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August 2, 1989

Samuel J. Chilk, Secretary
United States Nuclear Regulatory
Commission
Washington, D.C. 20555

In the Matter of
Philadelphia Electric Company
(Limerick Generating Station, Unit 2)
Docket No. 50-353-OL-2
(Severe Accident Mitigation Design Alternatives)

Dear Mr. Chilk:

Enclosed for filing are "Response by Licensee Philadelphia Electric Company to Commission's Request for Comments by Memorandum and Order dated July 26, 1989" and "Affidavit of Corbin A. McNeill, Jr."

The original, executed Affidavit of Corbin A. McNeill, Jr. will be substituted for the copy attached hereto when received.

Sincerely,

Robert M. Rader

Robert M. Rader
Counsel for Licensee

RMR:sdd
Enclosures

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June 23, 1989

Docket Nos. 50-352
50-353
License Nos. NPF-39
NPF-83

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Limerick Generating Station, Units 1 and 2
Response to Request for Additional Information
Regarding Consideration of Severe Accident
Mitigation Design Alternatives

Gentlemen:

NRC letter dated May 23, 1989, requested Philadelphia Electric Company (PECo) to provide additional information concerning severe accident mitigation design alternatives (SAMDAs) for the Limerick Generating Station (LGS). The issue of SAMDAs is being litigated before an Atomic Safety and Licensing Board as a result of the decision of the United States Court of Appeals for the Third Circuit remanding this matter to the NRC for further consideration. The additional information was requested in order to allow preparation of an NRC staff position with respect to this issue. The specific NRC questions and our responses are provided in the attachment to this letter.

With respect to the information provided in the attachment, it should be recognized the importance of utilizing the most up-to-date information as to plant design and analysis methods when modeling the facility and the phenomenology associated with severe accidents when examining SAMDAs and the question of whether they are cost-beneficial. If, for example, the base case off-site risk from severe accidents is over-estimated, the benefits of any mitigation design alternative which would reduce that risk would likely also be over-estimated. Similarly, if the most up-to-date information concerning the dominant accident sequences and associated radioactivity releases were not utilized, the mitigation measures

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being examined might appear to be cost-beneficial, but in fact would not be since the mitigation design alternatives would not be based on potential actual sequences. The evaluation of SAMDAs conducted as part of preparation of the attached responses should be considerd as a screening process only. Should any SAMDA appear to be close to cost-beneficial as a result of this initial screening, this mitigation design alternative would be required to be optimized so as to maximize its benefit and, at the same time, minimize its cost. Moreover, a detailed examination of the associated dominant accident sequences being mitigated and phenomenology must be conducted to validate the result.

Please note also that there is a significant scope and regulatory impact uncertainty factor associated with the design alternatives discussed in the attachment, particularly given the short response time. There is little, and in some cases, no actual design, licensing, or installation experience with most of these design alternatives. Should detailed design, licensing, and ultimately, construction efforts proceed, additional complexities and problems would most likely arise that would further increase the final installed costs. Therefore, we consider that the likelihood of the estimated costs given in the attachment being overstated is extremely small.

If you should have any question, or require additional information, please contact us.

Very truly yours,

G. A. Hunger, Jr.

G. A. Bunger, Jr.
Director
Licensing Section
Nuclear Support Division

cc: W. T. Russell, Administrator, Region I, USNRC
T. J. Kenny, USNRC Senior Resident Inspector, LGS

QUESTION 1

On the basis of PRA results to-date, identify those accident sequences that are expected to dominate the overall mean frequency projected for severe core damage and for the significant off-site risks (i.e., projected risk of early fatalities and person-rem). It is suggested that those sequences that collectively contribute 90% to the overall mean frequency for severe core damage be identified as dominant and each described. For these dominant sequences, present the projected mean value for each, considering that three categories (i.e., internal initiations, fire initiations and earthquake initiations) will likely contribute to the overall results.

RESPONSE

The current estimate of core damage frequency (CDF) for Limerick Generating Station Unit 1 (LGS-1) is given in Table 1-1. The sequences that dominate the CDF are identified in Table 1-2. The sequences expected to dominate the offsite risk (population dose and early fatalities) are identified in Tables 1-3 and 1-4, respectively. All values are point estimates except seismic which are the means of calculated distributions.

Subsequent to the initial development of the LGS Probabilistic Risk Assessment (Reference 1), in response to the Commission's May 6, 1980 letter, and the Severe Accident Risk Assessment (Reference 2), developed in accordance with the requirements of the National Environmental Policy Act, Philadelphia Electric Company's (PECo) PRA activities have concentrated on the updating and use of the internal initiator portion of the Level 1 PRA in accordance with the Commission's June 7, 1984 letter and PECo's July 23, 1984 response.

The core damage frequencies for the internally-initiated sequences given herein are based on the November 1988 update of the LGS-PRA modified to include a Limerick turbine trip frequency of 2.55 scrams/year justified by actual Limerick operating experience (first two operating cycles). The frequency of other initiators (other transients and LOCAs) remains the same. The current total transient frequency utilized is 6.7/year. This is conservative and is expected to go down further as additional site-specific data are accumulated.

In order to provide a reasonable basis for evaluating mitigating designs, the externally-initiated sequences have been updated, to the extent possible in the time available, to account for major new information as described in the next three paragraphs.

The fire CDF has been updated to reflect the current plant fire protection design (Rev. 11, of the LGS Fire Protection Evaluation Report - Reference 3), the latest plant logic models of the November 1988 update of the PRA, and the initiator frequency and suppression probability from the Sandia Fire Risk Scoping Study (Reference 4). Even after this updating, the results remain conservative. Areas of conservatism include: the modeling and assumptions on the extent of damage given failure to suppress a fire i.e., it is assumed that all unprotected shutdown methods in a zone fail if any fire in the zone is not suppressed in 10 minutes; the modeling of fire suppression, mainly based on manual detection and suppression data (Reference 4); and in the determination of initiator frequency, which took no credit for cables at LGS upgraded in accordance with IEEE 383.

The seismic CDF has been updated to include revised fragilities based on actual LGS equipment seismic qualification data for a number of components (electrical equipment, SLC test tank, N_2 accumulators and RHR heat exchangers) versus the generic or surrogate plant data used originally in SARA where plant specific data were not then available, a more recent assessment of ceramic insulator fragility and analysis of recoverable electrical system failures (i.e., circuit breaker trips).

The flooding CDF has been revised to reflect the results of the detailed flooding protection analyses recently completed, the updated logic models of the November 1988 PRA update and the occurrence of spurious fire suppression initiation summarized in the Sandia Fire Risk Scoping Study (Reference 4).

The relative risk rankings of sequences given in Tables 1-3 and 1-4 were arrived at considering the accident class, as defined in SARA and given in Table 1-5, and the associated conditional risk for that class as calculated in SARA.

References For Question 1 Response

1. "Probabilistic Risk Assessment, Limerick Generating Station", Philadelphia Electric Company, September 1982.

2. "Severe Accident Risk Assessment, Limerick Generating Station", Philadelphia Electric Company, April 1983.

3. "Fire Protection Evaluation Report, Limerick Generating Station Units 1 and 2", Philadelphia Electric Company, Rev. 11, February 1989.

4. Lambright, J. A., et al., "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues", Sandia National Laboratories, NUREG/CR-5088, January 1989.

