UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
LONG ISLAND LIGHTING COMPANY	)	Docket No. 50-322 (O.L.)
(Shoreham Nuclear Power Station, Unit 1)	)	

PREPARED DIRECT TESTIMONY OF

GREGORY C. MINOR

ON BEHALF OF SUFFOLK COUNTY

Regarding

SUFFOLK COUNTY CONTENTION 16

ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

June 29, 1982

SUMMARY OUTLINE OF SUFFOLK COUNTY CONTENTION 16

TESTIMONY - ATWS

Suffolk County contends that the present standby liquid control system ("SLCS") at Shoreham is inadequate to mitigate the range of ATWS consequences that might occur at the facility. The Shoreham SLCS is manually initiated and non-redundant. By automating the system and increasing its flow rate, the SLCS would mitigate a larger range of ATWS consequences and thus reduce the risk of core melt accidents at the facility.

Attachments

1. Shoreham FSAR Fig. 7.4.1-3
2. NUREG-0460, Vol. 4, p. 57, Table 2
3. NUREG-0460, Vol. 4, p. 11, Table 1
4. NUREG-0460, Vol. 4, p. E-2, Fig. E.1

PREPARED DIRECT TESTIMONY OF

GREGORY C. MINOR

SUFFOLK COUNTY CONTENTION 16-ATWS

I. INTRODUCTION

1. My name is Gregory C. Minor. I am employed by MHB Technical Associates of San Jose, California. My experience and qualifications have been submitted earlier in this hearing.

II. PURPOSE OF TESTIMONY

2. Suffolk County Contention 16 regarding ATWS was originally stated as follows:

> Suffolk County contends that LILCO and the NRC Staff have not adequately demonstrated that Shoreham meets the requirements of 10 C.F.R. 50, Appendix A, GDC 20, regarding correction of the anticipated transients without scram (ATWS) problem. 1/

3. The Board heard oral discussion of this subject at the March 9 and 10, 1982 Conference of Parties and admitted a reworded ATWS contention as follows:

> Although the anticipated transients without scram issue is generically before the Commission in a rulemaking proceeding, Suffolk County contends that LILCO and the NRC Staff have not adequately demonstrated that Shoreham meets the requirements of 10 C.F.R. Part 50, Appendix A, GDC 20, regarding correction of the ATWS problem in the interim period of several years pending completion and implementation of the result of the rule-making for Shoreham. This is because the interim measures to be taken at Shoreham, including operational procedures and operator training, will not compensate for the lack of

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1/ ASLB Board Memorandum and Order Confirming Rulings made at the Conference of Parties, March 15, 1982, p. 15.

> an automaticaly initiated and totally redundant standby liquid control system (SLCS) which meets the single failure criterion. 2/

This testimony is in response to the revised ATWS/Standby Liquid Control contention.

III. DESCRIPTION OF SHOREHAM STANDBY LIQUID CONTROL SYSTEM (SLCS)

4. Shoreham's SLCS provides for the injection of sodium pentaborate solution into the reactor core to provide reactivity control in the event of a failure to scram. The SLCS is manually activated by a single key-locked switch in the main control room. The system uses redundant pumps and valves but only one storage tank and a single piping run to the reactor from the explosive valves. There are also single pipe lines between the SLC storage tank and the pumps and between the pumps and the explosive valves. FSAR Figure 7.4.1-3, Attachment 1 hereto, illustrates the redundant and non-redundant aspects of the SLCS. Activation of the system causes the firing of the two explosive valves, and starts up one of the two 43 gallons-per-minute (gpm) pumps. 3/ 4/ The second SLCS pump provides redundancy, but only one pump may function at a time. 5/ The firing of the explosive valves clears

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2/ Ibid 1, p. 19.

3/ Shoreham FSAR, Figure 7.4.1-3.

4/ Pump rate is discussed in the Deposition of David J. Robare and Leonard Calone, April 22, 1982, pps. 19-21.

5/ Ibid 3, p. 4.2-84.

a flow path to the reactor pressure vessel. If the SLC is activated, the sodium pentaborate solution would be pumped along that path and into the reactor at a nozzle entering the lower plennum of the pressure vessel. When the SLCS is not in use, the sodium pentaborate solution is held in a 5,000-gallon storage tank,[6/][7/] where it is provided with heaters and stirring mechanisms to keep the sodium pentaborate in solution.

5. The following indicators provide information as to the status of the SLCS and its components:

a. Pump motor contactor (open/closed) - indicated by lights.

b. Pump discharge pressure. (flow is not indicated)

c. Explosive valves (open/closed) - indicated by lights.

d. Storage tank level.[3/]

6. The entire Shoreham SLCS (expect for the control switch and the final piping run to the pressure vessel) is located in one area of the reactor building outside the primary containment and without physical separation of redundant components.

