information in order to determine if the valves can close under blowdown conditions.

(2) Planned Future Efforts

The preliminary BNL review of the Shoreham PRA study did not explicitly address the RWCU line break because the RWCU line is 6" in diameter and is much smaller than the HPCI line which is 10" in diameter. If a RWCU line break occurs, there is more time for the operator to take recovery action. However, the RWCU line is always open, so that the advantage of a closed outboard isolation valve, as is the case with the HPCI line, is lost.

BNL is pursuing this issue. Furthermore, BNL is examining other line breaks in the reactor building in addition to breaks in HPCI, RCIC and RWCU. We believe that in general the other lines are less than 4" in diameter and there would be greater time for the operator to take recovery action. We have expanded the BNL contract effort to devote more manpower and resources to this issue. BNL will perform sensitivity analysis and provide estimates of core-vulnerable frequencies due to these breaks, assuming that the isolation valves fail to isolate during blowdown conditions (the probability of the valve failing to close is assumed to be 1). We expect BNL to complete this effort by March 8, 1985.

In addition to the sensitivity studies, our program of effort includes assessing whether the valves can close under blowdown conditions. The schedule for this

FEB 19 1985

depends on when LILCO submits the test data and procedures; the length of time needed by EQB to perform the review will depend on the nature of the material submitted.

Arthur Buslik

Arthur Buslik, Section Leader
Reliability and Risk Assessment Branch

cc: D. Eisenhut
R. Bernero
H. Thompson
R. Vollmer
A. Schwencer
B. Sheron
M. Hodges
G. Thomas
[redacted]
R. Wright, EQB
R. Silver

BLDG-130

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Department of Nuclear Energy

March 1, 1985

Mr. George Thomas
Reactor Systems Branch
Division of Systems Integration
U.S. Nuclear Regulatory Commission
Mail Stop P-1132
Washington, D. C. 20555

Dear George,

I have enclosed the revised list of information items for the containment response review which you requested (Attachment 1). I have included a schematic of the secondary containment (reactor building) indicating the expected failure location (see Appendix M of the Shoreham PRA) and the water relocation for gross overpressure failures (Attachment 2).

The previously transmitted BNL decontamination factors are included as Attachment 3. These are time-averaged pool scrubbing factors based on SPARC (NUREG/CR-3317) and do not take credit for in-vessel retention and primary containment hold-up. Note that for the Class-I sequences the suppression pool is subcooled and that the BNL decontamination factors are high and in substantial agreement with the Shoreham PRA results. However, for Class-IV sequences, the pool is heated to saturation before core melt and the BNL decontamination factors are much lower. For the Class-IV ATWS with failure at the basemat (γ''), it is assumed that the pool is relocated to the annular region of the reactor building which surrounds the primary containment (see Attachment 2). Thus, the in-vessel release through the SRV's see the same scrubbing as the ex-vessel release through the vent pipes. The slight difference in the decontamination factors (9 compared to 14) for the two releases depend on gas blowing rates at the time the scrubbing occurs.

If you have any questions, please call.

Very truly yours,

Ken

Kenneth R. Perkins

KRP:tr
Attachments
cc: W. Y. Kato (w/attachments)
R. A. Bari " "
W. T. Pratt " "
J. Rosenthal (w/attachments)
M. Wohl " "
F. Eltawila " "

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ATTACHMENT 1

REQUEST FOR INFORMATION

1. Table II of Appendix M gives different pressure limits for the longitudinal reinforcement bars at the base of the containment and in the wetwell region. However, the longitudinal bars appear to be continuous and should therefore have the same stress. Please explain the basis for the different results.

2. Table II of Appendix M indicates that the shear bars at the base and drywell head have the lowest pressure holding capability (121 psi and 120 psi, respectfully) but the discussion indicates that the additional reinforcement will preclude this failure mode. Since the containment failure mode is a key ingredient of the release estimates, please provide a quantitative estimate of the additional shear strength provided by the non-shear reinforcement bars.