TABLE 1-1

CURRENT ESTIMATED
CORE DAMAGE FREQUENCY
(Per Reactor Year)

Internal Initiators		5.9E-06
Transients	(2.1E-06)	
Loss of Offsite Power	(2.3E-06)	
ATWS	(1.2E-06)	
LOCA	(2.7E-07)	
Seismic		3.4E-06
Internal Fires		4.2E-06
Others (Internal Floods and Other Special Initiators)		0.2E-06
Total Estimated CDF		1.[illegible]7E-05

TABLE 1-2

DOMINANT CORE DAMAGE SEQUENCES
(See Notes)

1. 1.90E-006 13.9% F44QUV 1
Fire in Fire Zone 44 (F44) with core damage resulting from combination of fire-induced and random failures leading to failure of high pressure (QU) and low pressure injection (V). The frequency of this sequence is conservative.

2. 1.80E-006 13.2% TSESUX 1
Seismically-induced loss (TSES) of offsite power followed by seismic and random failures of high pressure injection (U) and depressurization (X).

3. 8.60E-007 6.3% TSRB 1S
Seismic (TS) failure of reactor building (RB) resulting in failure of all injection.

4. 8.20E-007 6.0% F2QUV 1
Fire in Zone 2 (F2) with core damage resulting from the combination of fire-induced and random failures leading to failure of high pressure (QU) and low pressure injection (V). The frequency of this sequence is conservative.

5. 7.30E-007 5.3% TE5OSP2DG2RmC 1
Loss of offsite power followed by failure of all onsite power (TE5) and failura to recover offsite (OSP2) or onsite (DG2) power in 2 hours and failure to initiate alternate room cooling (RmC) in 2 hours.

6. 6.70E-007 4.9% TCVQUV 1
Loss of condenser vacuum (TCV) followed by failure of high pressure (QU) and low pressure injection (V).

7. 5.10E-007 3.7% F45QUV 1
Fire in Fire Zone 45 (F45) with core damage resulting from combination of fire-induced and random failures leading to failure of high pressure (QU) and low pressure injection (V). The frequency of this sequence is conservative.

8. 4.90E-007 3.6% TE5OSP2DG2OSP5DG5OSP10DG10 1
Loss of offsite power followed by failure of all onsite power (TE5) and failure to recover either in 10 hours.

9. 4.80E-007 3.5% TSRPV 3/S
Seismically-induced failure of the reactor pressure vessel supports (RPV).

10. 3.80E-007 2.8% TEBCC 1
Loss of offsite power (TE) and common cause failure of all batteries (BCC).

TABLE 1-2 Continued

DOMINANT CORE DAMAGE SEQUENCES
(see notes)

11. 3.30E-007 2.4% TCVQUX 1
Loss of condenser vacuum (TCV) followed by loss of high pressure injection (QU) and failure to depressurize the reactor (X).

12. 3.20E-007 2.3% F47QUV 1
Fire in Fire Zone 47 (F47) with core damage resulting from combination of fire-induced and random failures leading to loss of high pressure (QU) and low pressure injection (V). The frequency of this initiator is conservative.

13. 3.10E-007 2.3% TMQUV 1
Isolation transient (TM) followed by loss of high pressure (QU) and low pressure injection (V).

14. 2.00E-007 1.5% TCP2LHU' 4
Loss of condenser vacuum ATWS (TCP2), SBLC works, operator successfully lowers level (LH) but fails to control low pressure injection after depressurization (U').

15. 1.80E-007 1.3% TTQUV 1
Turbine trip (TT) event followed by failure of high pressure (QU) and low pressure injection (V).

16. 1.80E-007 1.3% F2QWFWECC 2
Fire in Fire Zone 2 (F2) followed by fire-induced and random failure of all heat removal (WFW). Containment vented successfully but injection fails (ECC).
The frequency of this initiator is conservative.

17. 1.70E-007 1.2% TE1UHURX 1
Loss of offsite power (TE1) followed by failure of HPCI (UH), RCIC (UR), and depressurization (X).

18. 1.60E-007 1.2% TSESCMC2 3/4
Seismically-induced loss of offsite power (TSES) followed by either random or seismic failure to insert control rods (CM) and failure of SBLC (C2).

19. 1.50E-007 1.1% TMQUX 1
Isolation transient (TM) followed by loss of high pressure injection (QU) and failure to depressurize the reactor (X).

TABLE 1-2 Continued

DOMINANT CORE DAMAGE SEQUENCES
(see notes)

20. 1.20E-007 0.9% TMP2LHU' 4
Isolation transient ATWS (TMP2), SBLC works, operator successfully lowers level (LH), but fails to control low pressure injection after depressurization (U').

21 1.20E-007 0.9% TMSQUV 1
Manual shutdown (TMS) followed by failure of high pressure (QV) and low pressure injection (V).

22. 1.20E-007 0.9% TSRBCM 1S
Seismic (TS) failure of Reactor Building (RB) results in failure of all injection and failure to scram (CM).

23. 1.20E-007 0.9% TTPPU' 4
Turbine trip ATWS (TTP) with a stuck open relief valve (P) followed by failure of operator to control low pressure injection after depressurization (U').

24. 1.00E-007 0.7% VR1 3/S
Random reactor vessel failure.

The above sequences add up to approximately 82% of the Total CDF. An additional 18 sequences bring the total to 90%. Each of these additional sequences contribute less than 1% and do not add any additional new functional failures not included in the top 24 sequences.

The information provided for each sequence is: its rank by CDF, the annual sequence frequency, the percent contribution to the total, the failure event making up the sequence and the accident class. The accident classes are as defined in SARA with Arabic numerals replacing Roman numerals. See Table 1-5.

TABLE 1-3

DOMINANT POPULATION DOSE SEQUENCES

Rank	Sequence	Accident Class	% Contribution to Total Population Dose
1	F44QUV	1	10.6
2	TSRB	1S	10.1
3	TSESUX	1	10.1
4	TSRPV	3/S	8.2
5	F2QUV	1	4.6
6	TE5OSP2DG2R_mC	1	4.1
7	TCVQUV	1	3.7
8	TCP2LHU'	4	3.3
9	F45QUV	1	2.8
10	TE5OSP2DG2OSP5DG5OSP10DG10	1	2.7
11	TEBCC	1	2.1
12	TMP2LHU'	4	1.9
13	TTPPU'	4	1.9
14	TCVQUX	1	1.8
15	F47QUV	1	1.8
16	TMQUV	1	1.7
17	TSRBCM	1S	1.4
18	TCP2U'	4	1.1
19	TSESCMC2	3/4	1.1
20	TTQUV	1	1.0
21	F2QWFWECC	2	1.0

These sequences contribute about 80% of the estimated population dose. The next 28 sequences would bring the total to approximately 90%. Each of these would add less than 1% of the population dose. The only additional functional failures occurring in these additional sequences are random reactor vessel failure and failure of pressure suppression following a large LOCA.

The sequence definitions are given in Table 1-2 except for the following:

TCP2U' - Loss of Condenser vacuum ATWS (TCP2), SLBC works and operator fails to control low pressure injection after depressurization (U').

TABLE 1-4

DOMINANT EARLY FATALITY SEQUENCES

Risk Rank	Sequence	Accident Class	% Contribution to Total Early Fatality Risk
1	TSRPV	3/S	49
2	TCP2LHU'	4	9
3	TMP2LHU'	4	6
4	TTPPU'	4	5
5	TCP2U'	4	3
6	TTPPLHU'	4	3

These sequences contribute about 75% of the total early fatality risk. The next 12 sequences would bring the total to approximately 90%. Each sequence would add 2% or less to the total.

The only additional functional failures occurring in these additional sequences are random reactor vessel failure, seismically induced failure to scram and failure of SBLC, failure of HPCI following a turbine trip ATWS, failure to restore feedwater following HPCI failure for a turbine trip ATWS, failure to bypass level 1 MSIV closure before lowering level after a turbine trip ATWS, and failure to inhibit ADS after an ATWS.