6/ Ibid 3.

7/ Storage tank capacity is 5,000 gal. but normal levels are between 3500 and 4000 gal., as shown in Shoreham FSAR, Figure 4.2.3-12.

3/ Shoreham FSAR, p. 4.2-84.

IV. NEED FOR AUTOMATION OF SLC

7. When a transient calls for a scram of the Shoreham reactor, but a failure of the protection system/control rods scram function prevents shutdown, the reactor is generating higher power than the usual decay heat levels considered in ECCS analysis. The result of an ATWS, if not mitigated, could lead to a core degradation and/or containment breach. Thus, the time for an operator to decide to activate the SLCS system and the time for the system to function are important to the consequences of the ATWS event.

8. During an ATWS event the operator must first recognize the existence of an ATWS condition and that there is a need for ATWS mitigation. Next he must make the decision to exercise the SLCS manually. This involves the use of a key to activate the SLCS via the key-lock switch on the control room panel.

9. The operator's decision to activate the SLCS will have to be weighed against the chance that the SLC solution is not really needed, in which case an erroneous injection of the sodium pentaborate solution into the reactor will require possibly several months of down time to clean up and inspect the reactor.[9/] The potential conflict facing the operator between a prudent safety decision to protect the plant vs. possible operator error which will result in an extensive outage must be decided in a

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9/ NUREG-0460, Vol. 3, p. 28-29 discusses the financial impact of such an outage. See also Attachment 2.

brief period of time, perhaps only a few minutes.[10/] Normal design practice in the U.S. nuclear industry has been to automate safety functions which must be initiated in less than 10 minutes following a transient or accident (often referred to as the "10 minute rule").[11/]

10. The other variable related to effectiveness of the SLCS and the need for automation is the time required for the SLC system to perform its function of shutting down the reactor after the SLCS has been activated. This is a function of several factors, such as the pumping rate, point of injection in the vessel, mixing time, and the starting power level in the core.[12/] Generally, the longer the time for the injection to be effective, the greater the need for SLCS automation to ensure prompt initiation of the system.

11. For systems with a 43 gpm flow rate, the time for the SLCS to shut down the reactor will be greater than for systems with higher flow rates. Thus, for 43 gpm systems the need for automation would be greater than for a system with higher pumping capability.

12. The Staff analysis in NUREG-0460 shows that ATWS events for

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10/ NUREG-0460, Vol. 3, p. 26 discusses the benefit of a two-minute automatically initiating SLCS in avoiding high suppression pool temperatures. The Shoreham FSAR in Chapter 15 references NEDO 20626 (GE, Oct. 1974 and amendments thru July 1975), which discusses an automated SLCS which would activate in 40 seconds.

11/ Some European countries have adopted the practice of automating to eliminate the need for operator action for longer periods of time (e.g., Sweden uses a 30 minute rule).

12/ NUREG-0460 Vol. 4 pages 8, 9 mentions power oscillations particularly in BWR-4's, which may necessitate different injection rates. See also Vol. 3, p. 26.

a non-automated, 43 gpm system (Alternative 2A)[13/] more often lead to core vulnerability or core melt than for an automated 86 gpm system[14/] (Alternative 3A). Additionally, NUREG-0460 shows that an automated, higher flow rate (2-400 gpm) system (Alternative 4A) is capable of mitigating even a wider range of ATWS consequence.[15/]

13. Automation of SLCS is also consistent with GDC 20 which requires the protection system function to:

> . . . be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

The SLCS is considered by LILCO to be a "special safety system" and provides the function of a reactivity control system. Without automation, there is no assurance that the SLCS will be able to prevent exceeding fuel design limits for many ATWS accidents.

V. IMPROVEMENT IN SAFETY BY AUTOMATING SLC

14. One of the uses of a Probabilistic Risk Assessment (PRA) is to provide a means of comparing the value of proposed plant modifications in terms of relative reduction in probability of accidents

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13/ NUREG-0460 Vol. 4, pages 11 and 12. See Attachment 3.

14/ NUREG-0460 Vol. 3, page 26, discusses the need for a higher flow rate SLC.

15/ NUREG-0460 Vol. 4, pages E2, E5, E6 and E7. See Attachment 4.

or risks. The value of an improved and automated SLCS may be analyzed by comparing the PRAs on two similar BWR-4/Mark II plants: Limerick which has an automated higher flow rate SLCS, and Shoreham which has a manual 43 gpm SLCS. Both plants have the RPT, ARI and SDV improvements.[16/] The two PRAs show results for five categories of releases in increasing order of severity from Class I to Class V. The frequency of occurrence of core vulnerability or core melt accidents for the classes are listed in Table 1.

