

August 24, 1989

Docket Nos. 50-352  
and 50-353

Mr. Richard Sanderson, Director  
Office of Federal Activities  
Environmental Protection Agency  
Room 2119 Mall (A-104)  
ATTN: Management Information Unit  
401 M Street, SW  
Washington, DC 20460

DISTRIBUTION: w/o enclosure  
Docket File\* OGC  
NRC PDR\* EJordan  
Local PDR\* BGrimes  
PDI-2 Reading\* ACRS (10)  
SVarga \*w/enclosure  
BBoger  
WButler  
MO'Brien  
GSuh  
RClark  
ETrottier

Dear Mr. Sanderson:

SUBJECT: SUPPLEMENT TO THE FINAL ENVIRONMENTAL STATEMENT

RE: LIMERICK GENERATING STATION, UNITS 1 AND 2

Please find enclosed a loose-leaf copy of a supplement, with abstract, to NUREG-0974, "Final Environmental Statement related to the operation of Limerick Generating Station, Units 1 and 2." As part of its response to a February 28, 1989 decision by the U.S. Court of Appeals for the Third Circuit, the NRC prepared this supplement to the Final Environmental Statement to present its evaluation of the alternative of facility operation with the installation of further severe accident mitigation design features. This supplement was prepared pursuant to 10 CFR § 51.92(b) of the Commission's regulations.

Bound copies of the supplement will be available in the near future. We are transmitting the enclosed copy for your information in advance of the bound copies to provide the earliest possible delivery of the supplement to the Environmental Protection Agency and the Commonwealth of Pennsylvania. An advance copy has also been sent to the Philadelphia Electric Company and other interested parties. Distribution of the bound report will be made to the complete distribution list and notice of availability will be made in the Federal Register as soon as printing of the report permits.

Sincerely,

/s/

8908310208 890824  
PDR ADOCK 05000352  
P PNU

Gene Y. Suh, Project Manager  
Project Directorate I-2  
Division of Reactor Projects I/II  
Office of Nuclear Reactor Regulation

Enclosure:  
As stated

cc:  
See next page

[SANDERSON LETTER]

PDI-2/LA  
MO'Brien  
8/24/89

PDI-2/PM  
GSuh:mj  
8/24/89

PDI-2/D  
WButler  
8/24/89

OGC  
8/24/89

DF01  
1/1



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555  
August 24, 1989

Docket Nos. 50-352  
and 50-353

Mr. Richard Sanderson, Director  
Office of Federal Activities  
Environmental Protection Agency  
Room 2119 Mall (A-104)  
ATTN: Management Information Unit  
401 M Street, SW  
Washington, DC 20460

Dear Mr. Sanderson:

SUBJECT: SUPPLEMENT TO THE FINAL ENVIRONMENTAL STATEMENT

RE: LIMERICK GENERATING STATION, UNITS 1 AND 2

Please find enclosed a loose-leaf copy of a supplement, with abstract, to NUREG-0974, "Final Environmental Statement related to the operation of Limerick Generating Station, Units 1 and 2." As part of its response to a February 28, 1989 decision by the U.S. Court of Appeals for the Third Circuit, the NRC prepared this supplement to the Final Environmental Statement to present its evaluation of the alternative of facility operation with the installation of further severe accident mitigation design features. This supplement was prepared pursuant to 10 CFR § 51.92(b) of the Commission's regulations.

Bound copies of the supplement will be available in the near future. We are transmitting the enclosed copy for your information in advance of the bound copies to provide the earliest possible delivery of the supplement to the Environmental Protection Agency and the Commonwealth of Pennsylvania. An advance copy has also been sent to the Philadelphia Electric Company and other interested parties. Distribution of the bound report will be made to the complete distribution list and notice of availability will be made in the Federal Register as soon as printing of the report permits.

Sincerely,

A handwritten signature in cursive script that reads "Gene Y. Suh".

Gene Y. Suh, Project Manager  
Project Directorate I-2  
Division of Reactor Projects I/II  
Office of Nuclear Reactor Regulation

Enclosure:  
As stated

cc:  
See next page

Director (A-104)  
Office of Federal Activities  
Environmental Protection Agency  
Room 2119 Mall  
ATTN: Management Information Unit  
401 M Street, SW  
Washington, DC 20460

cc: Director, Criteria and Standards  
(ANR-460)  
Office of Radiation Programs  
U.S. Environmental Protection Agency  
Washington, DC 20460

Director, Office of Radiation  
Programs  
Las Vegas Facility  
U.S. Environmental Protection Agency  
P.O. Box 18416  
Las Vegas, Nevada 89114

EIS Review Coordinator  
U.S. Environmental Protection Agency  
Region III  
841 Chestnut Street  
Philadelphia, Pennsylvania 19107

Director, Eastern Environmental  
Radiation Facility  
U.S. Environmental Protection Agency  
1890 Federal Drive  
Montgomery, Alabama 36109

## ABSTRACT

In April 1984 the staff of the Nuclear Regulatory Commission issued its Final Environmental Statement (NUREG-0974) related to the operation of Limerick Generating Station, Units 1 and 2 (Docket Nos. 50-352 and 50-353), located on the Schuylkill River, near Pottstown, in Limerick Township, Montgomery and Chester Counties, Pennsylvania.

The NRC has prepared this supplement to NUREG-0974 to present its evaluation of the alternative of facility operation with the installation of further severe accident mitigation design features. The NRC staff has discovered no substantial changes in the proposed action as previously evaluated in the Final Environmental Statement that are relevant to environmental concerns nor significant new circumstances or information relevant to environmental concerns and bearing on the licensing of Limerick Generating Station, Units 1 and 2.

## SUMMARY AND CONCLUSIONS

In February 1989, the U. S. Court of Appeals for the Third Circuit ruled that NRC failed to consider a "reasonable set" of Severe Accident Mitigation Design Alternatives (SAMDA) in the Final Environmental Statement (FES) for the Limerick Generating Station (NUREG-0974, April 1984). The NRC staff has completed consideration of a reasonable set of severe accident mitigation design alternatives. The staff has discovered no substantial changes in the proposed action as previously evaluated in the FES that are relevant to environmental concerns nor significant new circumstances or information relevant to environmental concerns and bearing on the licensing of Limerick Generating Station, Units 1 and 2.

In assessing the risk reduction potential, the value of each SAMDA was initially scoped based on risk information reported in the original Limerick Generating Station Probabilistic Risk Assessment and Severe Accident Risk Assessment and reviewed by the staff in the 1983-1984 timeframe (NUREG-1068, August 1984). Modifications were made to this information base to account for the effect of two plant improvements identified in NUREG-1068 and subsequently implemented by PECO. The risk reduction scoping estimates were compared to the estimated costs associated with each SAMDA. Based on a screening criterion of \$1000 per averted person-rem, the comparison indicated that some candidate SAMDAs warranted further evaluation.

The staff then further evaluated each of the SAMDAs, considering the qualitative effect of several plant improvements made at Limerick since the time of the staff review reported in NUREG-1068. Key plant improvements include the implementation of: procedures for battery power load shedding, MSIV air supply improvements, BWR Owners' Group Emergency Procedure Guidelines, Rev. 3 (and parts of Rev. 4), the hardened containment vent line, and procedures for the use of diesel-driven fire protection system pumps for core injection. The staff also gave consideration to the results of a recent update to the Limerick PRA described in an April 25, 1989 ACRS subcommittee meeting, a June 23, 1989 utility submittal concerning SAMDAs, and a July 27, 1989 meeting with the staff concerning the SAMDA submittal. That study calculated values of CDF and offsite dose which were about four times lower than the staff's. While the staff has not reviewed these results in sufficient detail to confirm the quantitative results, the staff believes that these plant features would reduce the CDF and offsite doses. As a result, the averted offsite dose from candidate SAMDAs could be appreciably less than estimated by the staff.

The staff also considered uncertainty in the cost and effectiveness of candidate SAMDAs. For instance, the ATWS vent analyzed by the utility uses an existing 18 inch containment penetration which would be capable of removing 10 percent of full power. There are existing analyses which predict ATWS power levels as high as 30 percent for some scenarios. The staff identified operational disadvantages for some of the candidate SAMDAs (Table 4).

Of the seven SAMDAs which passed the screening cost/benefit test, the staff has identified two which have been implemented at Limerick. These are the Decay Heat Sized Vent Without Filter (3.C.) and the Low Pressure Reactor Makeup Capability (6.) The staff has not quantified the effectiveness of these SAMDAs

in reducing risk. However, the staff believes that these features will result in an appreciable net decrease in CDF and risk.

In summary, the risks and environmental impacts of severe accidents at Limerick are acceptably low. We have found no new information that would call into question the FES conclusion that, "the risks of early fatality from potential accidents at the site are small in comparison with risks of early fatality from other human activities in a comparably sized population, and the accident risk will not add significantly to population exposure and cancer risks. Accident risks from Limerick are expected to be a small fraction of the risks the general public incurs from other sources. Further, the best estimate calculations show that the risks of potential reactor accidents at Limerick are within the range of such risks from other nuclear power plants," (NUREG-0974, Page 5-126).