The sequences are defined in Table 1-2, except for the following:

TTPPLHU' - Turbine Trip ATWS (TTP) with stuck open relief valve (P), operator successfully lowers level (LH) but fails to control low pressrue injection after depressurization (U').

Revised July 1989

TABLE 1-5

ACCIDENT CLASSES

CLASS	DESCRIPTION	EXAMPLE
1 (or I)	Transients or LOCA's involving loss of coolant makeup to the core. Core melts in an intact containment.	TCVQUV
2 (or II)	Transient or LOCA's involving loss of long term heat removal. Long-term core melts in a failed or open containment.	F2QWFWECC
3 (or III)	Transients with failure to scram with failure of all injection. Rapid core melt in an intact containment.	TCP2LHV
4 (or IV)	Transient with failure to scram and failure to shutdown. Rapid core melt in a failed or open containment.	TCP2LHU'
S	Core melt due to reactor pressure vessel failure with early containment failure.	VR1
1S	Earthquake initiated transient with failure of all injection. Core melts into an open containment	TSRB

QUESTION 2

For the internal and fire initiated sequences, assess the potential severe accident design mitigation alternative (s), that (if put in place or installed) have a reasonable chance of reducing the projected severe core damage frequency and off-site risks and (1) which may result in a substantial increase in the overall protection of the public health and safety, and (2) which are justified by the attendant direct and indirect costs associated with putting the alternative into place. As noted, this assessment should be limited only to those internal and fire initiated sequences (exclude those sequences initiated by earthquakes over any portion of the earthquake hazard spectrum). Regarding this exclusion, it is the staff's opinion that the incremental severe accident risks due to the nuclear plant relative to all other risks that could potentially be presented by severe earthquakes (up to those large enough to cause the severe core damage accident) would be negligibly small, (i.e., so small that the projected risk reduction benefits attendant to seismic related plant improvements would represent a very remote and speculative projection given the uncertain, competing risks presented to the public off-site from the severe earthquake itself).

RESPONSE

Scope

For the purpose of this evaluation, the range of Severe Accident Mitigating Design Alternatives (SAMDAs) identified in the basis of the LEA contention as defined by the Atomic Safety & Licensing Appeal Board (ALAB-819, dated October 22, 1985) were initially considered. The SAMDAs identified by R&D Associates (Reference 1), were then considered. The seven SAMDAs listed in Table 2-1 were then further evaluated as representative of the classes of SAMDAs applicable to Limerick. Each is discussed below after a general discussion of the approach to the evaluation.

Evaluation Approach

The design for each of the SAMDAs developed in Reference 1 was reviewed and a revised design basis developed by adding or eliminating features which were considered either needed or not needed to achieve the desired mitigation objectives. The basic design requirements were then translated into design concepts for cost estimating purposes.

The cost estimates include both initial and annual costs as appropriate in such categories as engineering, materials, construction, replacement power, regulatory, health physics support, training, maintenance, and QA. It was assumed in estimating the costs and benefits that:

- New equipment is non-safety related unless failure of the equipment could have an adverse impact on other safety-related equipment.

- Structures, systems and components added by the modification and in the reactor enclosure and control structure will meet LGS Seismic Category IIA criteria. As described in the Limerick FSAR, those components listed as Seismic Category IIA are either designed to Seismic Category I criteria or are reviewed to identify those whose failure could result in loss of required function of Seismic Category I structures, equipment, or systems required after an SSE. Components identified by this review are considered safety-impacted items and are either analytically checked to confirm their integrity against collapse when subjected to seismic loading from the SSE or are separated from Seismic Category I equipment by a barrier. Structures, systems, and components not located in safety-related area, whose sole function is mitigation of severe accidents will be designed and constructed to Seismic Category II (non-seismic Category I) criteria. Such structures, systems and components will comply with high-quality industrial codes and standards, e.g., the Uniform Building Code.

- The designs should not compromise or invalidate the existing design basis of the plant.

Costs were estimated for two units and then divided by 2 to obtain a per unit cost. The present worth of the annual costs was calculated using a 40 year plant life and a discount rate of 10%. All costs are in 1989 dollars.

It should be noted that there is a significant scope and regulatory impact uncertainty factor in the design concepts, which were developed over a short period of time for this report. There is little or, in some cases, no actual design, licensing or installation experience with these concepts. Should detailed design, licensing and construction proceed, it is therefore likely that additional complexities and problems would arise to further increase the final installed costs. In any case, it is very unlikely that the estimated costs provided herein have been significantly overestimated.

The benefit associated with each SAMDA was quantitatively assessed in terms of the estimated man-rems/per year averted as a result of its installation. The basis of this assessment were the internal, fire, and flood core damage frequencies summarized in the response to Question 1 and the containment analysis, source term analysis and consequence analysis of the Limerick Severe Accident Risk Assessment (SARA). The conditional population dose out to 50 miles, given an accident of the various internal, fire and flood accident classes, is given in Table 2-2 along with the total accident class frequency. The classes are defined in Table 1-5. The source terms and resulting population dose are believed to be conservative as they are based on source term technology of the 1981-1983 time frame. An adjustment was made to the SARA results to account for the benefit of the existing plants' capability to spray or inject water into the drywell after a core melt. The original PRA/SARA did not include this. The averted dose was then assessed by examining the effectiveness of each SAMDA on each accident class.

The benefit of the estimated reduction in population dose was estimated using $1000 per man-rem (References 2, 3 and 4) and the present worth at 10% for 40 years. The $1000 figure is used as a surrogate to represent all the offsite effects. Details of the assessment of each SAMDA are provided on pages 2-8ff.

Summary of Cost Benefit Results

The costs and benefits of the mitigation systems are summarized in Table 2-3. The table provides the following:

Benefit: The estimated risk reduction in dollars per year calculated from the estimated man-rem per year averted by the mitigation device times $1000 per man-rem.

Total Benefit: The present worth in dollars of the yearly benefit assuming a 40 year plant life and a 10% discount rate.

Total Cost: The total cost of the mitigation device including construction costs and the present worth of annual operating costs over a 40 year plant life.

Benefit/Cost Ratio: The ratio of the total benefits to total costs. A value greater than 1.0 would indicate a cost beneficial mitigation device.

Cost/Man-rem Averted: The cost per man-rem averted. A cost less than $1000/man-rem would indicate a cost beneficial mitigation system.

The results presented in Table 2-3 show that none of the mitigation systems examined are cost beneficial. In fact, the results indicate that no mitigation system is within an order of magnitude (factor of 10) of being cost beneficial.

References for Question 2 Response

1. Dooley, J.L., et,al., "Mitigation Systems for Mark II Reactors", RDA-TR-127303-001 (Preliminary), May 1984.

2. Heaberlin, S.W., et al., "A Handbook for Value Impact Assessment", NUREG/CR-3568, December 1983.

3. Kastenburg, W.E., et al., "Value/Impact Analysis for Evaluating Alternative Mitigating Systems", NUREG/CR-4243, January, 1988.

4. Stello, V., Jr., to the NRC Commissioners, "Mark I Containment Performance Improvement Program", SECY-89-017, January 23, 1989.

TABLE 2-1

SEVERE ACCIDENT MITIGATING DESIGN ALTERNATIVES EVALUATED

- POOL HEAT REMOVAL SYSTEM
  A separate independent dedicated system for transferring heat from the suppression pool to the spray pond utilizing a diesel driven 3,200 gpm pump and heat exchanger without dependence on the Station's present AC electrical power or other systems. The diesel is cooled with water tapped off the spray pond suction line.

- DRYWELL SPRAY
  A new dedicated system for heat and fission product removal using the Pool Heat Removal System described above to inject water into the drywell.