TABLE 1

COMPARISON OF FREQUENCY OF CORE VULNERABILITY OR CORE MELT FOR CLASSES OF RELEASES ON LIMERICK AND SHOREHAM REACTORS

CLASS	CHARACTERISTICS	SHOREHAM	LIMERICK
I	Fast core melt, containment fails after, P_{low}	2.7×10^{-5}	1.3×10^{-5}
II	Slow core melt, containment fails prior	1.1×10^{-5}	1.0×10^{-6}
III	Fast core melt, containment fails after, P_{high}	3.6×10^{-7}	1.0×10^{-6}
IV	Fast core melt, containment fails prior, P_{high}	6.1×10^{-6}	2.0×10^{-7}
V	Fast core melt, containment fails due to initiation of accident	2.0×10^{-8}	(Not Listed)
TOTAL		4.4×10^{-5}	1.5×10^{-5}

Sources of Data: Shoreham PRA by SAI (Preliminary Draft) p.4-3
Limerick PRA by SAI (Mar. 1981) p. 3-96
(values for classes scaled from histogram; total from Fig. 3.5.4).

All values are Mean Frequency/Reactor Year

16/ RPT is recirculation pump trip; ARI is alternative rod insert; SDV is scram discharge volume.

The major contributor to Class IV vulnerabilities at both Limerick and Shoreham is stated in the PRAs to be ATWS. (For Limerick, ATWS is also listed as a significant contributor to the lesser release category, Class III.) Limerick shows a substantially lower frequency (indeed, a factor of 30 less) of Class IV releases compared to Shoreham. The impact of Shoreham operating procedures and training apparently was not sufficient to overcome the greater calculated frequency of ATWS events resulting in core vulnerability due to the differences in the mitigating systems.

15. Another available comparison is to look at a "pie chart" distribution of core vulnerability events resulting from different contributing sequences. Since both PRAs have a similar chart, these are reproduced in Figure 1 for comparison. They also show the Wash-1400 values as a third comparison. Ignoring for now the exact values, the relative data are meaningful to show how the addition of ATWS mitigating techniques (including SLCS automation, larger SLC flow rate, and containment overpressure relief (COR)) have substantially reduced at Limerick the ATWS contribution to the calculated frequency of core vulnerability and core melt accidents. In addition, ATWS for Limerick is a smaller piece of a smaller pie. Improvements in the Shoreham SLC, including automation, would reduce both the ATWS contribution and the overall calculated frequency of core vulnerability and core melt.

FIGURE 1

COMPARISON OF ATWS CONTRIBUTION TO FREQUENCY TO CORE MELT ACCIDENTS IN TWO BWR-4 REACTORS; ONE WITH AUTOMATED SLC (LIMERICK) AND ONE WITH MANUAL SLC (SHOREHAM). */