Furthermore, while the screening cost/benefit analysis performed above indicates that several candidate SAMDAs might be cost effective based on a criterion of \$1000 per person-rem averted, a more recent utility PRA presents lower risk estimates which indicate that SAMDAs are not justified. While the staff has not verified the utility estimates, the staff is convinced that risk is now lower for Limerick than the estimates used in our cost/benefit study. Moreover, there are uncertainties about the costs, effectiveness, and/or operational disadvantages of some SAMDAs. In light of these considerations, the staff has no clear basis at this time for concluding that modifications to the plant are justified for the purpose of further mitigating severe accident risks.

In the longer term, these same severe accident issues are currently being pursued by the NRC in a systematic way for all utilities through the Severe Accident Program described in SECY-88-147, "Integration Plan for Closure of Severe Accident Issues" (Reference 7). The plan includes provisions for an Individual Plant Examination (IPE) for each operating reactor, a Containment Performance Improvement (CPI) program, and an Accident Management (AM) program. These programs will produce a more complete picture of the risks of operating plants and the benefits of potential design improvements, including SAMDAs. The staff believes that the severe accident program is the proper vehicle for further review of severe accidents at nuclear power plants, including Limerick.

For example, the Containment Performance Improvement (CPI) program is in the process of performing an integrated assessment of generic containment improvements for Mark II plants. The assessment entails a broad perspective of all Mark II plants, including their vulnerabilities and potential improvements. A set of SAMDAs is being considered which deals with the overall issue of containment performance and fission product control, using the most current understanding of source term behavior.

This supplement has made use of the risk insights and cost estimates from that program for the purpose of performing our screening assessment of SAMDAs. However, further work on SAMDAs for nuclear power plants including Limerick should continue within the CPI program. To do otherwise would duplicate effort, and would not result in a consistent resolution for Mark II plants.

In addition, many of the candidate SAMDAs (2., 5.B., 6., and 7.) fall into the category of Accident Management. The severe accident program is currently developing, in concert with the industry, an analytical "framework" which

utilities will use for the purpose of identifying and implementing accident management strategies. The identification process will include a balanced assessment of risk contributors, a systematic evaluation of candidate strategies, an evaluation of downsides and an assessment of plant specific problems associated with implementation. The implementation process will include consideration of instrumentation needs, training (including periodic exercises), consideration of decision making processes, and associated information requirements (such as computer codes to follow accident progression). The staff believes that accident management strategies should be implemented in an integrated fashion in the context of the NRC/industry framework.

Finally, the IPE, which consists of a full evaluation of the accident sequences which lead to core melt, will be performed by the licensee and reviewed by the staff. This process will produce an up-to-date picture of plant vulnerabilities for each plant individually, and will produce a pool of information concerning generically applicable insights. The IPE process is thus the most complete and efficient way of resolving the uncertainties discussed above associated with the core damage frequency for nuclear power plants including Limerick.

Most significantly, the three efforts described above (as well as several other related activities), will, as discussed in SECY-88-147, be brought to closure in an integrated fashion to assure a balanced resolution of severe accident issues.

## FOREWORD

In February 1989, the U. S. Court of Appeals for the Third Circuit ruled that the NRC failed to consider a "reasonable set" of Severe Accident Mitigation Design Alternatives (SAMDA) in the Final Environmental Statement (FES) for the Limerick Generating Station (NUREG-0974, April 1984). The NRC staff has completed consideration of a reasonable set of severe accident mitigation design alternatives. The staff has discovered no substantial changes in the proposed action as previously evaluated in the FES that are relevant to environmental concerns nor significant new circumstances or information relevant to environmental concerns and bearing on the licensing of Limerick Generating Station, Units 1 and 2.

Copies of this supplement are available for inspection at the NRC Public Document Room, 2120 L Street N.W., Washington, D.C. and at the Local Public Document Room at the Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Gene Y. Suh is the NRC Project Manager for the evaluation presented in this supplement. He may be contacted by telephone at (301) 492-1426 or by mail at the following address:

U.S. Nuclear Regulatory Commission  
Office of Nuclear Reactor Regulation  
Washington, DC 20555



## LIST OF CONTRIBUTORS

The following personnel of the U.S. Nuclear Regulatory Commission participated in the preparation of the supplement to the Final Environmental Statement:

- |                  |  |
|------------------|--|
| R. J. Barrett    | Chief, Risk Applications Branch; Ph.D. (Nuclear Physics) 1972; 17 years experience.                                |
| R. L. Palla, Jr. | Senior Reliability and Risk Analyst; M.S., B.S. (Mechanical Engineering) 1981, 1975; 13 years experience.          |
| E. S. Chelliah   | Reliability and Risk Analyst; M.S. (Nuclear Engineering) 1972, (Electrical Engineering) 1975; 15 years experience. |
| S. E. Feld       | Senior Industrial Economist; Ph.D. (Resource Economics) 1973; 16 years experience.                                 |
| W. B. Hardin     | Senior Reactor Systems Engineer; Ph.D., M.S. (Mechanical Engineering) 1987, 1965; 24 years experience.             |
| J. N. Ridgely    | Senior Reliability and Risk Analyst/Engineer; B.S. (Nuclear Science) 1972; 17 years experience.                    |
| G. Y. Suh        | Project Manager; M.S. (Mechanical Engineering) 1981; 8 years experience.   |

## ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ADS	automatic depressurization system
AM	Accident Management
ATWS	anticipated transient without scram
CDF	core damage frequency
CPI	Containment Performance Improvement Program
FES	Final Environmental Statement
IPE	Individual Plant Examination
MVSS	multi-venturi scrubber system
PECo	Philadelphia Electric Company
PRA	probabilistic risk assessment
RDA	R & D Associates
RWCU	reactor water cleanup system
SAMDA	severe accident mitigation design alternative
SARA	severe accident risk assessment

Supplement To NUREG-0974  
"Final Environmental Statement Related to the Operation of  
Limerick Generating Station, Units 1 and 2"

NRC Staff Evaluation of Severe Accident  
Mitigation Design Alternatives for Limerick

Summary and Conclusions

In February 1989, the U. S. Court of Appeals for the Third Circuit ruled that NRC failed to consider a "reasonable set" of Severe Accident Mitigation Design Alternatives (SAMDA) in the Final Environmental Statement (FES) for the Limerick Generating Station (NUREG-0974, April 1984). The NRC staff has completed consideration of a reasonable set of severe accident mitigation design alternatives (SAMDA). The staff has discovered no substantial changes in the proposed action as previously evaluated in the FES that are relevant to environmental concerns nor significant new circumstances or information relevant to environmental concerns and bearing on the licensing of Limerick Generating Station, Units 1 and 2.

In assessing the risk reduction potential, the value of each SAMDA was initially scoped based on risk information reported in the original Limerick Generating Station Severe Accident Risk Assessment (LGS-SARA, 1983) and reviewed by the staff in the 1983-1984 timeframe (NUREG-1068, August 1984). Modifications were made to this information base to account for the effect of two plant improvements identified in NUREG-1068 and subsequently implemented by PECO. The risk reduction scoping estimates were compared to the estimated costs associated with each SAMDA. Based on a screening criterion of \$1000 per averted person-rem, the comparison indicated that some candidate SAMDAs warranted further evaluation.

The staff then further evaluated each of the SAMDAs, considering the qualitative effect of several plant improvements made at Limerick since the time of the staff review of the LGS-SARA. Key plant improvements include the implementation of: procedures for battery power load shedding, MSIV air supply improvements, BWR Owners' Group Emergency Procedure Guidelines, Rev. 3 (and parts of Rev. 4), the hardened containment vent line, and procedures for the use of diesel-driven fire spray pumps for core injection. The staff also gave consideration to the results of a recent update to the Limerick PRA described in an April 25, 1989 ACRS subcommittee meeting, a June 23, 1989 utility submittal concerning SAMDAs, and a July 27, 1989 meeting with the staff concerning the SAMDA submittal. That study calculated values of CDF and offsite dose which were about four times lower than the staff's. While the staff has not reviewed these results in sufficient detail to confirm the quantitative results, the staff believes that these plant features would reduce the CDF and offsite doses. As a result, the averted offsite dose from candidate SAMDAs could be appreciably less than estimated by the staff.

The staff also considered uncertainty in the cost and effectiveness of candidate SAMDAs. For instance, the ATWS vent analyzed by the utility uses an existing 18 inch containment penetration which would be capable of removing 10 percent of full power. There are existing analyses which predict ATWS power levels as high as 30 percent for some scenarios. The staff identified operational disadvantages for some of the candidate SAMDAs (Table 4).

Of the seven SAMDAs which passed the screening cost/benefit test, the staff has identified two which have been implemented at Limerick. These are the Decay Heat Sized Vent Without Filter (3.C.) and the Low Pressure Reactor Makeup Capability (6.) The staff has not quantified the effectiveness of these SAMDAs in reducing risk. However, the staff believes that these features will result in an appreciable net decrease in CDF and risk.

In summary, the risks and environmental impacts of severe accidents at Limerick are acceptably low. We have found no new information that would call into question the FES conclusion that, "the risks of early fatality from potential accidents at the site are small in comparison with risks of early fatality from other human activities in a comparably sized population, and the accident risk will not add significantly to population exposure and cancer risks. Accident risks from Limerick are expected to be a small fraction of the risks the general public incurs from other sources. Further, the best estimate calculations show that the risks of potential reactor accidents at Limerick are within the range of such risks from other nuclear power plants," (NUREG-0974, Page 5-126).