- CORE DEBRIS CONTROL ("CORE CATCHERS")
  Two techniques, either a basemat rubble bed, or using a dry crucible approach, to contain the debris in a known stable condition in the containment.

- ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) VENT
  A large wetwell vent line to an elevated release point to remove heat added to the pool in an ATWS event.

- FILTERED VENT
  Drywell and Wetwell vents to a large filter (two types - gravel or enhanced water pool) to remove heat and fission products.

- LARGE H_2 RECOMBINER
  Independently powered recombiners to remove H_2 from the containment in the long-term after a severe accident.

- LARGE CONTAINMENT VACUUM BREAKER
  To restore containment pressure to atmospheric level through 20" valves in certain severe accident cases where a vacuum has been produced.

TABLE 2-2

LIMERICK RISK (POPULATION DOSE) PROFILE BY CLASS

CLASS	FREQUENCY (per year)	CONDITIONAL 50 MILE POPULATION DOSE (Man-Rem)	RISK (man-Rem/Yr)
1	8.8E-6	5.4E+6	48
2	1.7E-7	9.3E+6	2
3	2.7E-7	5.4E+6	1
4	1.1E-6	2.7E+7	28
S	1.0E-8	4.6E+7	0

TABLE 2-3

COST/BENEFIT COMPARISON

MITIGATING SYSTEM	BENEFIT	TOTAL BENEFIT	TOTAL COST	BENEFIT/ COST RATIO	COST/ MAN-REM AVERTED
Dedicated Suppression Pool Cooling	\$6,000/Yr	\$57K[1]	\$25,600K	.002	\$449,000
Enhanced Drywell Sprays	\$54,000/Yr	\$516K	\$46,500K[2] \$27,500K[3]	.011 .019	\$ 90,100 \$ 52,300
Rubble Bed Core Retention	\$13,000/Yr	\$124K	\$38,400K	.003	\$310,000
Dry Crucible Core Retention	\$57,000/Yr	\$545K	\$119,000K	.005	\$218,000
ATWS Vent	\$27,000/Yr	\$258K	\$ 3,900K	.066	\$ 15,100
Filtered Vent (Gravel Bed)	\$24,000/Yr	\$229K	\$11,300K	.020	\$ 49,300
Filtered Vent (MVSS)	\$24,000/Yr	\$229K	\$ 5,700K	.040	\$ 24,900
Large Hydrogen Recombiner	\$ 0/Yr	\$ 0	\$ 5,200	.0	-
Large Vacuum Breakers	\$ 0/Yr	\$ 0	0	.0	-

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1 K denotes that the item is in thousands of dollars

2 New drywell spray nozzle distribution header

3 Use of existing drywell spray header

INDIVIDUAL

SAMDA

ASSESMENTS

Dedicated Suppression Pool Cooling

System Description: This system is designed to remove heat from the containment (suppression pool) during an accident where other means of pool cooling have been lost. It provides an independent means of pool cooling by circulating suppression pool water through a heat exchanger and returning the water to the suppression pool. Cooling water from the spray pond will be circulated through the shell side of a heat exchanger and returned to the spray pond. Pump motive power is provided by an independent diesel located in a new structure; the pumps are shaft driven from the diesel engine. Consistent with Reference 1 the assumed capacity of each pump is 3200 gpm, and the heat exchanger (approx. 4000 ft^2) removes 45 MWt.

The new structure 25' x 40' x 20' high, will be located underground. Three new power supplies will be housed in the new structure. A diesel engine will be mechanically connected to both the pool and pond pumps. A diesel generator (D/G) will provide a small source of AC power for operating the isolation valves at the containment penetrations and at the service water tie-ins, for operating the HVAC, and for miscellaneous services. The third power supply is a battery-backed power supply in the new structure for cranking the diesel sets. The system will be either manually or automatically actuated.

Sequences Mitigated: This system will mitigate accident sequences where containment failure occurs due to steam overpressurization. It will prevent containment failure and core melt for Class 2 sequences involving loss of containment heat removal (e.g., TW). The heat removal capacity of the system as designed, is insufficient to prevent pool heatup, containment overpressure failure and the resulting core melt for the Class 4 ATWS sequences. This system has a low probability of mitigating Class 1 and Class 3 sequences since drywell failure from other mechanisms (eg., overtemperature) is not prevented.

Qualitative Benefit: This system can be highly effective in preventing containment failure and the resulting core melt for Class 2 sequences. Class 4 ATWS sequences will not be mitigated. Class 1 and 3 sequences will be successfully mitigated only if drywell overtemperature failure is avoided. Overtemperature drywell failure can be prevented if the drywell sprays are operating (see section on Enhanced Drywell Sprays).

Negative Safety Implications: This system involves extending the containment boundary outside of the secondary containment. A leak or break in the piping carrying radioactive fluids could lead to an uncontained radioactive material release, draining of the suppression pool and loss of containment integrity.

Quantitative Benefit: The dedicated pool cooling system is estimated to provide the following risk reduction in man-rem per year.

Class	Man-rem per year Reduction
1	5
2	1
3	0
4	0
Total	6

6 Man-rem per year at $1000 per man-rem yields $6,000 per year or an approximate present worth benefit of $57,400.

Costs:	Initial Investment	$ 23,117,500
	O & M (Present Worth):	$ 2,495,000
	Total	$ 25,612,500

Conclusion: These benefits do not exceed the estimated costs of $25.6 million and this mitigation device is not considered cost-beneficial.

Enhanced Drywell Spray System (EDSS)

System Description: This system is designed to remove heat from the containment, provide cooling water to debris in the drywell following vessel failure, prevent high temperatures in the drywell and scrub fission products from the drywell atmosphere and/or limit radionuclide release from core debris/concrete interactions during a severe accident, where other means of containment heat removal and the existing sprays are inoperable.

The system is designed to circulate 3200 gpm of suppression pool water through a heat exchanger and to spray this cooled water into the drywell. The dedicated suppression pool cooling system (DSPCS) (previously described) removes heat by cooling the suppression pool water and discharging the removed heat to the spray pond. The suppression pool water is discharged through the drywell sprays and is returned to the suppression pool via the downcomers between the drywell and the wetwell. The incorporation of the EDSS requires, in addition to the distribution headers, additional valves and control circuitry from those envisioned for the DSPCS. The spray system will be initiated on very high drywell pressure or very high drywell temperatures; if the DSPCS portion of the system was previously initiated, the flow will be diverted to the EDSS. If, for some reason, the DSPCS is not operating, these same pressure or temperature signals will initiate EDSS operation. The appropriate indications and controls will be provided in the control room. This system is a extension of the dedicated pool cooling system discussed separately in this report.

Sequences Mitigated: This system will mitigate all classes of accident sequences. It will prevent containment failure and core melt for Class 2 sequences involving loss of containment heat removal (e.g., TW). The heat removal capacity of the system as designed is insufficient to prevent pool heatup, containment overpressure failure and the resulting core melt for the Class 4 ATWS sequences. However, this system will partially mitigate the radionuclide releases by attenuating radionuclides in the drywell atmosphere. It will prevent containment overpressure failure and drywell overtemperature failure for Class 1 and 3 loss of core coolant injection sequences. Hence, there is a high probability of this system mitigating Class 1 and 3 sequences.

Qualitative Benefit: This system can be highly effective in preventing containment failure and the resulting core melt for Class 2 sequences. Class 4 ATWS sequences will be only partially mitigated. Class 1 and 3 sequences will be successfully mitigated.

Negative Safety Implications: Same as for dedicated suppression pool cooling system.

Quantitative Benefit: The enhanced drywell spray system is estimated to provide the following risk reduction in man-rem per year.