3. If shear failure is precluded as discussed in Section 3.2 of Appendix M, "it appears that the ultimate capacity is controlled by the yield of the longitudinal and the hoop bars at about 123 psi." These two failure modes appear to be very important to subsequent fission product release (particularly for Class IV ATWS) since they will both occur in the wetwell region. Please provide an estimate of the size, location and direction (vertical or horizontal) containment failures for each of the three possible failure modes.

4. Section 3.6 of the PRA takes credit for containment leakage which will prevent gross containment failure for all pressurization rates except the very rapid pressurization associated with large breaks. However, the structural analysis by Stone and Webster (Appendix M) did not identify any significant source of leakage. The basis for the expected leakage source and the leakage rate as a function of pressure should be provided.

5. The basis for the partitioning between release category 10 and 11 (no pool bypass vs. partial pool bypass) should be provided. The phenomenological basis for the estimate of only 10% bypass should be provided. Preliminary results from IDCOR indicate that for some BWR sequences the vessel will fail with only 20% of the core molten. Presumably 80% of the melt release would bypass the SRV's and be released into the drywell.

6. The basis for binning into release categories is poorly described and the transfer from Tables H.4-8 etc. into the 16 release categories is difficult to interpret. A table listing the specific sequences which are binned into each category should be provided.

7. The lack of R_5 sequences in the release categories makes it apparent that these releases have been binned "downward" into the lesser release category R_4. The basis for this "downward" binning and any other sequences that are moved to less severe categories should be provided.

8. Table H.4-25 appears to be incomplete in that it does not include sequences D6 and D8. The completed table should be provided.

9. The source escape fractions used for end state screening (Table 3.6-10) appears to be quite arbitrary yet it greatly influences the importance ranking. In particular: the use of I as the surrogate for melt release ignores the fact that there are noble gases in the melt release which will not be scrubbed at all; the use of a large scrubbing factor (500) for C_4 transients is inappropriate since most of the melt release will be released directly to a failed containment; the reduction factor of 0.01 for γ" failures is indefensible since the event tree precludes everything but large ruptures where the pool will be blown out into the reactor building at high pressures.

   Table 3.6-10 should be replaced by a table with defensible reduction factors. As a minimum the table should include a separate category for C_4 transients, which recognizes the defined sequence of events (containment failure before core melt). In addition, a detailed justification for each reduction factor should be provided along with the numerical results of the ranking process. This revised table will provide the basis for our independent importance ranking based on revised estimates of accident frequency and reduction factors.

10. Sheet 1 of Figure H.4.2 has been reduced so that it is illegible. A full-size legible copy should be provided.

11. Appendix L provides a detailed discussion of the disposition of the corium (90% is expected to go down the vent pipes) based on the revised reactor pedestal geometry illustrated in Figure L.3-2. However, this figure is inconsistent with other descriptions of the geometry (e.g., Figure 2.3-2) and provides inadequate information for an independent assessment of the corium disposition. Please provide detailed (as built) drawings of the vent pipes and their covers within and external to the reactor pedestal region. Include a description of whether the air ducts and manways in the reactor support wall will be blocked during operation.

12. Provide the estimate of the fraction of the molten corium which is expected to spread out of the pedestal area through the open manways and air ducts in the reactor support wall.