Furthermore, while the screening cost/benefit analysis performed above indicates that several candidate SAMDAs might be cost effective, based on a criterion of \$1000 per person-rem averted a more recent utility PRA presents lower risk estimates which indicate that SAMDAs are not justified. While the staff has not verified the utility estimates, the staff is convinced that risk is now lower for Limerick than the estimates used in our cost/benefit study. Moreover, there are uncertainties about the costs, effectiveness, and/or operational disadvantages of some SAMDAs. In light of these considerations, the staff has no clear basis at this time for concluding that modifications to the plant are justified for the purpose of further mitigating severe accident risks.

In the longer term, these same severe accident issues are currently being pursued by the NRC in a systematic way for all utilities through the Severe Accident Program described in SECY-88-147, "Integration Plan for Closure of Severe Accident Issues" (Reference 7). The plan includes provisions for an Individual Plant Examination (IPE) for each operating reactor, a Containment Performance Improvement (CPI) program, and an Accident Management (AM) program. These programs will produce a more complete picture of the risks of operating plants and the benefits of potential design improvements, including SAMDAs. The staff believes that the severe accident program is the proper vehicle for further review of severe accidents at nuclear power plants, including Limerick.

For example, the Containment Performance Improvement (CPI) program is in the process of performing an integrated assessment of generic containment

improvements for Mark II plants. The assessment entails a broad perspective of all Mark II plants, including their vulnerabilities and potential improvements. A set of SAMDAs is being considered which deals with the overall issue of containment performance and fission product control, using the most current understanding of source term behavior.

This supplement has made use of the risk insights and cost estimates from that program for the purpose of performing our screening assessment of SAMDAs. However, further work on SAMDAs for nuclear power plants including Limerick should continue within the CPI program. To do otherwise would duplicate effort, and would not result in a consistent resolution for Mark II plants.

In addition, many of the candidate SAMDAs (2., 5.B., 6., and 7.) fall into the category of Accident Management. The severe accident program is currently developing, in concert with the industry, an analytical "framework" which utilities will use for the purpose of identifying and implementing an optimum set of accident management strategies. The identification process will include a balanced assessment of risk contributors, a systematic evaluation of candidate strategies, an evaluation of downsides and an assessment of plant specific problems associated with implementation. The implementation process will include consideration of instrumentation needs, training (including periodic exercises), consideration of decision making processes, and associated information requirements (such as computer codes to follow accident progression). The staff believes that accident management strategies should be implemented in an integrated fashion in the context of the NRC/industry framework.

Finally, the IPE, which consists of a full evaluation of the accident sequences which lead to core melt, will be performed by the licensee and reviewed by the staff. This process will produce an up-to-date picture of plant vulnerabilities for each plant individually, and will produce a pool of information concerning generically applicable insights. The IPE process is thus the most complete and efficient way of resolving the uncertainties discussed above associated with the core damage frequency for nuclear power plants including Limerick.

Most significantly, the three efforts described above (as well as several other related activities), will, as discussed in SECY-88-147, be brought to closure in an integrated fashion to assure a balanced resolution of severe accident issues.

#### Estimate of Risk for Limerick

An estimate of the core damage frequency associated with operation of Limerick was developed by the staff based on the review of the original Limerick Generating Station Severe Accident Risk Assessment (LGS-SARA, 1983) as documented in NUREG-1068 (1984). Since the staff review, Philadelphia Electric Company (PECo) has made numerous modifications to plant hardware and procedures. These are described by the utility in References 1-3. Two of the modifications identified made by PECo were in response to insights/recommendations identified in NUREG-1068. These involve improvements to Automatic Depressurization System (ADS) initiation logic following the potential loss of high pressure coolant sources, and improvements to achieve an alternate method of room cooling for

high pressure injection systems during loss of offsite power events. These improvements were estimated in NUREG-1068 to reduce core damage frequency from internal events by about a factor of 2.5 if implemented. The staff believes that PECO has satisfactorily implemented the plant improvements involving ADS logic and room cooling and accordingly has applied this reduction factor in establishing a baseline core damage frequency (CDF) and offsite dose estimate for Limerick. The original and modified values for CDF are presented in Table 1 by accident class. A description of the accident classes is also provided. These frequency estimates are for internally-initiated events and fire- and flood-initiated events, but do not include seismically-initiated events for reasons discussed in NUREG-1068, pages C 41-42. For comparison, the results of a recent (June 1989) update to the Limerick PRA are also provided in Table 1.

The Final Environmental Statement for Limerick, NUREG-0974, provides estimates of societal risks from severe accidents initiated by internal events and external events. These risk estimates were based on core damage frequency estimates, containment performance, source terms, and an offsite consequence analysis appropriate at that time. For purposes of evaluating SAMDAs, the staff requested its contractor, Brookhaven National Laboratory (BNL), to requantify the risk estimates to reflect the implementation of the two plant modifications identified in NUREG-1068 described above. The new risk estimates reflect only the changes in accident class frequencies. The containment performance, source terms, and offsite consequence analysis remain the same as given in the FES. The modified estimates are provided in Table 2 for selected risk measures, along with the values previously reported in NUREG-0974.

The risk associated with all significant containment failure modes considered for Limerick is provided in Table 3. This provides some insight into the risk reduction potential of SAMDAs which influence a particular containment challenge or failure mode. These insights were considered by the staff in developing a set of candidate SAMDAs, recognizing that the analyses in the risk assessment include many assumptions and uncertainties which can skew the results (NUREG-0974, pages 5-108 to 5-115).

In considering the risk estimates, it is important to note that the core damage frequency estimates on which the risk reduction estimates are based do not reflect many plant improvements made since the staff's review of the original Limerick PRA. Core damage frequency estimates from the licensee's current Limerick PRA would indicate that these improvements have reduced risk.

#### Development of a Set of SAMDAs

In order to develop a reasonable set of SAMDAs for consideration for Limerick, the staff reviewed the 1985 report of R&D Associates (Reference 4) and the more recent work performed in support of the Containment Performance Improvement Program. Based on this review, the staff assembled a set of candidate SAMDAs. Each SAMDA and its intended function is summarized briefly below. A qualitative assessment of the relative advantages and disadvantages of the SAMDAs is presented in Table 4.

1. Dedicated Suppression Pool Cooling

An independent dedicated system could be installed for transferring heat from the suppression pool to the spray pond. PECO evaluated this alternative assuming 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 would be cooled with water tapped off the spray pond suction line. This system can mitigate accident sequences where containment failure by overpressure occurs prior to core degradation for Class 2 sequences, such as in the TW sequence. Also some benefit may be obtained in Class 1 and 3 sequences if overtemperature failures can be avoided. It is not clear that an independent power system is needed to obtain the risk reduction associated with this SAMDA. Thus, the staff considered an alternative means of performing this function as SAMDA #2.

2. Alternate Means of Decay Heat Removal

Existing pumps, piping, and heat exchangers in the reactor water cleanup (RWCU) or other installed system may be used to remove decay heat energy. Use of the RWCU system could prevent core degradation, for Class 2 sequences, such as the TW sequence, where the reactor scrams and normal AC power is available. This means of heat removal has been identified and analyzed by the licensee of another Mark II plant and appears to be a viable alternative to containment venting. While the feasibility for Limerick has not been addressed by the staff, this option has been included here on the basis that it might prove feasible after further study.

3. Improved Venting Capability

Three cases were considered; these differed in terms of the system flow capacity (sized for ATWS versus decay heat power levels), and whether the system included a filter external to the containment.

A. ATWS-Sized Vent (without filter)

This SAMDA involves routing a large (3' to 5' diameter) hardened wetwell vent line to an elevated release point. The system would be passive and would operate without dependence on the station's present AC electrical power or other systems. A 70 psig rupture disk would be installed to minimize the likelihood of inadvertent opening. This vent could prevent containment failure, and thereby prevent core melt for accident sequences where the overpressurization is produced by Class 4 ATWSs.

B. Decay Heat-Sized Vent With Filter

This SAMDA involves routing a small hardened wetwell vent line to a filter located outside containment. The system would be capable of preventing containment overpressure for those sequences in which the steam generation rates are less than the system flow capacity, but would be ineffective for ATWS and containment bypass sequences. The system would operate without dependence on the station's present

support systems. The filter would be similar in design to the Multi-Venturi Scrubber System (MVSS) and would remove essentially all particulates. This system can mitigate the consequences of all slow to moderate overpressure containment failures.

C. Decay Heat-Sized Vent Without Filter

This SAMDA entails a small hardened wetwell vent line. The system would be capable of preventing containment over-pressure, thereby averting core damage, for those sequences in which the steam generation rates are less than the vent flow capacity, but would be ineffective for ATWS and containment bypass sequences. The system would be remote-manually operated from the main control room and would not be dependent on the station's present AC electrical power system. Releases would be scrubbed by the suppression pool provided the pool is not bypassed.

4. Core Debris Control

Core debris control involves, conceptually, a hardware modification that would serve to achieve a coolable debris bed and long-term decay heat removal. Two debris control systems were evaluated by PECO: a rubble bed device and a cooled dry crucible device. The rubble bed device consists of a floodable rubble bed in the lower pedestal pool area of the wetwell. The in-pedestal drywell floor would be modified with one foot diameter holes to allow the corium to flow onto the thoria plate cover the rubble bed in the lower pedestal area. A stainless steel liner would protect the pedestal concrete from excessive decomposition. The rubble bed would be kept dry until the corium had penetrated into the rubble bed, thus minimizing the potential for steam explosion. The cooled dry crucible device is a truncated 70 foot long cone which has a forced cooled water jacket to remove the decay heat. The cone starts at the basemat and extends under the current plant foundation. One foot diameter holes are drilled into the in-pedestal floor to allow the corium to flow into the cone. These designs may prevent overpressure drywell failure by limiting core-concrete interactions for Class 1 and 3 sequences, but would not prevent containment failure and subsequent core melt for Class 2 and 4 sequences. Given the expected disruption of existing structures and equipment due to installation of this SAMDA, it may not be a feasible option.