Class	Man-rem per year Reduction
1	43
2	1
3	1
4	9
Total	54

54 man-rem per year at $1000 per man-rem yields $54,000 per year or a approximate present worth benefit of $516,000.

Costs: The costs shown here for the EDSS also includes the costs associated with the dedicated suppression pool cooling system into which the EDSS is integrated.

Option 1 presents the costs assuming new and separate drywell spray headers are required. Option 2 presents the costs assuming the spray headers and nozzles from one train of the existing drywell spray system can be used.

	Option 1	Option 2
Initial Investment	$44,016,500	$24,517,000
O & M (present worth)	$ 2,533,000	$ 2,514,000
Total	$46,549,500	$27,031,000

Conclusion: These benefits do not exceed the estimated costs of $ 46.5 million and $27.0 million and this mitigation device is not considered cost-beneficial.

Rubble Bed Core Retention Device

System Description: This system consists of a floodable rubble bed core retention device located in the lower pedestal area of the wetwell. It is designed to hold and cool the debris, and prevent debris penetration through the basemat into the soil.

In the Limerick plant, the suppression pool water extends into the lower central pedestal area. In this concept, the hot core melt debris would be directed through 12-inch diameter holes in the diaphragm floor and allowed to drop into the lower pedestal area onto a bed of rubble covered by thoria plates. The inside diameter of the pedestal at the basemat is approximately 20 feet and therefore, the volume of the core material would fill this area to a depth of less than 4 feet even allowing for 50 percent voids.

This concept is similar to the design illustrated schematically in Figure 3-13 in Reference 1. A stainless steel cylinder is constructed to act as a heat shield for the concrete walls and prevent excessive decomposition. Heat would be removed from the steel cylinder by surrounding water at the lower elevations and radiation and convection at the higher elevations. Thoria plates would also be added and extended up the sides a few feet, if necessary.

To preclude a steam explosion and minimize ex-vessel hydrogen generation, the core debris retention system is kept essentially dry until after the hot core debris falls onto the rubble bed. Only after the material has penetrated into the rubble bed area and been cooled somewhat would water be allowed to percolate up through the bed.

Sequences Mitigated: Aside from assuring that the debris will not penetrate into the surrounding soil (a low probability event in any case) this system will provide limited additional mitigation. This system will not prevent containment failure and the resulting core melt for the Class 2 loss of containment heat removal system sequences or for the Class 4 ATWS sequences. This system may be successful in preventing containment overpressure failure and overtemperature drywell failure by directing the debris away from the drywell onto the rubble bed in the wetwell pedestal and cooling the debris for Classes 1 and 3 loss of core cooling injection sequences.

Qualitative Benefit: This system has a limited potential for successfully mitigating Class 1 and 3 sequences and essentially no mitigation potential for Classes 2 and 4.

Negative Safety Implication: None found.

Quantitative Benefit: The rubble bed is estimated to provide the following risk reduction in man-rem per year:

Class	Man-rem per year Reduction
1	12
2	0
3	1
4	0
Total	13

13 man-rem per year at $1000/per man-rem yield $13,000 per year or an approximate present worth benefit of $124,000.

Costs:	Initial Investment:	$37,979,000
	O & M (Present Worth)	$ 377,500
	Total	$38,356,500

Conclusion:

The benefits of this system are far below the estimated cost of $38.4 million and this mitigation device is not considered to be cost effective.

Cooled Dry Crucible Core Retention Device

System Description: The dry crucible retention device is located below the basemat of the present containment. The truncated cone-shaped crucible shown in Figure 3-5 of Reference 1, is 6 feet in diameter at the top, 3 feet in diameter at the bottom and about 70 feet long to allow for easy entrance of the molten mass. For this concept, a number of large holes (at least 4 - 12" diameter) will be drilled through the diaphragm floor to direct debris flow to the pedestal area. These holes will be sealed during normal operation by fusible metal plates.

The pedestal area at the basemat is filled with water. This must be blocked off so the area is dry and the core debris can drop through the holes formed after melting the plates in the diaphragm slab. Then the hot debris will readily melt through a succession of thin steel barriers and drop into the lower crucible cone. The cone is waterjacketed and supplied with forced circulation to remove residual heat. The cooling water would be pumped and cooled by a dedicated heat removal system similar to the system described in the dedicated suppression pool cooling system option. Suppression pool water would be removed from the core catcher area, pumped through the heat exchanger, core catcher and then the drywell sprays.

This option would require a 6 to 8 foot diameter hole through the basemat which accommodates the upper section of the core retainer. The material can be broken up and removed out of the access tunnel. The access tunnel will be used for carrying all the required material for fabrication and installation of the core catcher crucible. When installation of the dry crucible and supporting equipment is completed, the tunnel will be used for normal access to the supporting equipment.

Unidentified complexities and problems are likely to arise during the licensing, design and implementation of this concept. Since no plant has attempted a similar modification, these unidentified problems are expected to significantly increase the estimated costs. Examples of the uncertainties involved include: impact to the plant during excavation, the effects on the seismic design resulting from a major change to the containment design, and the effort required to drill an 8 foot diameter hole through the containment basemat.

Sequences Mitigated: Aside from assuring that the debris will not penetrate into the surrounding soil (a low probability event in any case), the core retention portion of this mitigation system will provide limited additional mitigation. However, the drywell spray portion of this system will provide substantial benefits comparable to the Enhanced Drywell Spray System described previously.

Qualitative Benefit: Comparable to Enhanced Drywell Spray System

Negative Safety lications: A break or le in the line carrying radioactive fluids outside containment could lead to release of radionuclides, draining of the pool and loss of containment integrity.

Quantitative Benefits: The dry crucible with drywell spray is estimated to provide the following risk reduction in man-rem per year:

Class	Man-rem per year Reduction
1	45
2	1
3	1
4	10
Total	57

57 Man-rem per year at $1000 per man-rem yields $57,000 per year or an approximate present worth benefit of $545,000.

Costs:		
	Initial Investment:	$ 116,817,000
	O & M (Present Worth)	$ 1,945,000
	Total	$ 118,762,500

Conclusion:

The benefits of this system are far below the estimated cost of $119 million and this mitigation device is not considered to be cost effective.

ATWS Clean Steam Vent

System Description: This system consists of an unfiltered high capacity vent pathway from the wetwell airspace to the atmosphere. This system is designed to relieve the steam generated during an ATWS (Anticipated Transient Without Scram) when reactor coolant makeup is available and where the reactor stabilizes at an average power level of 10% of full rated power. Steam is relieved to the suppression pool via the main steam safety relief valves; "clean" steam is then vented to the stack from the suppression pool air space. The system consists of piping from the Unit 1 and Unit 2 suppression chambers to the north stack which is shared by both Units. New piping would be connected to the existing 18-inch purge lines close to the containment penetration and upstream of the containment isolation valves.

Containment isolation is maintained by two normally-closed, air-operated, valves in series followed by a rupture disc. Following an ATWS, the operator could open these valves by means of a key-locked, administratively-controlled switch; if suppression chamber pressure exceeds approximately 70 psig, the rupture disc will open, allowing the excess steam associated with the ATWS to be vented to the atmosphere via the north stack. The air-operated valves are provided with a dedicated power supply and accumulator backup.

The vent lines from Unit 1 and Unit 2 are joined just before entering the stack. In addition to the normally-closed isolation valves and the rupture disc, each line is provided with a check valve as a further means of preventing the spread of radioactivity from the Unit undergoing the accident to the other Unit.