WASH-1400

MEAN-3×10^{-5} /REACTOR YEAR

LIMERICK

MEAN-1.5×10^{-5} /REACTOR YEAR

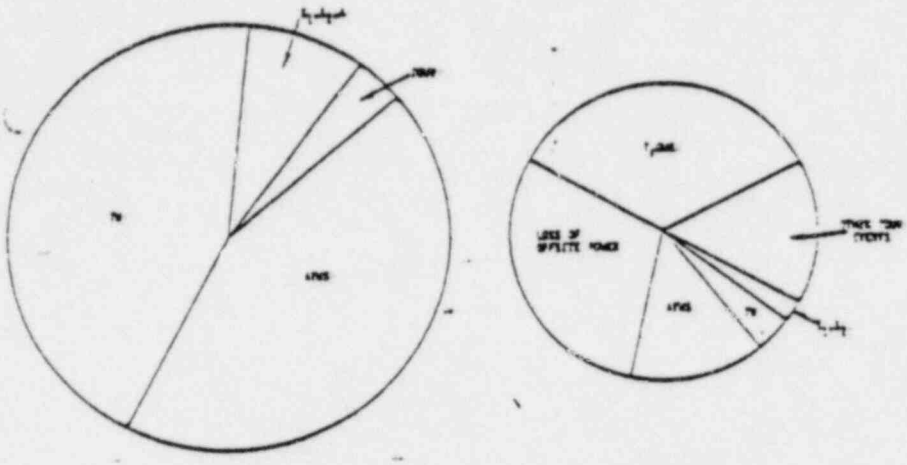


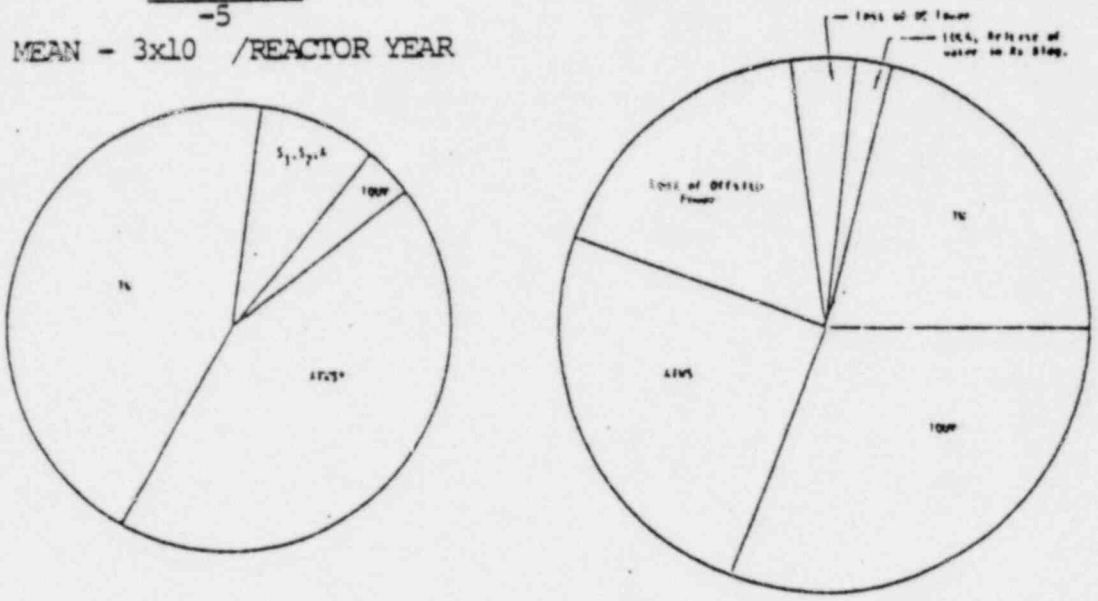
Figure 3.5.4 Comparison of the Contributing Accident Sequence to the Calculated Frequency of Core Melt from WASH-1400 and the Limerick Analysis (Area of "Pie Chart" is proportional to Mean Frequency)

SHOREHAM

MEAN = 4.5×10^{-5} /REACTOR YEAR

WASH-1400

MEAN - 3×10^{-5} /REACTOR YEAR



* Subsequent to WASH-1400, NRC evaluations of the potential contribution of ATWS to core melt (e.g., NUREG-0460) placed the frequency of ATWS is a BWR at nearly twenty times that evaluated in WASH-1400. If this were incorporated into the figure it would be the single dominant contributor to core melt, and would be significantly larger than the frequency of core melt calculated for Shoreham.

Figure 3.6.6 Comparison of the Contributing Accident Sequences to the Calculated Frequency of Core Melt from WASH-1400 and the Shoreham Analysis (Area of "Pie Chart" is proportional to Mean Frequency)

*/ Source of Data: Limerick PRA (Mar. 1981) p. 3-100
Shoreham PRA (Prelim. Draft 3/82) (p. 3-150)

VI. BENEFITS OF INSTALLING ATWS MITIGATION SYSTEMS NOW

16. The NRC Staff has estimated that the SLCS modifications in Alternative No. 3A of NUREG-0460 will provide a risk reduction for BWR's by a factor of 20.[17/] Shoreham has implemented some of the recommended ATWS prevention techniques (ARI and SDV) which has resulted in some improvement in the predicted frequency of ATWS events. Shoreham has also implemented RPT as a mitigation technique. However, despite these improvements, the Shoreham PRA estimated that ATWS events account for twenty-five percent of the Shoreham core melt probability.

17. The Limerick/Shoreham PRA comparison verifies that substantial safety improvements are available by acting now to improve ATWS mitigation at Shoreham. On the other hand, waiting until after the NRC's ATWS rulemaking could have a negative safety impact. NUREG-0460, Volume 4 compares the results of various ATWS sequences for BWR's using Alternatives No. 2A, 3A, and 4A.[18/] The 2A plant suffers a core melt in many ATWS sequences. In terms of ATWS mitigation systems, the Shoreham plant stands somewhere between Alternatives No. 2A and 3A. There is no assurance, based on a review of NUREG-0460 and the Shoreham PRA, that a 43 gpm, manually actuated SLCS can adequately mitigate the consequences of an ATWS event.

17/ NUREG-0460, Vol. 4, p. 13.

18/ See Attachment 2.