5. Drywell Overpressure/Overtemperature Protection

Two options that could help mitigate drywell failure were considered: an enhanced drywell spray system, and drywell head flooding.

A. Enhanced Drywell Spray System

An enhanced drywell spray system would recirculate suppression pool water through a heat exchanger and to the drywell sprays. PECO modelled this option as an extension to the dedicated suppression pool cooling system, discussed in Item 1 above. However, we have used cost estimates consistent with a simpler design discussed in



Reference 5. The suppression pool cooling system would prevent containment overpressure failure and core melt for Class 2 sequences. Operation of sprays will cool the drywell atmosphere and the core debris during Class 1 and 3 accidents and minimize the threat from overtemperature. However, unless the sprays terminate core-concrete interactions, the non-condensibles released from the concrete will still cause the containment to eventually fail by overpressure. In either case, the sprays would reduce the airborne fission product concentration and thus, lower the source term.

B. Drywell Head Flooding

Intentional post-accident flooding of the area above the drywell head would cool the drywell head seal and provide fission product scrubbing in the event of drywell leakage. In Limerick, this area is serviced by the standby gas treatment system (SGTS) which is normally plugged with a blind flange during refueling. To implement this SAMDA this flange must be left in place during normal plant operation. It is expected that flooding of this area must be initiated early in the accident scenario and would prevent the over-temperature failure of the drywell head flange seals.

6. Makeup to Reactor Using Low Pressure Diesel-Driven Pump

The diesel-driven low pressure reactor makeup water pump would be an existing or new pump(s) which can provide sufficient flow to the reactor vessel when the reactor is at low pressure. If there has been no core degradation, core melt could be prevented. If core melt has commenced, this flow would prevent additional fuel degradation for the intact portion of the core and may prevent or delay bottom head failure from the corium on the bottom head. This does not reduce the risk for ATWS sequences.

7. Enhanced Reactor Depressurization Capability

This SAMDA involves enhancement of the existing reactor depressurization capability to provide additional backup power (and nitrogen if needed) to operate the safety relief valves (SRVs), either individually or as part of the manually initiated automatic depressurization system (ADS). Depressurizing the reactor would permit low pressure injection, and would convert high pressure melt ejection sequences to low pressure sequences, thereby reducing the potential for early containment failure. This SAMDA was evaluated assuming it would be implemented in combination with other SAMDAs, as discussed below.

A. In Conjunction with Decay Heat-Sized Hardened Filtered Venting (Item 3.B)

If core debris is ejected from the reactor vessel under pressure, then it is possible to fail containment during the blowdown and bypass the filtered vent. With the reactor depressurized, the challenges to containment from early over-pressure are significantly reduced, thereby increasing the effectiveness of the filtered vent.

B. In Conjunction with Core Debris Control (Item 4)

Unless the core debris control device includes some means of collecting or diverting the debris into the device, it would not be effective for accident: in which the reactor fails at high pressure. Reactor depressurization would increase the effectiveness of the core debris control device by assuring that debris is released into the device.

C. In Conjunction with Enhanced Drywell Sprays (Item 5.A)

With the reactor depressurized, the corium would tend to exit the reactor vessel in a more coherent mass and the time to containment failure would be delayed. This would increase the effectiveness of the sprays in scrubbing the aerosols and cooling the debris.

D. In Conjunction with Drywell Head Flooding (Item 5.B)

With the reactor depressurized, early containment challenges would be reduced and the time to containment failure would be delayed. This would increase the likelihood of drywell head failure/leakage as a containment failure mode, and would enhance the risk reduction potential of drywell head flooding.

E. In conjunction with reactor vessel makeup (Item 6)

Reactor depressurization would permit the use of the diesel-driven pump(s) discussed in Item 6 for injection into the reactor. This would prevent core damage for some sequences that otherwise would lead to core melt and reactor vessel failure at high pressure.

8. Reactor Building Decontamination Factor Improvement

This SAMDA involves modifications to the fire protection and/or standby gas treatment system hardware/procedures to enhance the fission product removal capabilities of the reactor building. The fire protection system consists of diesel and motor driven pumps which discharge into compartments or areas of the plant. Some of the plant areas have complete spray coverage, other areas have partial or no spray coverage. The plant would be retrofitted to have complete spray coverage. The capacity of the fire pumps would need to be increased (either by capacity or number of pumps) to ensure continuous spraying of the entire reactor building. Such a capability would provide scrubbing of fission products, given that containment fails.

The risk reduction potential of each of these candidate SAMDAs was estimated by the staff as described below. An additional SAMDA analyzed by R&D Associates in Reference 4 is Vacuum Breaker enhancements. The staff did not give further consideration to this system because our assessment is that it does not contribute appreciably to the reduction of risk. Similarly, the staff did not give further consideration to the hydrogen recombiner SAMDA, because the Limerick containment atmosphere is inerted as a defense against hydrogen burns.

### Risk Reduction Potential of Candidate SAMDAs

In assessing the risk reduction potential, the value of each SAMDA was initially scoped based on the core damage frequency estimates reported in NUREG-1068, modified to reflect the improvements to ADS initiation logic and improvements to room cooling discussed therein. The modified core damage frequency estimates are reported in Table 1. The corresponding risk estimates (person-rem per reactor-year) within 50 miles of the plant for each containment failure mode are listed in Table 3. As noted above, these risk reduction estimates do not account for some features which have been added to the Limerick plant since completion of the LGS-SARA study.

Estimates of the risk reduction potential of each SAMDA were developed in consultation with the staff's contractor, Brookhaven National Laboratory (BNL). The estimated reductions, in terms of person-rem and early fatalities per reactor year are presented in Table 5. Details of the assessment for each SAMDA are presented in Appendix A.

### Cost Impacts of Limerick Severe Accident Mitigation Design Alternatives

The cost impacts of the various SAMDA mitigation systems have been investigated by the staff. To fully integrate any one of these proposed systems into the Limerick Station, costs on the order of millions to tens of millions of dollars are likely to be incurred.

Relatively large costs are to be anticipated whenever physical modifications are imposed on operating or existing nuclear power reactors. This is because labor productivity is severely constrained due to problems with congestion, access, and security requirements. Also, retrofits on existing power reactors frequently require the removal and/or replacement of existing systems due to access considerations or the new system's interdependency with existing equipment and control panels. In addition, the introduction of a new system will trigger a whole series of related requirements such as incremental training, procedural changes, and licensing requirements. Finally, the retrofit could impose significant replacement energy cost penalties on the licensee and its customers if it results in incremental downtime or if it postponed the date of initial full power operation for Unit 2. These are all legitimate costs that require consideration in a comprehensive cost estimate.

Cost analyses for most of the modifications under consideration have been developed elsewhere.<sup>4,5</sup> The approach taken by the staff was to evaluate these estimates in order to arrive at a representative cost for each mitigation system. It should be recognized that only gross approximations of the costs of specific mitigation systems are possible at this time. Large uncertainties exist because detailed designs are not available and there is limited experience with construction and licensing problems that could surface with this type of work. Nevertheless, the staff views the results of this review as adequate given the uncertainties surrounding these underlying cost estimates, and the level of precision necessary given the greater uncertainty inherent on the benefit side, with which these impacts were compared.

Table 6 depicts the cost estimates available from R & D Associates (RDA),<sup>4</sup> and Bechtel Power Corp.<sup>5</sup> whose report was prepared for the Philadelphia Electric Company (PECo). It should be noted that RDA's report provides cost results on a component basis and in several instances the staff has summed the component costs to produce systems comparable with those costed by PECo (Bechtel report).

Where aggregation of this nature occurs, it is noted in Table 6. Also, the RDA report provides different cost estimates based on reactor status (A - reactor in design stage, B - reactor under construction, and C - operating reactor). Cost estimates for operating reactors (case C), were judged most consistent with the current status of the Limerick Station and are adopted in Table 6. When comparable systems are costed by PECo and RDA, PECo's estimates are consistently higher, in most instances by an order of magnitude. Smaller cost differences are observed for the ATWS vent option (factor of 2), and for the gravel bed venting and filtering system (factor of 4).

The final column of Table 6 contains the staff's estimate for each mitigation system. These costs reflect decrements and increments to the PECo and RDA estimates based on a critical assessment of the assumptions embedded in their analyses and the staff's technical judgement. A general discussion of the cost elements contributing to the staff's cost estimates is provided in Appendix B.

#### Cost/Benefit Comparison for Candidate SAMDAs

A comparison of the estimated costs and benefits of the various SAMDAs is presented in Table 7. For those SAMDAs that were not addressed by the licensee, the costs estimates developed as part of the NRC Containment Performance Improvements (CPI) program were used where available.<sup>6</sup> The risk reduction potential for each option is based on the estimates given in Table 5. The averted offsite dose (person-rem per reactor year) was used as a surrogate measure of risk and environmental impact. A screening criterion of \$1000 per person rem averted was used to identify SAMDA's which warrant further evaluation.