Sequences Mitigated: This system will mitigate accident sequences where containment failure occurs due to overpressurization from slow or moderate steam production rates. It will prevent containment failure and the resulting core melt for Class 4 ATWS sequences [1]. It will also prevent containment failure and core melt for Class 2 (e.g., TW sequences) characterized by loss of containment heat removal. The system will also prevent overpressure containment failure and provides attenuation of the radionuclides for Class 1 (and 3) sequences (such as TQUV and station blackout) characterized by loss of coolant injection to the core. However, to achieve this benefit drywell failure by other failure modes such as overtemperature and drywell to wetwell pool bypass (e.g., drywell pedestal liner plate failure) must be prevented.

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(1) In the absence of containment failure it is assumed that core makeup continues for a sufficient time period to allow alternative means of reactor shutdown to succeed.

Qualitative Value: This system will be effective in preventing core melt in Class 2 and 4 sequences and can be effective in mitigating Class 1 and 3 sequences if drywell overtemperature failure and drywell to wetwell pool bypass are prevented. Class 4 sequences appear to be more difficult to mitigate than other types of sequences. This analysis assumes that the steam can be successfully vented at the design flow rate and that the ATWS sequences will be mitigated.

Negative Safety Implications: Inadvertent venting during an accident after radionuclide release has occurred to the containment atmosphere prior to containment overpressurization could release noble gases and a moderately small fraction of the other radionuclides. After vessel failure the release could be large because of pool bypass.

Quantitative Value: The ATWS clean steam vent is estimated to provide the following risk reduction in man-rem/year.

Class	Man-rem per year Reduction
1	1
2	1
3	0
4	25
Total	27

27 man-rem per year at $1000/man-rem yields $27,000/year or an approximate present worth benefit of $258,000.

Costs:		
	Initial Investment:	$3,526,500
	O & M: (Present Worth)	$ 353,500
	Total	$3,880,000

Conclusion:

The benefits do not exceed the estimated cost of $3.9 million of the system and this mitigation device is not considered to be cost-beneficial.

Filtered-Vent System

System Description: This system provides a vent pathway from the drywell to a steam condensing and fission product removal device and from there to an elevated release point. The system is designed to provide the following functions: (1) remove 99% of the radionuclides in particulate form and 99% of the molecular iodine, (2) accept primary system stored energy and decay heat for 24 hours without external cooling, and (3) process 35 lbm/s of steam/non-condensible gases at 70 psig drywell pressure.

A hard pipe vent path is provided from each unit to a common filtering device. Valving, a rupture disk, and vacuum breakers are located in each vent path for operational purposes. A new vent stack is located at the filter to provide an elevated release point for the filtered stream.

Two filter options have been included in this assessment. The first option is a gravel bed filter (similar to the FILTRA device used at Barseback in Sweden) and the second option is a multi-venturi wet scrubber (similar to the filtering devices used on all other reactors in Sweden). Both devices will meet the design performance requirements.

Sequences Mitigated: This device will mitigate sequences where containment failure occurs due to slow steam overpressurization. This system will prevent overpressure containment failure and mitigate the radionuclide release for Class 1 and 3 sequences such as transient initiated and fire initiated sequences which are characterized by loss of core coolant injection (e.g., TQUV, station blackout). This device will prevent overpressure containment failure and subsequent core melting for Class 2 sequences such as transient sequences characterized by loss of containment heat removal (e.g., TW). This device does not have sufficient capacity to relieve the steam generated by an ATWS event and hence will not prevent containment failure and core melt for the Class 4 sequences. This device is insensitive to drywell to wetwell pool bypass events (such as drywell pedestal drain line plate failure). However, drywell failure from other mechanisms such as overtemperature will compromise the system.

Qualitative Benefit: This system can be highly effective in mitigating Class 1, 2 and 3 sequences if drywell failure from overtemperature can be prevented.

Negative Safety Implications: Inadvertent or early opening of the filtered-vent during an accident could release noble gases and a very small fraction of other radionuclides at a time when the containment is not threatened.

Quantitative Benefit: The filtered vent is estimated to provide the following risk reduction in man-rem per year.

Class	Man-rem per year Reduction
1	23
2	1
3	0
4	0
Total	24

24 Man-rem per year at $1000/man-rem yields $24,000/year or an approximate present worth of $229,000.

Costs:

	Gravel Bed Filter	Multi-Venturi-Scrubber System
Initial Investment:	10,898,000	5,285,500
O & M (Present worth)	420,500	406,500
Total	$11,318,500	$ 5,692,500

Conclusion:

The benefits do not exceed the estimated cost of $11.3 million for the gravel bed filter or $5.7 million for the multiventuri scrubber and neither mitigation device is considered to be cost beneficial.

Large Hydrogen Recombiners

System Description: The purpose of this system is to recombine free hydrogen with oxygen to eliminate the potential for uncontrolled combustion. Hydrogen is generated during a postulated severe accident during the oxidation of metals and from radiolysis of water. The recombiners are not expected to be required prior to venting. After the containment has been vented, oxygen may be introduced to the containment and the volume percent oxygen may be increased with operation of the containment sprays which would tend to condense the steam in the containment atmosphere. Hydrogen/oxygen recombination will then be required to prevent the long-term formation of combustible concentrations, as hydrogen and oxygen will continue to be generated due to radiolysis of water and steam inside the containment.

Limerick's primary containment is inerted with nitrogen. The existing hydrogen recombiners are designed and operated to control the containment oxygen concentration to below 5% to prevent hydrogen combustion. The proposed system is specified to be designed for 70 psig containment pressure and capable of processing the containment volume within 2-3 weeks. A dedicated power supply is provided but is probably not required since normal plant power sources should be available over the long periods of time when the system is to be used.

The existing Limerick Hydrogen Recombiner System consists of redundant combiners located outside primary containment in the reactor enclosure. The existing hydrogen recombiners can meet the specified capacity requirement for a severe accident and the design concept for this system is to employ the existing hydrogen recombiners, upgrading them to withstand the specified design conditions and providing a dedicated power supply.

Sequences Mitigated: This system does not prevent (early) containment failure or mitigate radionuclide release for any identified accident sequence. It is viewed as more of a long-term accident recovery system than a short-term mitigation system.

Qualitative Benefit: Reduces the risk of a hydrogen burn if air is reintroduced into the containment following venting to relieve an internal underpressurization condition.

Negative Safety Implication: None found.

Quantitative Benefit: No PRA to-date has assessed the risk of very late hydrogen combustion resulting from air introduction following venting into a normally inerted containment. It is judged that the risk reduction potential of this system is small.

Costs:	Initial Investment:	$4,819,500
	O & M (present worth)	$ 392,000
	Total	$5,211,500

Conclusions:

Since this system is assessed as having a very small benefit and its costs are high, it is not considered a cost-beneficial system.

Large Containment Vacuum Breaker System

System Description: This system provides a large diameter path from atmosphere to containment for use when a high degree of vacuum occurs in containment. In essence, it would consist of a large pipe with at least two check valves in the line.

Sequence Mitigation: As in Reference 1 the purpose of this system would be to avert containment failure due to external overpressure. A qualitative assessment by the Boiling Water Reactor Owners' Group of the conditions that would lead to large negative pressures concluded that such conditions are not expected following recovery of normal containment heat removal and termination of venting. Additionally the reinforced concrete Mark II containments such as Limerick are not expected to fail even for pressure differentials exceeding twice the design differential pressure of 5 psid. Therefore the vacuum breaker would not mitigate any accident sequences currently identified.

Qualitative Benefit: None

Negative Safety Implications: Any vacuum breaker actuation would introduce oxygen into the containment and may produce conditions suitable for hydrogen combustion to occur.

Quantitative Benefit: None

Costs: Not estimated

Conclusion:

This system was not quantitatively assessed because of the determination of no benefit.