18. Table 2 of NUREG-0460, Volume 4, lists the NRC Staff's estimates of values and impacts for Alternatives No. 3A and 4A.[19/] Even for the worst case, which is for fully-built plants, the net value of Alternatives No. 3A and 4A modifications is favorable compared to its impact.

19. If the NRC elects to require an improved SLC comparable to Alternative No. 3A for BWR's, LILCO will already have complied, and will avoid the expense of a shutdown or an extended refueling outage for later implementation. It will also avoid worker exposure to radiation (consistent with ALARA) which would occur if the modifications were made after start-up of the reactor.[20/]

VII. CONCLUSION

To summarize this testimony;

1. A 43 gpm manual SLCS is incapable of mitigating some ATWS sequences.
2. The NRC Staff has recommended improved ATWS mitigating systems in the form of automatic higher volume SLCS.
3. An improved automatic, higher flow SLC has been designed and implemented on a comparable BWR-4/Mark II plant (Limerick).

19/ See Attachment 2.

20/ Ibid 15, p. 28.

4. The Limerick PRA shows substantially reduced ATWS caused releases and reduced frequency of core vulnerability compared to Shoreham.
5. Making SLCS changes now has safety, economic, and ALARA advantages for Shoreham.
6. LILCO has not demonstrated that Shoreham complies with GDC 20.