Based on this screening analysis, a set of seven potential SAMDAs was identified for more detailed evaluation. These included:

- Alternate Means of Decay Heat Removal (Options 2. and 3.C.)
- ATWS-Sized Vent
- Enhanced Reactor Coolant System Depressurization
- Enhanced Drywell Sprays
- MVSS Filtered Containment Vent
- Low Pressure Makeup to Reactor
- Drywell Head Flooding

#### Evaluation

For the seven candidate SAMDAs which passed the cost/benefit screening, the staff performed a further evaluation. The evaluation accounted for a number of factors which were not considered in the screening analysis. These included: plant improvements made since the publication of NUREG-1068 which were not

considered in the staff's estimates of CDF; SAMDAs which exist in the plant which were not credited in the screening analysis; uncertainties in the cost and effectiveness of candidate SAMDAs; and potential operational disadvantages of SAMDAs.

1. Alternate Means of Decay Heat Removal

Given the cost/benefit analysis performed above, this option appears to have significant potential for risk reduction by lowering the core damage frequency due to loss-of-containment-heat-removal sequences (TW). However, a feature which is already installed in the plant, containment venting, appears to be a viable means for achieving this function. The staff has performed a preliminary assessment of the hardware and procedures associated with this capability. It is the staff's judgment that the use of the existing system and procedures could be a viable option for reducing the frequency of TW sequences, especially given the slow moving nature of these sequences (20-30 hours to core melt). The efficacy of this system and potential operational disadvantages have not been reviewed by the staff. Accordingly, the benefit that an additional heat removal system might provide would be minimal.

2. ATWS-Sized Vent

In Class IV ATWS sequences core melt occurs as a result of containment failure. The ATWS vent is intended to reduce risk by preventing containment failure thereby lowering the ATWS core damage frequency. As shown in Table 7, this system not only passes the screening analysis based on averted offsite dose, but it could also reduce the principal source of early fatalities. This is the only candidate SAMDA which substantially reduces early fatalities.

A closer look at this system, however, raises questions about its effectiveness. First, a large fraction of the risk reduction attributed to this option in Table 5 is from Class II (TW) sequences. As noted above, the staff believes that the existing containment vent appears capable of effectively dealing with this class of sequences. Thus, the risk reduction benefit of the ATWS vent would be confined to Class IV ATWS sequences (an averted risk of 18 rather than 88 person rem per reactor year). The licensee estimates an averted risk of 27 person rem per reactor year.

An additional source of uncertainty is the basis for the utility's proposal to use an existing 18 inch purge line penetration, based on the assumption that ATWS power would be 10% of full power. Depending on the circumstances of the event, and the assumptions used in the analysis, some existing studies predict ATWS power to be considerably higher than 10 percent. This would require a new large containment penetration and would, therefore, considerably increase the cost of this SAMDA.

### 3. Enhanced Reactor Containment System Depressurization

Class I and Class II sequences consist of transients (and ATWS) in which the core melts with the containment intact. The radiological consequences of those sequences can be mitigated significantly if early containment failure can be avoided. For instance, a delay of several hours in the time to containment failure can result in a significant reduction of the fission product inventory in the containment atmosphere, as a result of natural processes such as aerosol deposition and operation of active systems such as drywell sprays.

An important uncertainty about early containment failure for Limerick is the possibility of vessel failure at high pressure due to unavailability of the Automatic Depressurization System (ADS). Despite the ADS improvements at Limerick since publication of the LGS-SARA study, the risk estimates used for the screening analysis indicate a high likelihood of reactor pressure vessel failure at high pressure versus low pressure.

A more recent assessment of core damage frequency performed by the licensee concludes that (1) the overall frequency of Class I and Class II sequences is considerably lower than the staff estimates and (2) the fraction of high pressure sequences is much lower than indicated in the FES. If this conclusion is correct, further improvements to assure reactor depressurization would have a minor impact on risk reduction. The staff has not reviewed the licensee analysis in sufficient detail to verify these quantitative estimates.

### 4. Enhanced Drywell Sprays

Drywell sprays can be effective in delaying containment failure and reducing the radiological releases for Class I and Class III sequences in which the containment does not fail early. In combination with the depressurization of the RCS, enhancements to containment sprays appear to have considerable risk reduction potential (Table 5) and pass the screening analysis (Table 7). However, the perceived risk reduction benefits from enhanced sprays result from mitigation of Class I and Class III sequences. As noted above, the licensee's estimates of risk from Class I and Class III sequences are considerably lower than those used by the staff in our screening analysis.

### 5. Filtered Containment Vent

The MVSS filtered vent appears to have significant potential for risk reduction (Table 5) for Class I and Class III sequences and warrants further evaluation based on cost/benefit ratio (Table 7). However, as noted above, the licensee's estimates of Class I CDF are considerably lower than the staff's. Furthermore, if the existing containment vent is effective in mitigating Class II sequences, the perceived benefit of MVSS would be further reduced.

#### 6. Low Pressure Makeup To Reactor

This SAMDA appears to have risk reduction potential for those Class I accident sequences in which core melt would result from a failure of low pressure injection.

There is a significant potential disadvantage of this type of SAMDA. If the piping and hardware associated with this system is not designed to withstand reactor system pressure, the possibility exists of creating a LOCA outside of containment in the event that the RCS returned to high pressure after the SAMDA was connected.

The staff is aware that Limerick has already implemented a SAMDA of this type, using the existing diesel-driven fire suppression pump and piping for injection into the RWCU. The staff has not reviewed this existing capability in detail.

#### 7. Drywell Head Flooding

Examination of the table of costs, benefits and cost-effectiveness ratios for Limerick indicates support for this SAMDA option. However, the scoping analysis needs further refinement in order to be in a better position to determine whether this option is worthwhile. The potential benefit envisioned for this SAMDA is directed toward reducing the risks from Class I and III accidents. The averted offsite risk estimated for this option in table 7 is approximately 50 person-rem. The utility has performed an analysis with substantially lower core damage frequency and risk-reduction benefits based on recent modifications made to the plant. Although the staff has not verified the quantitative risk estimates it is reasonable to expect that the plant modifications would reduce offsite risk. Also, cost estimates are very uncertain due to unavailability of detailed design information on modifying the drywell head configuration and on corresponding cost estimates. Furthermore, this SAMDA does not appear to preclude the possibility of other failures during accident progression that would lead to source terms for radioactivity released to the environment equivalent to those from the unmitigated case.

#### Summary and Conclusions

The NRC staff has completed consideration of a reasonable set of severe accident mitigation design alternatives (SAMDA's). The staff has discovered no substantial changes in the proposed action as previously evaluated in the FES that are relevant to environmental concerns nor significant new circumstances or information relevant to environmental concerns and bearing on the licensing of Limerick Generating Station, Units 1 and 2.

In assessing the risk reduction potential, the value of each SAMDA was initially scoped based on risk information reported in the original Limerick Generating Station Severe Accident Risk Assessment (LGS-SARA, 1983) and reviewed by the staff in the 1983-1984 timeframe (NUREG-1068, August 1984).

Modifications were made to this information base to account for the effect of two plant improvements identified in NUREG-1068 and subsequently implemented by PECO. The risk reduction scoping estimates were compared to the estimated costs associated with each SAMDA. Based on a screening criterion of \$1000 per averted person-rem, the comparison indicated that some candidate SAMDAs warranted further evaluation.

The staff then further evaluated each of the SAMDAs, considering the qualitative effect of several plant improvements made at Limerick since the time of the staff review of the LGS-SARA. Key plant improvements include the implementation of: procedures for battery power load shedding, MSIV air supply improvements, BWR Owners' Group Emergency Procedure Guidelines, Rev. 3 (and parts of Rev. 4), the hardened containment vent line, and procedures for the use of diesel-driven fire spray pumps for core injection. The staff also gave consideration to the results of a recent update to the Limerick PRA described in an April 25, 1989 ACRS subcommittee meeting, a June 23, 1989 utility submittal concerning SAMDAs, and a July 27, 1989 meeting with the staff concerning the SAMDA submittal. That study calculated values of CDF and offsite dose which were about four times lower than the staff's. While the staff has not reviewed these results in sufficient detail to confirm the quantitative results, the staff believes that these plant features would reduce the CDF and offsite doses. As a result, the averted offsite dose from candidate SAMDAs could be appreciably less than estimated by the staff.

The staff also considered uncertainty in the cost and effectiveness of candidate SAMDAs. For instance, the ATWS vent analyzed by the utility uses an existing 18 inch containment penetration which would be capable of removing 10 percent of full power. There are existing analyses which predict ATWS power levels as high as 30 percent for some scenarios. The staff identified operational disadvantages for some of the candidate SAMDAs (Table 4).

Of the seven SAMDAs which passed the screening cost/benefit test, the staff has identified two which have been implemented at Limerick. These are the Decay Heat Sized Vent Without Filter (3.C.) and the Low Pressure Reactor Makeup Capability (6.) The staff has not quantified the effectiveness of these SAMDAs in reducing risk. However, the staff believes that these features will result in an appreciable net decrease in CDF and risk.