QUESTION 3

Provide the results from (1) and (2) above. In view of the positive choice by PECo to maintain its PRA in a "living" status since the PRA became available, you may elect to use the PRA insights to enumerate and briefly discuss those various alternatives considered in the interim and/or improvements actually made to the plant design and operational procedures, that would in your judgement, serve the objectives of (2) above and have served to increase the level of public protection through either prevention or mitigation of severe accidents.

RESPONSE

There are several areas where PRA insights have influenced design and procedural enhancements and increased the level of public protection through either prevention or mitigation of severe accidents.

Design Considerations

The Limerick PRA/Severe Accident Risk Assessment (SARA) influenced several design features that were installed in Unit 1 prior to its licensing:

1. ATWS Alternate 3A fixes including alternate rod insertion, recirculation pump trip, redundant and diverse scram volume instrument sensors, MSIV isolation setpoint change from level 2 (-38") to level 1 (-129"), and standby liquid control system enhancements including the addition of a third pump, automatic initiation, injection through the core spray sparger, use of redundant penetrations for injection, and arrangement of equipment for enhanced testability.

2. ADS air supply considerations including the type and location of backup supplies, physical arrangement of piping and valves, use of dual pilot solenoid valves, and the design of safety/non-safety interfaces.

3. MSIV air supply improvements.

4. Fire propagation barriers for reactor enclosure equipment hatches.

Other PRA supported design changes implemented subsequent to the NRC review of the Limerick PRA/SARA are:

1. Improved ADS initiation logic, in response to TMI Action Plan Item II.K.3.18, which uses a timer to bypass the high drywell pressure permissive.

2. Addition of manual ADS inhibit switches to improve implementation of the BWR Owners Group Emergency Procedure Guidelines (EPGs).

Additionally, it should be noted that even though they tend to reduce risk and core damage frequency, the benefit of the existing drywell spray and CRD systems have not been formally quantitatively assessed and included in the PRA at the present time.

A cost/benefit analysis of installation of a combustion gas turbine was performed as a possible design alternative. The conclusion reached was that installation of a combustion gas turbine for restoring power after a station blackout is not cost effective. The benefit gained is small compared to the cost of making the modification and maintaining it over the life of the plant.

<u>Procedural Considerations</u>
Improvements in current operational procedures over those in place at the time of the NRC review of the Limerick PRA/SARA, have reduced risk. The Transient Response Implementation Plan Procedures, the Limerick-specific emergency operating procedures, were found to give clear guidance to the operators to gain control of potential accident events. Operator actions of venting containment and maintaining injection to the vessel are considered in the updated PRA. Limerick has implemented Revision 3 of the BWR Owners Group EPGs and Secondary Containment Control and Radioactivity Release Control from Revision 4 of the BWR Owners Group EPGs. Limerick is scheduled to implement the remainder of Revision 4 of the BWR Owners Group EPGs by the end of 1989. The BWR Owners Group review of the applicability of EPG, Revision 4, to severe accidents concluded that EPG, Revision 4, is a set of effective accident management procedures capable of contributing to the prevention and mitigation of the consequences of core melt. The NRC Safety Evaluation Report Issued September 12, 1988, stated "We believe that the BWR Emergency Procedure Guidelines (EPG) provide a basis for a significant improvement in current emergency operating procedures."

Other operational procedures implemented subsequent to the NRC review of the Limerick PRA/SARA include procedures following a loss of offsite power or following a station blackout. Actions directed by the station blackout procedure include establishing alternate HPCI/RCIC room cooling, reducing reactor pressure to minimize drywell heatup, and isolating unnecessary DC loads.

In the process of performing the work associated with incorporating the TRIP procedures into the PRA, areas of the procedures were identified where enhancements were suggested and made. The following procedural enhancements have been accomplished:

1. The instruction to inhibit ADS for an ATWS has been moved to avoid possibly missing the instruction at a branch in the procedure.

2. The ATWS procedures have been revised to call for bypassing the level one MSIV closure signal prior to the required lowering of the reactor water level for turbine trip ATWS with a stuck open relief valve.

3. The instruction to intentionally deenergize the reactor enclosure when venting the containment with the large 18" and 24" lines has been eliminated.

4. The containment venting procedure has been modified so that with high rates of pressure rise the large (18" and 24") vent paths are opened rapidly.

TABLE 1 - 1

OCCUPATIONAL EXPOSURE EVALUATION

OPTION	AREA/ELEV.	LOCATION	MANHOURS	DOSE RATE	EXPOSURE (MAN REM)	TOTAL OPTION EXPOSURE
A1 HEAT REMOVAL-POOL	18/177	RHR COMPARTMENT	5300	5MR/HR	26.5	
	18/201	RHR COMPARTMENT	8800	2MR/HR	17.6	
	18/201	PIPE TUNNEL	6900	2MR/HR	13.8	57.9 MAN REM
A2 HEAT REMOVAL-SPRAY (NEW DW SPRAY HDR.)	18/177	RHR COMPARTMENT	5300	5MR/HR	26.5	
	18/201	RHR COMPARTMENT	8800	2MR/HR	17.6	
	18/201	PIPE TUNNEL	6900	2MR/HR	13.8	
	18/217-283	REACTOR ENCL. AREA	6000	3MR/HR	18.0	
		DRYWELL	34000	10MR/HR	340.0	415.9 MAN REM
A3 HEAT REMOVAL-SPRAY (USE EXISTING DW SPRAY HDR.)	18/177	RHR COMPARTMENT	5300	5MR/HR	26.5	
	18/201	RHR COMPARTMENT	14670	2MR/HR	29.3	
	18/201	PIPE TUNNEL	6900	2MR/HR	13.8	69.6 MAN REM
B1 ATWS CLEAN VENT	13/217	REACTOR ENCL. AREA	6000	1.5MR/HR	9.0	
	13/253	REACTOR ENCL. AREA	1800	1.0MR/HR	1.8	
	13/283	REACTOR ENCL. AREA	2200	0.5MR/HR	1.1	
	13/313	REA & NORTH STACK	5000	0.5MR/HR	2.5	14.4 MAN REM
B2 FILTERED VENT-GRAVEL BED	16/198	PIPE TUNNEL	900	0.2MR/HR	0.2	
	17/198	PIPE TUNNEL	600	0.2MR/HR	0.1	
	17/201	REACTOR ENCL. AREA	1800	0.5MR/HR	0.9	
	17/201	RHR COMPARTMENT	700	2.0MR/HR	1.4	
	17/217	RHR VALVE COMPARTMENT	1300	0.2MR/HR	0.3	
	17/238	RHR VALVE COMPARTMENT	100	0.2MR/HR	------	2.9 MAN REM
B3 FILT. VENT-MULT. VENTURI	16/198	PIPE TUNNEL	1100	0.2MR/HR	0.2	
	17/198	PIPE TUNNEL	700	0.2MR/HR	0.1	
	17/201	REACTOR ENCL. AREA	2100	0.5MR/HR	1.1	
	17/201	RHR COMPARTMENT	900	2.0MR/HR	1.8	
	17/217	RHR VALVE COMPARTMENT	1700	0.2MR/HR	0.3	
	17/238	RHR VALVE COMPARTMENT	100	0.2MR/HR	------	3.5 MAN REM
D1 CORE CATCHER DRY CRUCIBLE		INSIDE PEDESTAL	56000	0.2MR/HR	11.20	
		WETWELL	8000	0.2MR/HR	1.6	
		DRYWELL	16000	10.0MR/HR	160.0	172.8 MAN REM
D2 CORE CATCHER RUBBLE BED		INSIDE PEDESTAL	18900	0.2MR/HR	3.8	
		BELOW RPV	2000	5.0MR/HR	10.0	13.8 MAN REM

References For Question 1 Response

1. "Severe Accident Risk Assessment, Limerick Generating Station", Philadelphia Electric Company, April 1983.