ATTACHMENT 1

SLC DIAGRAM FSAR FIGURE 7.4.1-3

ATTACHMENT 1

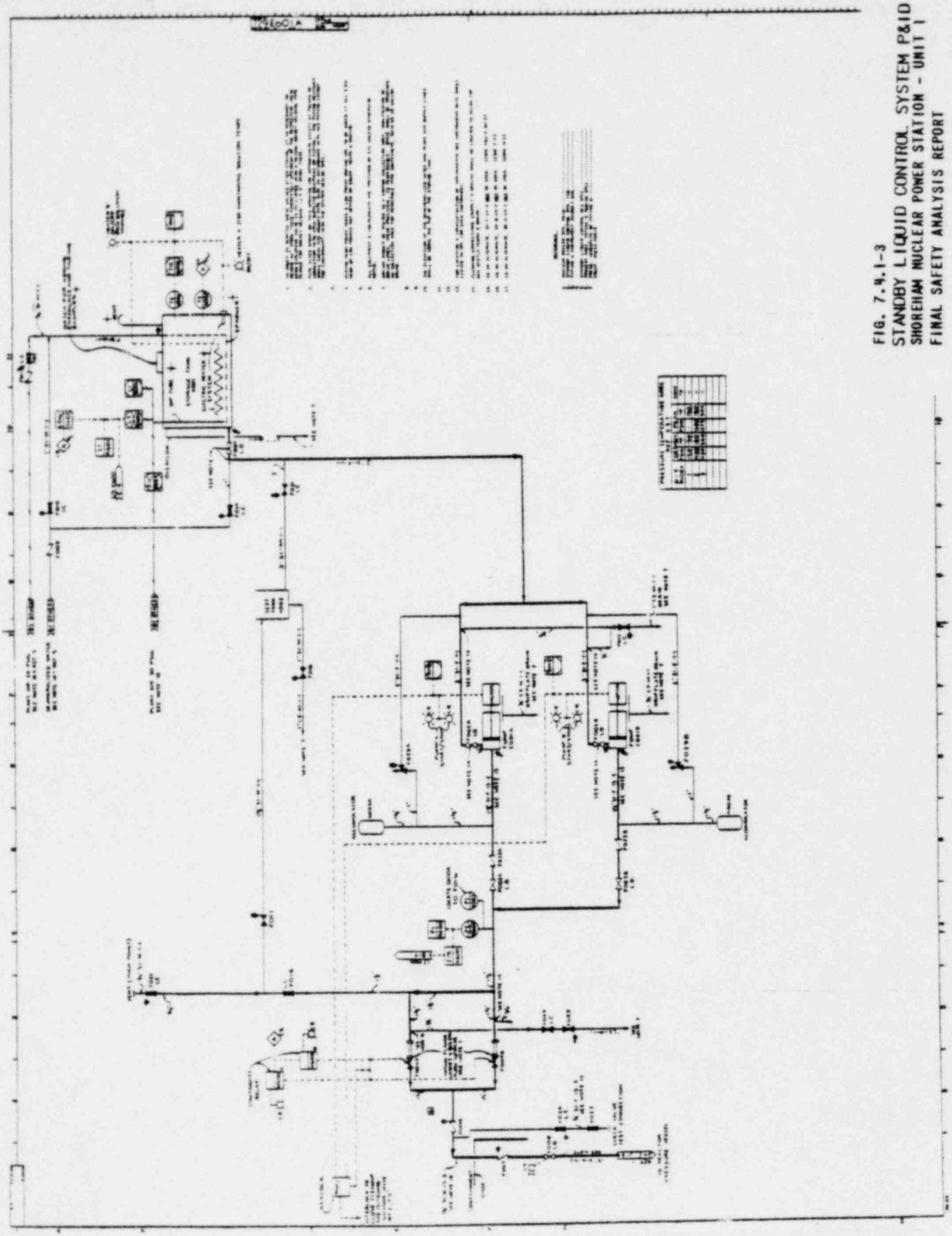


FIG. 7.4.1-3
STANDBY LIQUID CONTROL SYSTEM P&ID
SHOREHAM NUCLEAR POWER STATION - UNIT 1
FINAL SAFETY ANALYSIS REPORT

ATTACHMENT 2

"IMPACT AND VALUE OF ATWS MODIFICATIONS

FROM NUREG-0460, VOLUME 4"

TABLE 2. SUMMARY OF VALUE AND IMPACT ALTERNATIVE ATWS REQUIREMENTS, 1980 DOLLARS IN MILLIONS PER PLANT LIFETIME

Plant Designer	Operating Plants				Plants Under Construction (Construction Permit Issued)				Future Plants (Construction Permit Not Issued)	
	Alternative 3A		Alternative 4A		Alternative 3A		Alternative 4A		Alternative 4A	
	Impacts	Values	Impacts	Values	Impacts	Values	Impacts	Values	Impacts	Values
Babcock & Wilcox	2.8	2.6 - 7.4	4.4	6.0 - 17.0	2.0	2.6 - 6.8	2.9	6.2 - 12.8	2.5	5.4 - 11.7
General Electric	2.6	2.6 - 7.4	4.4	6.0 - 17.0	1.8	2.6 - 6.8	2.9	6.2 - 12.8	2.5	5.4 - 11.7
Westing-house	1.7	3.8 - 8.6	2.5	3.8 - 8.6	1.2	4.0 - 7.2	1.8	6.2 - 12.8	2.0	5.4 - 11.7
*General Electric	3.5	31.8 - 54.8	13.0	42.7 - 87.0	3.2	48.8 - 80.5	10.8	51.4 - 84.7	6.7	44.0 - 64.6

*No cost was included for cleanup and downtime resulting from inadvertent actuation of poison injection system (estimated $200,000 to $8,000,000 per plant lifetime).

ATTACHMENT 3

"ALTERNATIVE ATWS MODIFICATIONS

FROM NUREG-0460, VOLUME 4"

TABLE 1. COMPARISON OF REQUIREMENTS

Vendor	Alt. 2 (Vol. 3)	Alt. 2A	Alt. 3 (Vol. 3)	Alt. 3A	Alt. 4 (Vol. 3)	Alt. 4A
B&W	BUSS[1] AMSAC[2]	BUSS[1] AMSAC[2] Analysis[3]	BUSS[1] AMSAC[2] Analysis[4]	BUSS[1] AMSAC[2] Cont. Isol.[5] Inst.[6]	AMSAC[2] Add safety valves Analysis[4]	BUSS[1] AMSAC[2] Safety Valves Analysis[7] OPT[8] Cont. Isol.[9] Inst.[6]
CE	SPS[1] AMSAC[2]	SPS[1] AMSAC[2] Analysis[3]	SPS[1] AMSAC[2] Analysis[4]	SPS[1] AMSAC[2] Cont. Isol.[5] Inst.[6]	AMSAC[2] Add safety valves Analysis[4]	SPS AMSAC Safety Valves OPT[8] Analysis[7] Cont. Isol.[9] Inst.[6]
W	AMSAC[2]	AMSAC[2] MSS[1] Analysis[3]	AMSAC[2]	AMSAC[2] MSS[1] Cont. Isol.[5] Inst.[6]	AMSAC[2] Analysis[4]	AMSAC[2] MSS[1] Analysis[7] Cont. Isol.[9] Inst.[6]
GE	ARI[1] SD[11] RPT[12] Logic[14]	ARI[1] SD[11] RPT[12] Logic[14] Analysis[3]	ARI[1] RPT[12] Logic[14] Auto 86 gpm SLCS[15] SD[11] Analysis[4]	ARI[1] RPT[12] Logic[14] Auto 86 SLCS[15] SD[11] Cont. Isol.[5]	RPT[10] Automatic, high capacity liquid poison injection Analysis[4]	ARI[1] RPT[13] Auto Hi-Cap Poison Logic[14] SD[11] Analysis[7] OPT[8] Cont. Isol.[5]

Footnotes to Table 1

[1]A system that is diverse and independent from RPS, meeting IEEE-279 and acting as backup to the electrical portion of the current scram system.