ATTACHMENT 2

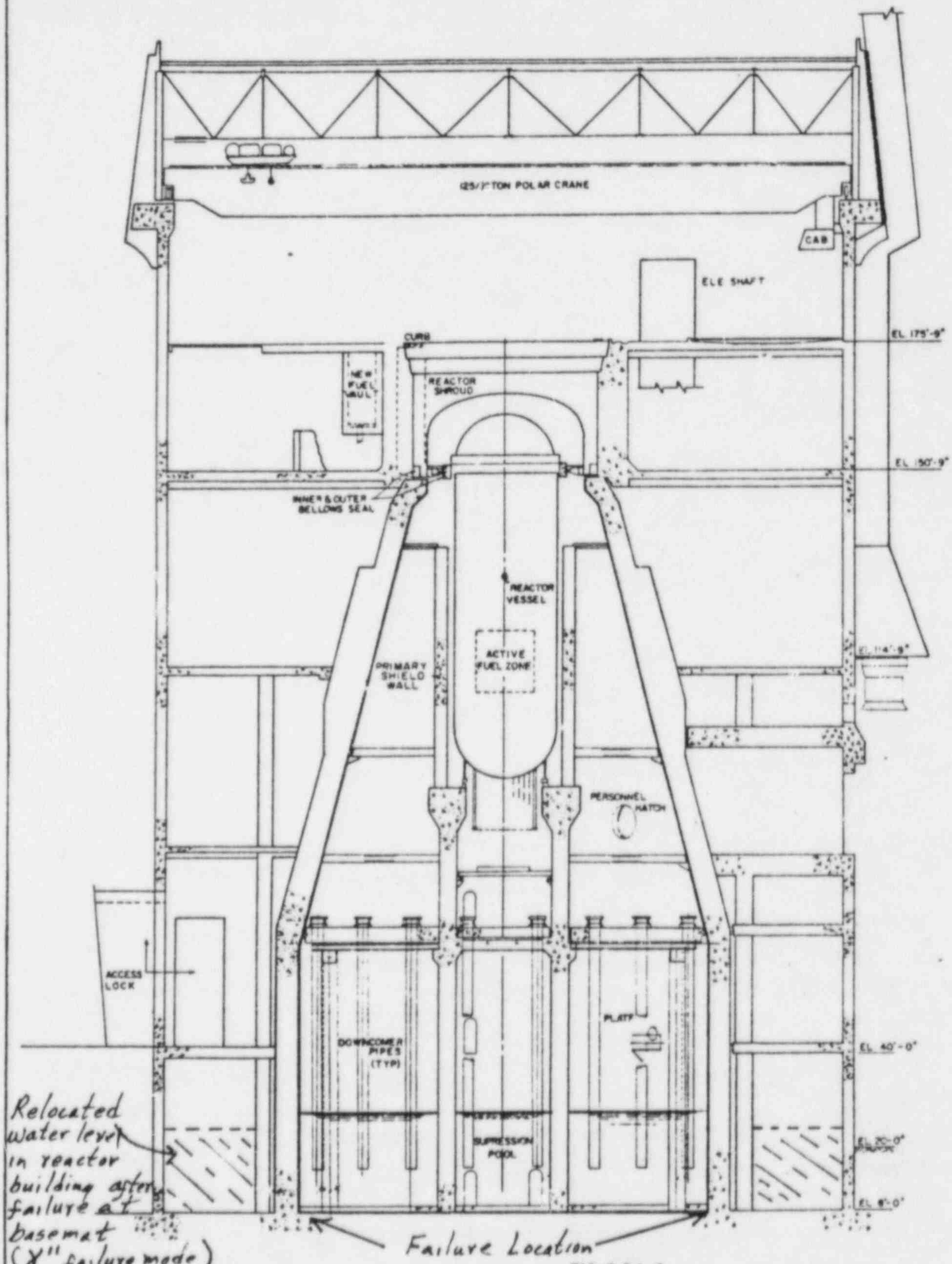

FIG. 6.2.1-4

PRIMARY AND SECONDARY CONTAINMENT

SHOREHAM NUCLEAR POWER STATION - UNIT I

FINAL SAFETY ANALYSIS REPORT

Note that 16% of pool is estimated to be vaporized during blowdown of primary containment after γ" failure

ATTACHMENT 3

Table 1 Comparison of Suppression Pool Decontamination Factors for Core Melt Accidents in Shoreham

Sequence	Decontamination Factors			
	Shoreham		BNL	
	SRV	Vents	SRV	Vents
TQUV (Class I-γ)	600	100	1000	75
ATWS (Class IV-γ')	600	100	50	22
ATWS (Class IV-γ'')	600	100	9	14