2. "Final Environmental Statement Related to the Operation of Limerick Generating Station Units 1 and 2" USNRC, NUREG-0974, March 1984.

3. "Review Insights on the Probabilitistic Risk Assessment for the Limerick Generating Station", USNRC, NUREG-1068, August 1984.

4. "Probabilitistic Risk Assessment, Limerick Generating Station", Philadelphia Electric Company, September 1982.

5. Lambright, J. A., et al., "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues", Sandia National Laboratories, NUREG/CR-5088, January 1989.

Response of Philadelphia Electric Company
to Question 2

2. Provide an evaluation of the incremental environmental effects from the risk of severe accidents of operation of Limerick Unit 2 with no SAMDAs in place for one fuel cycle. [Note that NUREG-1068 and the references cited therein provide numerical estimates of the public risk (e.g., early and latent fatalities per year, person-rem per year) associated with full power operation of the Limerick facilities.]

The Philadelphia Electric Company (PECO) assessment of the environmental effects due to severe accidents at the Limerick Generating Station was reported in the Severe Accident Risk Assessment (SARA) (Ref. 1) in 1983. The NRC Staff assessment was provided in the Limerick Final Environmental Statement (Ref. 2) and both were summarized in NUREG-1068 (Ref. 3). These assessments were based on the Limerick design, configuration and procedures in place in the 1981-1982 time frame as well as risk assessment methods and technology of that time.

Subsequent to 1981, a considerable number of plant upgrades and improvements in probabilistic risk assessment technology have occurred. In addition, we now have over three years of experience with the operation of Limerick Unit 1. Examples of changes at the plant include implementation of symptom based emergency operating procedures, installation of Automatic Depressurization System logic modifications

(TMI action item II.K.3.18) and lowering of the Main Steam Isolation Valve low water level closure setpoint. PRA technology changes include availability of more extensive data bases for transient initiator frequencies, component failure rates, fire and fire suppression rates, and a better understanding of severe accident phenomenon. Plant operating experience has been very good, indicating lower transient frequency than generic values.

In order to provide a better basis for the evaluation of the need for installation of SAMDAs at Limerick, PECO developed an updated risk analysis to account for the changes to plant design and operation since the earlier assessment and to address the NRC Staff as well as Brookhaven National Laboratory comments on the original Limerick PRA and SARA.

Plant risk is determined from the frequency of core damage sequences for all initiators (Level 1 PRA), an assessment of containment performance and resulting radionuclide source term for each sequence (or group of sequences) (Level 2 PRA) and an assessment of the consequences of the releases (Level 3 PRA).

Subsequent to the initial development of the Limerick Probabilistic Risk Assessment (Ref. 4), in response to the Commission's May 6, 1980 letter, and

the Severe Accident Risk Assessment which was developed in accordance with the requirements of the National Environmental Policy Act, PECO's PRA activities have concentrated on the updating and use of the internal initiator portion of the Level 1 PRA in accordance with the Commission Staff's June 7, 1984 letter and PECO's July 23, 1984 response.

The core damage frequencies for the internally-initiated sequences used for the current risk estimate are based on a November 1988 update of the LGS-PRA modified to include a Limerick turbine trip frequency of 2.55 scrams/year justified by actual Limerick operating experience over the first two operating cycles. The frequency of other initiators (other transients and LOCAs) remains the same. The current total transient frequency utilized is 6.7/year. This is conservative and is expected to go down further as additional site-specific data are accumulated.

The externally-initiated sequences have been selectively updated to account for significant new information. Further detail is contained in the attached PECO letter of June 23, 1989, responding to the NRC Staff's letter of May 23, 1989 (Question 1).

The resulting updated core damage frequency and comparison to the 1983 SARA results are given in Table 2-1.

The risk resulting from these severe accidents is based on the containment, source term and consequence analysis of SARA with only one modification to account for the benefit of the existing plant capability to spray or inject water into the drywell after core damage occurs. This was conservatively omitted from the original PRA/SARA analysis. Even with this modification, the source terms and resulting risks are believed to be conservative as they are based on source term technology of the 1981-1983 time frame.

The resulting risks of severe accidents at Limerick Unit 1 are given in Table 2-2. While all the estimates are for Unit 1, the units are essentially identical. Hence, Unit 1 results are applicable to Unit 2. Also shown in Table 2-2 is the risk after installation of the most nearly cost/beneficial SAMDA as evaluated for the June 23, 1989 submittal to the NRC Staff, i.e., the ATWS clean steam vent. As noted in that submittal, even that SAMDA has a projected benefit/cost ratio of only .066 (see letter of June 23, 1989 at Table 2-3), i.e., is greater than a factor of 15 from being cost beneficial.

To clarify, the values of risk given in Table 2-2 are derived from the fission product inventory that would exist for an equilibrium fuel cycle. For the first fuel cycle there is initially all new fuel, hence the fission product inventory and the risk are lower. To estimate the associated reduction in risk, the relative effect of fission product inventory on risk has been estimated for the middle of the first fuel cycle compared to the equilibrium fuel cycle. This has been done using inventories calculated by the ORIGEN code and early and latent fatality weights based on the approach developed for NUREG-1150. The result of this analysis is that the latent fatality risk for the first fuel cycle is about 88% of that at equilibrium while the early fatality risk is essentially unchanged at about 97% of that at equilibrium. The population dose and individual exposure reductions for the first fuel cycle are the same as those for latent fatalities.

In summary, because the projected environmental risk of a severe accident from the operation of Limerick Unit 2 is already very small, any risk reduction achieved through installation of a SAMDA will necessarily be proportionally small. The SAMDA estimated to be most nearly cost beneficial (but more than a factor of 15 less than a positive cost/benefit balance), the ATWS clean steam vent, would reduce the already low population exposure risk for the first fuel cycle by about 19%, or only 4.0×10^{-6} rem for each person within 50 miles of Limerick. Accordingly, the incremental environmental effects from postponing installation of SAMDAs, if any, at Limerick Unit 2 until the first refueling outage are almost negligible.

TABLE 2-1

CURRENT ESTIMATE OF
CORE DAMAGE FREQUENCY
(per Reactor Year)

Internal Initiators		5.9E-06
Transients	(2.1E-06)	
Loss of Offsite Power	(2.3E-06)	
ATWS	(1.2E-06)	
LOCA	(2.7E-07)	
Seismic		3.4E-06
Internal Fires		4.2E-06
Others (Internal Floods and Other Special Initiators)		0.2E-06
Total Estimated CDF		1.37E-05

Note: The total estimated CDF from the November 1983 SARA is 2.4E-05 per reactor year

TABLE 2-2

CURRENT ESTIMATE OF
SEVERE ACCIDENT RISK
LIMERICK GENERATING STATION
UNIT 2

	Risk Per Reactor Year		Risk For First Fuel Cycle[3]	
	No SAMDA	With SAMDA	No SAMDA	With SAMDA
Early Fatalities	3.9E-04	2.3E-04	5.7E-04	3.3E-04
Latent Fatalities[1]	1.5E-02	1.1E-02	2.0E-02	1.5E-02
Population Dose[1] (person-rem)	131	104	173	137
Individual Exposure[2] (rem)	1.6E-05	1.3E-05	2.1E-05	1.7E-05

Notes

1. Based on population out to 50 miles.
2. Mean individual exposure for population within 50 miles.
3. For 18 month fuel cycle with middle of cycle 1 fuel inventory.