[2]ATWS mitigating system actuation circuitry satisfying criteria in Appendix C, Volume 3.

[3]Analysis of Alt. 2A plants to decide if mitigation capability exists or is necessary in overall safety context.

[4]Analysis remains to be performed and reviewed to confirm expected mitigation capability as described in Sections 2.2 and 2.3 of Volume 3.

[5]Provisions to close containment isolation valves quickly if fuel failure should occur.

[6]Providing instruments necessary for shutdown that can withstand the ATWS peak pressure.

[7]Analysis of Alt. 4A plants to verify adequacy of plant modifications.

[8]Optimization study for Alt. 4A plants where full implementation is not practicable (Alt. 3 1/2).

[9]A system satisfying the criteria in Appendix C, Volume 3, that isolates containment quickly in the event of ATWS fuel failure.

[10]Recirculation pump trip satisfying criteria in Appendix C, Volume 3.

[11]Modification of scram discharge volume.

[12]The approved Monticello design is an acceptable RPT design for all BWR 4 plants. The approved Zimmer design is an acceptable RPT design for all BWR 5 and 6 plants. There may be other acceptable designs which must be treated on a plant specific basis.

[13]As in footnote 10, except that RPT installed before July 1, 1981, may be in accordance with footnote 12.

[14]Changes in logic to reduce vessel isolation events and permit feedwater runback.

[15]Modified SLCS piping to assure delivery of 86 gpm of poison and automatic actuation circuitry satisfying parts A through H of Appendix C, Volume 3, with reliability equivalent to the mechanical portion of the SLCS.

ATTACHMENT 4

"ATWS EVENT SEQUENCES AND CONSEQUENCES

FROM NUREG-0460, VOLUME 4"