In summary, the risks and environmental impacts of severe accidents at Limerick are acceptably low. We have found no new information that would call into question the FES conclusion that, "the risks of early fatality from potential accidents at the site are small in comparison with risks of early fatality from other human activities in a comparably sized population, and the accident risk will not add significantly to population exposure and cancer risks. Accident risks from Limerick are expected to be a small fraction of the risks the general public incurs from other sources. Further, the best estimate calculations show that the risks of potential reactor accidents at Limerick are within the range of such risks from other nuclear power plants," (NUREG-0974, Page 5-126).



Furthermore, while the screening cost/benefit analysis performed above indicates that several candidate SAMDAs might be cost effective, based on a criterion of \$1000 per person-rem averted a more recent utility PRA presents lower risk estimates which indicate that SAMDAs are not justified. While the staff has not verified the utility estimates, the staff is convinced that risk is now lower for Limerick than the estimates used in our cost/benefit study. Moreover, there are uncertainties about the costs, effectiveness, and/or operational disadvantages of some SAMDAs. In light of these considerations, the staff has no clear basis at this time for concluding that modifications to the plant are justified for the purpose of further mitigating severe accident risks.

In the longer term, these same severe accident issues are currently being pursued by the NRC in a systematic way for all utilities through the Severe Accident Program described in SECY-88-147, "Integration Plan for Closure of Severe Accident Issues" (Reference 7). The plan includes provisions for an Individual Plant Examination (IPE) for each operating reactor, a Containment Performance Improvement (CPI) program, and an Accident Management (AM) program. These programs will produce a more complete picture of the risks of operating plants and the benefits of potential design improvements, including SAMDAs. The staff believes that the severe accident program is the proper vehicle for further review of severe accidents at nuclear power plants, including Limerick. For example, the Containment Performance Improvement (CPI) program is in the process of performing an integrated assessment of generic containment improvements for Mark II plants. The assessment entails a broad perspective of all Mark II plants, including their vulnerabilities and potential improvements. A set of SAMDAs is being considered which deals with the overall issue of containment performance and fission product control, using the most current understanding of source term behavior.

This supplement has made use of the risk insights and cost estimates from that program for the purpose of performing our screening assessment of SAMDAs. However, further work on SAMDAs for nuclear power plants including Limerick should continue within the CPI program. To do otherwise would duplicate effort, and would not result in a consistent resolution for Mark II plants.

In addition, many of the candidate SAMDAs (2., 5.B., 6., and 7.) fall into the category of Accident Management. The severe accident program is currently developing, in concert with the industry, an analytical "framework" which utilities will use for the purpose of identifying and implementing an optimum set of accident management strategies. The identification process will include a balanced assessment of risk contributors, a systematic evaluation of candidate strategies, an evaluation of downsides and an assessment of plant specific problems associated with implementation. The implementation process will include consideration of instrumentation needs, training (including periodic exercises), consideration of decision making processes, and associated information requirements (such as computer codes to follow accident progression). The staff believes that accident management strategies should be implemented in an integrated fashion in the context of the NRC/industry framework.

Finally, the IPE, which consists of a full evaluation of the accident sequences which lead to core melt, will be performed by the licensee and reviewed by the staff. This process will produce an up-to-date picture of plant vulnerabilities for each plant individually, and will produce a pool of information concerning generically applicable insights. The IPE process is thus the most complete and efficient way of resolving the uncertainties discussed above associated with the core damage frequency for nuclear power plants including Limerick.

Most significantly, the three efforts described above (as well as several other related activities), will, as discussed in SECY-88-147, be brought to closure in an integrated fashion to assure a balanced resolution of severe accident issues.

References

1. Presentation by Philadelphia Electric Company to the Advisory Committee on Reactor Safety in the Matter of Limerick 2 Operating License, April 25, 1989.
2. Letter from G. A. Hunger, Philadelphia Electric Company to the NRC, Subject: Limerick Generating Station, Units 1 and 2, Response to Request for Additional Information Regarding Consideration of Severe Accident Mitigation Design Alternatives, June 23, 1989.
3. Presentation by Philadelphia Electric Company to the NRC Staff, July 27, 1989.
4. NUREG/CR-4025, "Design and Feasibility of Accident Mitigation Systems for Light Water Reactors," R & D Associates, August 1985.
5. "Cost Estimate for Severe Accident Mitigation Design Alternatives -- Limerick Generating Station for Philadelphia Electric Company," Bechtel Power Corporation, June 22, 1989.
6. "A Preliminary Assessment of BWR Mark II Containment Challenges, Failure Modes, and Potential Improvements," Draft NRC Report, August 4, 1989.
7. SECY-88-147, "Integration Plan for Closure of Severe Accident Issue," May 25, 1988.

TABLE 1 - ESTIMATES OF CORE DAMAGE FREQUENCY FOR LIMERICK  
(EXCLUDING SEISMICALLY-INITIATED EVENTS)

ACCIDENT CLASS <sup>1</sup>	FREQUENCY (PER REACTOR-YEAR)		
	<u>ORIGINAL (NUREG-1068)</u>	<u>MODIFIED<sup>2</sup></u>	<u>June 1989 PRA UPDATE</u>
I	8.0 E-5	3.4 E-5	8.8 E-6
II	4.1 E-6	4.1 E-6	1.7 E-7
III	3.3 E-6	3.3 E-6	2.7 E-7
IV	3.2 E-7	3.2 E-7	1.1 E-6
S	<u>2.7 E-8</u>	<u>2.7 E-8</u>	<u>1.0 E-8</u>
TOTAL	8.8E-5	4.2 E-5	1.0 E-5

<sup>1</sup> Accident Class Definitions

- CLASS 1 (or I) Transients or LOCAs involving loss of coolant makeup to the core. Core melts in an intact containment.
- CLASS 2 (or II) Transient or LOCA involving loss of long term heat removal. Long term core melts in a failed or open containment.
- CLASS 3 (or III) Transients with failure to scram with failure of all injection. Rapid core melt in an intact containment.
- CLASS 4 (or IV) Transient with failure to scram and failure to shutdown. Rapid core melt in a failed or open containment.
- CLASS 5 Core melt due to reactor pressure vessel failure with early containment failure.

<sup>2</sup> Modified to reflect ADS and room cooling enhancements identified in NUREG-1068.

TABLE 2 - RISK ESTIMATES FOR LIMERICK UNIT 2  
(EXCLUDING SEISMICALLY-INITIATED EVENTS)

ESTIMATED RISK WITHIN ENTIRE REGION,  
PER REACTOR YEAR

<u>CONSEQUENCE TYPE</u>	<u>FES (EXCLUDING SEISMIC)</u>	<u>MODIFIED<sup>1</sup> STAFF ESTIMATES</u>
Early fatalities with supportive medical treatment (persons)	2(-4)	1.9(-4)
Latent Cancer fatalities (excluding thyroid) (persons)	5(-2)	3.2(-2)
Total person-rems	1(3)	5.4(2)
Land area for long-term interdiction (m <sup>2</sup> )	N/A <sup>2</sup>	6.3(2)

<sup>1</sup> Based on modified accident class frequencies in Table 1 (excludes seismically-initiated events).

<sup>2</sup> Not Available

TABLE 3 - CONTRIBUTION TO RISK BY CONTAINMENT FAILURE MODE

<u>CONTAINMENT FAILURE MODE</u>	<u>ESTIMATED RISK (PERSON-REM/REACTOR-YEAR)<sup>1</sup></u>	
	<u>Entire Region</u>	<u>50 Mile Region</u>
Overpressure due to failure of decay heat removal - core melts into failed containment (Class II)	114	80
Overpressure due to ATWS - core melts into failed containment (Class IV)	25	18
Transient leads to core melt followed by drywell failure (Class I and III)	129	90
Transient leads to core melt followed by wetwell failure (Class I and III)	46	32
Transient leads to core melt - containment leakage exceeds standby gas treatment system capacity (Class I and III)	198	139
<u>Other</u>	<u>15</u>	<u>11</u>
TOTAL	527	370

<sup>1</sup> Based on modified accident class frequencies in Table 1.

TABLE 4 QUALITATIVE ASSESSMENT OF THE RELATIVE ADVANTAGES AND DISADVANTAGES OF SAMDAS

<u>Potential Improvement</u>	<u>Advantages</u>	<u>Disadvantages</u>
1. Dedicated Suppression Pool Cooling	<ul style="list-style-type: none"><li>◦ Helps to maintain suppression pool subcooled</li><li>◦ Reduces overpressure challenge from Class II sequences</li><li>◦ Reduces pressurization rate for ATWS</li></ul>	<ul style="list-style-type: none"><li>◦ Very expensive</li></ul>
2. Alternate Means of Decay Heat Removal (e.g., use of RWCU system)	<ul style="list-style-type: none"><li>◦ Helps to maintain pool subcooled</li><li>◦ Reduces overpressure challenge for Class III sequences</li><li>◦ Reduces pressurization rate from ATWS</li><li>◦ Less expensive than dedicated pool cooling system</li></ul>	<ul style="list-style-type: none"><li>◦ Less reliable than dedicated system due to reliance on shared components</li></ul>
3A. ATWS-Sized Vent	<ul style="list-style-type: none"><li>◦ Reduces overpressure failures for ATWS and Class II sequences</li><li>◦ Preemptive venting reduces base pressure prior to core damage</li></ul>	<ul style="list-style-type: none"><li>◦ Suppression pool bypass would result in unscrubbed release</li><li>◦ Can lead to inadvertent releases</li></ul>
3B. Decay Heat-Sized Vent with Filter	<ul style="list-style-type: none"><li>◦ Reduces overpressure failures for transients with scram</li><li>◦ Delays ATWS</li><li>◦ Preemptive venting reduces base pressure prior to core damage</li><li>◦ Helps to assure all releases will be scrubbed</li><li>◦ Unaffected by suppression pool bypass</li></ul>	<ul style="list-style-type: none"><li>◦ Can lead to inadvertent releases of noble gases</li></ul>

<u>Potential Improvement</u>	<u>Advantages</u>	<u>Disadvantages</u>
3C. Decay Heat-Sized Vent without Filter	<ul style="list-style-type: none"><li>◦ Reduces overpressure failures for transients with scram</li><li>◦ Delays ATWS</li><li>◦ Preemptive venting reduces base pressure prior to core damage</li><li>◦ Less expensive than filtered vent</li></ul>	<ul style="list-style-type: none"><li>◦ Suppression pool bypass would result in unscrubbed release</li><li>◦ Can lead to inadvertent releases</li></ul>
4. Core Debris Control (Conceptual)	<ul style="list-style-type: none"><li>◦ Helps to maintain core debris coolable</li><li>◦ Helps to eliminate containment challenges following reactor vessel failure</li></ul>	<ul style="list-style-type: none"><li>◦ May not be effective if reactor pressure vessel fails at high pressure</li><li>◦ Very expensive</li></ul>
Adding in-pedestal downcomers and debris barrier	<ul style="list-style-type: none"><li>◦ Increases likelihood of quenching the core ex-vessel</li><li>◦ Reduces importance of containment sprays and venting</li></ul>	<ul style="list-style-type: none"><li>◦ Increases the likelihood of steam explosion/spikes</li><li>◦ Increases the probability of suppression pool bypass</li><li>◦ Requires re-analysis of containment pressure suppression capability and seismic design</li><li>◦ Expensive</li></ul>
Strengthening ex-pedestal downcomers	<ul style="list-style-type: none"><li>◦ Decreases the probability of suppression pool bypass</li></ul>	<ul style="list-style-type: none"><li>◦ Does not reduce erosion of the drywell floor</li><li>◦ Requires re-analysis of containment pressure suppression capability and seismic design</li><li>◦ Expensive</li></ul>
5A. Enhanced Drywell Spray System	<ul style="list-style-type: none"><li>◦ Reduces containment overpressure from condensibles</li><li>◦ Reduces drywell over-temperature failure</li><li>◦ Scrubbing of fission products</li><li>◦ Reduce core-concrete interactions</li></ul>	<ul style="list-style-type: none"><li>◦ None identified</li></ul>



<u>Potential Improvement</u>	<u>Advantages</u>	<u>Disadvantages</u>
5B. Drywell Head Flooding	<ul style="list-style-type: none"><li>◦ Mitigates drywell head seal overtemperature failure</li><li>◦ Drywell head leakage would be scrubbed by overlaying water pool</li></ul>	<ul style="list-style-type: none"><li>◦ Must be initiated early in the accident</li></ul>
6. Makeup to Reactor Using Low Pressure Diesel-Driven Pump	<ul style="list-style-type: none"><li>◦ Helps to prevent core melt in low pressure transients with scram</li><li>◦ Some cooling and scrubbing of ex-vessel debris</li><li>◦ Independent of RHR</li><li>◦ Relatively low cost, if fire system pumps are used</li></ul>	<ul style="list-style-type: none"><li>◦ Requires reactor at low pressure for injection</li><li>◦ Potential conflict for concurrent fire, if fire system used</li><li>◦ Requires many operator actions</li></ul>
7. Enhanced Reactor Depressurization Capability	<ul style="list-style-type: none"><li>◦ Can prevent high pressure core melt transients</li><li>◦ Reduces containment challenges from high pressure melt ejection</li><li>◦ Relatively low cost</li></ul>	<ul style="list-style-type: none"><li>◦ None identified</li></ul>
8. Reactor Building Decontamination Factor Improvement	<ul style="list-style-type: none"><li>◦ Scrubbing of fission products</li><li>◦ Much of the hardware already in place</li></ul>	<ul style="list-style-type: none"><li>◦ Existing hardware provide limited spray coverage</li><li>◦ May provide a greater benefit as an alternate containment spray or RPV injection system</li><li>◦ Increased probability of hydrogen fumes</li></ul>

TABLE 5 STAFF ESTIMATES OF RISK REDUCTION BENEFITS FOR SANDAS BASED ON MODIFIED NUREG-1068 ACCIDENT FREQUENCIES

MITIGATION FEATURE	EVALUATED BY PECO?	REDUCTION IN PERSON-REM/R <sup>1</sup>				TOTAL	REDUCTION IN EARLY FATALITIES/R <sup>1</sup>
		CLASS I	CLASS II	CLASS III	CLASS IV		
1. DEDICATED SUPPRESSION POOL COOLING	YES	0/193 <sup>1</sup>	80/80	0/79	0/18	80/370	
2. ALTERNATE MEANS OF DECAY HEAT REMOVAL (e.g. RMCU SYSTEM)	NO	0	80	0	0	80	
3. IMPROVED VENTING CAPABILITY (HARDENED, PASSIVE)							
A. ATWS-SIZED VENT	YES	0	70	0	18	88	2 x 10 <sup>-4</sup>
B. DECAY HEAT-SIZED VENT W/FILTER	YES	106	70	39	0	215	
C. DECAY HEAT-SIZED VENT W/O FILTER	NO	0	70	0	0	70	
4. CORE DEBRIS CONTROL	YES	20	0	0	0	20	
5. DRYWELL OVERPRESSURE/OVER-TEMPERATURE PROTECTION							
A. DRYWELL SPRAYS	YES	71	80	27	0	178	
B. DRYWELL HEAD FLOODING	NO	46	0	4	0	50	
6. MAKEUP TO REACTOR USING LOW PRESSURE DIESEL-DRIVEN PUMP	NO	20	80 <sup>3</sup>	0	0	100	

<sup>1</sup> Numbers to the right of the "slash" represent the current value of offsite dose for that accident class, before installation of SANDAS.

MITIGATION FEATURE	EVALUATED BY PECO?	REDUCTION IN PERSON-REM/R <sup>1</sup>				TOTAL	REDUCTION IN EARLY FATALITIES/R <sup>1</sup>
		CLASS I	CLASS II	CLASS III	CLASS IV		
7. ENHANCED REACTOR DEPRESSURIZATION CAPABILITY	NO						
A. IN CONJUNCTION WITH 3B		193	70	39	0	302	Footnote 2
B. IN CONJUNCTION WITH 4		193	0	0	0	193	Footnote 2
C. IN CONJUNCTION WITH 5A		129	80	27	0	236	Footnote 2
D. IN CONJUNCTION WITH 5B		84	0	4	0	88	Footnote 2
E. IN CONJUNCTION WITH 6		193	80	3	0	273	Footnote 2
8. REACTOR BUILDING DEF IMPROVEMENT	NO	46	0	4	0	50	

<sup>1</sup> [VALUE OF REDUCTION]/[TOTAL FOR CLASS], STAFF ESTIMATES BASED ON LGS-SARA (1983) MODIFIED TO REFLECT ADS AND ROOM COOLING ENHANCEMENTS IDENTIFIED IN NUREG-1068. VALUES PRESENTED ARE FOR THE 50-MILE REGION.

<sup>2</sup> THE UPPER BOUND IN NUREG-1068, WHICH ASSUMED CONTAINMENT FAILURE AT VESSEL BREACH FOR CLASS I EVENTS, WILL BE REDUCED.

<sup>3</sup> ASSUMES THAT CORE INJECTION CAN BE MAINTAINED AFTER CONTAINMENT FAILURE.

TABLE 6 PER REACTOR COSTS FOR SAMDAS\*  
(Millions of 1990 Dollars)

	RDA	PECo	NRC <sup>a</sup>
I. <u>DEDICATED SUPPRESSION POOL COOLING (SAMDA 1.)</u>			
I.1 Dedicated Suppression Pool Cooling		25.6	20.9
I.2 Dedicated Surface Sited Heat Removal System	2.8		19.4
I.3 Dedicated Underground Heat Removal System	2.5		19.0
II. <u>DRYWELL SPRAY (SAMDA 5.A.)</u>			
II.1 Enhanced Drywell Spray System (new spray headers) <sup>b</sup>		46.5	37.3
II.2 Enhanced Drywell Spray System (existing spray headers) <sup>b</sup>		27.0	21.4
II.3 External Drywell Spray System <sup>c</sup>	3.7		35.9
II.4 Internal Drywell Spray System <sup>c</sup>	3.3		35.2
III. <u>CORE DEBRIS CONTROL (SAMDA 4.)</u>			
III.1 Rubble Bed Core Retention Device		38.4	35.5
III.2 Central Basemat Core Retention System	3.4		33.3
III.3 Dry Crucible Core Retention Device		118.8	108.8
III.4 Dry Crucible <sup>d</sup>	18.7		116.1
III.5 Core Distribution on Diaphragm Floor	3.3		9.2
IV. <u>ATWS-SIZED VENT (SAMDA 3.A.)</u>			
IV.1 ATWS Clean Steam Vent		3.9	2.6
IV.2 Clean Steam Venting to Stack	1.7		2.7
V. <u>DECAY HEAT SIZED VENT WITH FILTER (SAMDA 3.B.)</u>			
V.1 Gravel Bed Filter		11.3	9.2
V.2 Venting and Filtered System	2.8		5.9