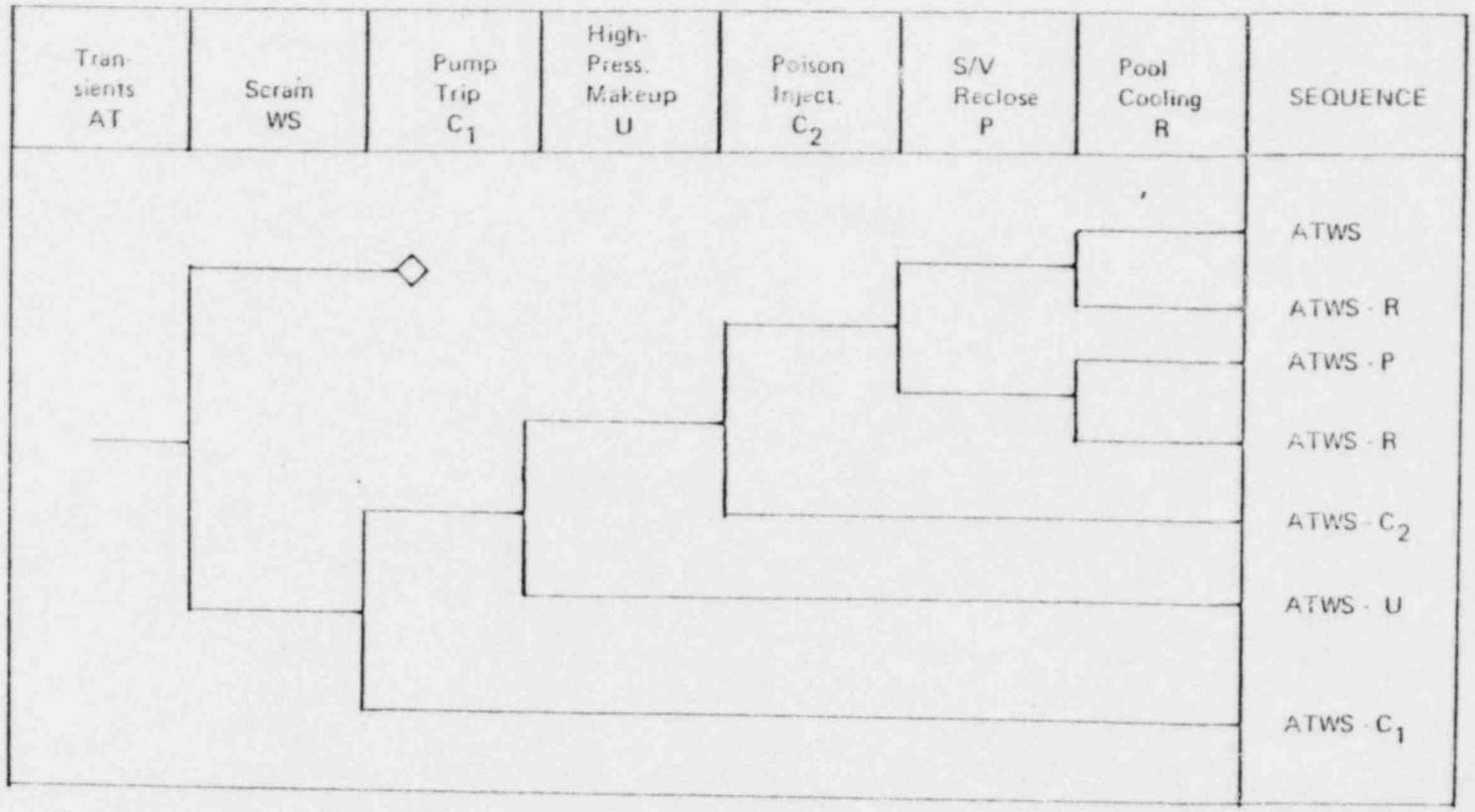


Figure E.1. Simplified BWR event tree.

TABLE E.1 ATWS FREQUENCY AND CONSEQUENCE IN BWRs

Sequence	Alt. 2A Solution[1]		Alt. 3A Solution[1]		Alternative 4A Solution[1]	
	Frequency	Consequence	Frequency	Consequence	Frequency	Consequence
ATWS	$\sim 9 \times 10^{-5}$ (2)	Likely core melt (3)	$\sim 5 \times 10^{-5}$ (4)	Likely OK (5)	$\sim 5 \times 10^{-5}$	Acceptable
ATWS-C_1 (6)	$< 10^{-6}$ (7)	Core melt	$< 10^{-6}$	Core melt	$< 10^{-6}$	Core melt
ATWS - U	------ (8)		$\sim 5 \times 10^{-6}$ (9)	Likely core melt (10)	$\sim 5 \times 10^{-6}$	Acceptable (11)
ATWS-C_2	------ (8)		$\sim 3 \times 10^{-6}$ (12)	Likely core melt	$< 10^{-6}$ (13)	Likely core melt
ATWS - P	------ (8)		$\sim 5 \times 10^{-6}$	High containment temperature	$\sim 5 \times 10^{-6}$	Acceptable
ATWS - R	------ (8)		$\sim 5 \times 10^{-6}$ (14)	High containment temperature	$\sim 5 \times 10^{-6}$	Acceptable

FOOTNOTES FOR TABLE E.1

[1]See Chapter of this report.

[2]Current scram systems are highly reliable and thus difficult to significantly improve. This analysis assumes, conservatively, we believe, that modifications in the electrical portions would reduce unreliability for scram system from 3×10^{-5} to 1.5×10^{-5}.

[3]Inability to shut down reactor; containment failure; makeup water depleted; core melt may occur.

[4]Changes in setpoints and feedwater runback (in conjunction with at least 86 GPM boron system) may reduce isolation events to 4/RY (estimate).

[5]Core wide oscillations, severity plant dependent, safety impact unknown, pool temperature may exceed limit.

[6]ATWS followed by failure of recirculation pumps to trip.

[7]Failure of one to two safety/relief valves to reclose (estimate derived from the RSS) following an ATWS.

[8]These sequences are included in the core-melt sequence above.

[9]Failure to provide high-pressure makeup coolant (10^{-1} from RSS). HPCS (BWR 5/6 plants) may be more reliable (factor of 2 to 5).

[10]Inability to keep core covered.

[11]Alternate 4A increased SLCS capability (flow rate) is expected to keep the core covered in the event HPCI(s) fails.

[12]Assumed unreliability of delivering 86 or more GPM poison in 2 minutes (not operator (manual) actuated). If conservative values for the initial pool temperature and the ΔT between local and bulk temperature are used, the calculated peak local pool temperature may be as high as 240°F with 86 GPM System.

[13]The unreliability of the Alternative 4A SLCS design is required to be much lower than 10^{-2}.

[14]Operator action required (10 to 15 minutes) to begin pool cooling. Further delay of several minutes could result in significant increase in containment temperature. Probability of failure of this action is assumed to be 0.1 (similar value is suggested in the RSS).

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NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of LONG ISLAND LIGHTING COMPANY (Shoreham Nuclear Power Station, Unit 1)	Docket No. 50-322 (OL)

CERTIFICATE OF SERVICE

I hereby certify that copies of the Suffolk County Testimony on Contentions 12-15 and on Contention 16 were served this 29th day of June, 1982, to the following by U.S. Mail, first class, except as otherwise noted.

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(**) By hand 6/30/82
(#) By Federal Express 6/29/82