Table 6 (Con't)

	RDA	PECo	NRC <sup>a</sup>
V.3 Multi-Venturi Scrubber System		5.7	4.0
V.4 Hardened Wet Well Vent		3.1	2.0
V.5 Combination Venting System	4.2		9.0
V.6 Large Chilled Filter System	2.9		6.7

-----  
Footnotes

- \* Systems that are grouped together are viewed as reasonably comparable (e.g., VI and V2.)
- a. NRC estimates were derived based on adjustments to PECo and RDA estimates. PECo estimates were revised downward in the following two areas:
1. all AFUDC was disallowed;
  2. engineering cost was recalculated based on 25% of direct construction cost.
- RDA estimates were revised upward based on the following adjustments:
1. RDA options I.2, I.3, III.2, and III.4 are assumed to incur replacement energy cost penalties. Costs are based on number of days assumed for comparable systems costed by PECo and daily cost of \$500,000 based on NUREG/CR-4012, Vol. 2. RDA items II.3 and II.4 also include replacement costs because they include option I.2 (see footnote c). For all these options, this is the dominant NRC adjustment;
  2. engineering cost was recalculated based on 25% of direct construction cost;
  3. cost allowance was made for the present worth of 40 years of operation and maintenance expenses;
  4. cost allowance was made for regulatory/licensing, and procedural activities;
  5. cost allowance was made for training;
  6. labor installation cost was increased to reflect lower labor productivity for completed and operating reactors, and learning curve effects;
  7. total cost is adjusted to account for general inflation between 1983-4 and 1990; and
  8. RDA's contingency factor of 1.25 is applied to the recalculated total cost.
- b. These systems include costs of system I.1
- c. These systems include costs of system I.2
- d. This system includes cost of system I.2

TABLE 7  
COMPARISON OF COSTS AND BENEFITS OF SAMDAs FOR LIMERICK

<u>Design Alternative</u>	<u>Estimated Cost (Millions of 1990 Dollars)</u>	<u>Averted Risk (Person-rem per Reactor-Year)</u>	<u>Dollars per Person- Rem Averted<sup>1</sup></u>
1. Dedicated Suppression Pool Cooling	21	80	6600
2. Alternate Means of Decay Heat Removal	Minimal <sup>2</sup>	80	300
3. Improved Venting Capability			
A. ATWS-Sized Vent	3	88	850
B. Decay Heat-Sized Vent with Filter	4 <sup>3</sup>	215	500
C. Decay Heat-Sized Vent without Filter	2	70	700
4. Core Debris Control	35	20	44000
5. Drywell Overpressure/Overtemperature Protection			
A. Drywell Sprays	3 <sup>5</sup>	178	400
B. Drywell Head Flooding	Minimal <sup>2</sup>	50	500
6. Makeup to Reactor Using Low Pressure Diesel-Driven Pump	Minimal <sup>2</sup>	100	250
7. Enhanced Reactor Depressurization Capability	2 <sup>5</sup>		
A. In Conjunction with #3B	6	302	500
B. In Conjunction with #4	37	193	4800
C. In Conjunction with #5A	5	236	500
D. In Conjunction with #5B	Minimal <sup>2</sup>	88	300
E. In Conjunction with #6	3	273	300
8. Reactor Building Decontamination Factor Improvement	3 <sup>6</sup>	50	1500

<sup>1</sup> Estimated assuming a 40 year plant life.

<sup>2</sup> Detailed cost estimates not available but expected to be minimal. SAMDA would involve minor modifications to hardware, procedures, and training. For purposes of estimating the cost/benefit ratio, a cost of 1 million dollars was assumed.

<sup>3</sup> Cost for a multi-venturi scrubber system (MVSS)

<sup>4</sup> Not available.

<sup>5</sup> Reference 6.

<sup>6</sup> This modification was assumed to be similar in cost to option 5.A.

## APPENDIX A: RISK REDUCTION BENEFITS FOR CANDIDATE SAMDAS

The risk reduction benefits<sup>1</sup> for the various candidate SAMDAs are based on the information in Tables 1, 2, and 3. The tables present total person-rem/reactor-year, land area for long-term interdiction and early fatality estimates. These risk estimates are based on accident frequency estimates that resulted from the BNL review (NUREG/CR-3028) of the Limerick PRA but which also take into account the NRC staff's recommendations given in NUREG-1068. The NRC recommendations have been implemented at Limerick and result in a 2.5 reduction in the Class 1 accident frequency estimates relative to the numbers given in NUREG/CR-3028.

1. Enhanced Suppression Pool Cooling

This SAMDA is designed to maintain suppression pool subcooling. The main potential benefit is to prevent the overpressure challenge for Class 2 accident sequences. The assumption is that the SAMDA would be designed for decay heat levels and would not therefore be effective for mitigating Class 4 accident sequences. In addition maintaining suppression pool subcooling does not mitigate the containment challenges for Class 1 and 3 accidents so that this SAMDA is only effective for Class 2 accidents.

Potential benefit: 80 person-rem/reactor-year

2. Alternative RHR System

This SAMDA will provide the same potential benefit as described above.

3. Improved Venting Capability3A. ATWS Sized Vent

This would be a "clean" vent system sized for mitigating Class 4 ATWS accidents. The vent would be opened prior to core damage in order to prevent structural failure of the containment. The main potential benefit is, therefore, to prevent containment failure and hence core damage for Class 4 accidents. However, the vent would also be helpful for preventing containment failure and core melt for Class 2 accidents. The vent could not be very effective for mitigating Class 1 and 3 accidents without some form of filtering. Even if the vent was taken from the wetwell air space suppression pool bypass mechanisms could still result in a significant fission product release (principally from core/concrete interactions and revolatilizations from the reactor vessel). Therefore no mitigation of Class 1 and 3 accidents was assumed for this vent.

Potential benefit:

Class 1 (No mitigation) =	--
Class 2 (Factor of 10 reduction) =	70
Class 3 (No mitigation) =	--
Class 4 (100% mitigation) =	18
TOTAL	<u>88</u> person-rem/year

<sup>1</sup> The risk reduction estimates in this appendix have been rounded in some cases. These approximations have no appreciable impact on the outcome of the cost benefit analysis.

3B. Decay Heat Sized Vent with Filter

This SAMDA would provide some mitigation of Class 1, 2, and 3 accidents but not Class 4 ATWS events. However, some fraction of Class 1 accidents and the majority of Class 3 accidents are predicted to have the reactor vessel at high pressure during core meltdown. If the core debris is ejected from the reactor vessel under pressure then it is possible for the containment to fail during the blowdown. Because of uncertainty in containment performance during high pressure core meltdown accidents, the vent is assumed to be only 50% effective for mitigating these events.

Potential benefit:

Class 1 high pressure (50% mitigation)	87
Class 1 low pressure (100% mitigation)	19
Class 2 (Factor of 10)	70
Class 3 high (50% mitigation)	39
Class 4 (no mitigation)	--
TOTAL	<u>215</u> person-rem/reactor-year

3C. Decay Heat Sized Vent Without Filter

This vent would be effective for mitigating only Class 2 accidents. It would not be effective for Class 4 ATWS events or for Class 1 and 3 accidents (because of suppression pool bypass).

Potential benefit:

Class 1 (No mitigation) =	--
Class 2 (Factor of 10) =	70
Class 3 (No mitigation) =	--
Class 4 (No mitigation) =	--
TOTAL	<u>70</u> person-rem/reactor-year

4. Core Debris Control

This SAMDA would be designed to prevent core/concrete interactions and remove decay heat from the core debris. The SAMDA would therefore be effective for mitigating containment challenges associated in the high pressures and temperatures caused by core/concrete interactions (i.e., Class 1 and 3 accidents only). However, unless the SAMDA includes some form of collection device (or way of directing the core into the SAMDA) it would not be effective for core meltdown accidents with the reactor vessel at high pressure. Thus the SAMDA is assumed to be effective for mitigating only those fraction of Class 1 accidents that are at low pressure during core meltdown.



## Potential benefit:

Class 1 high pressure (No mitigation)	--
Class 1 low pressure (100% mitigation)	20
Class 2 (No mitigation)	--
Class 3 high (No mitigation)	--
Class 4 (No mitigation)	--
TOTAL	<u>20</u> person-rem/year

5. Drywell Overpressure/Overtemperature Protection5A. Enhanced Drywell Spray System

Ensuring spray operation during Class 1 and 3 accidents has the potential to cool the drywell atmosphere and the core debris and thus minimize the threat from overtemperature. However, unless the sprays terminate core/concrete interactions, the non-condensibles released from the concrete will still cause the containment to eventually fail because of overpressure. However, even if the containment fails, the sprays would reduce the airborne fission product concentration and thus lower the source term. A DF of 3 was assumed for the sprays if the containment eventually fails. Again because of uncertainty associated with high pressure core meltdown the sprays are assumed to mitigate only 50% of the high pressure accident sequences.

The enhanced spray system would be designed to remove the decay heat so that it could potentially mitigate Class 2 sequences. However, it could not prevent containment failure and core melt for Class 4 ATWS events.

## Potential Benefit:

Class 1 high pressure (50% mitigation with DF-3) =	59
Class 1 low pressure (100% mitigation with DF-3) =	13
Class 2 (100% mitigation) =	80
Class 3 high pressure (50% mitigation with DF-3) =	26
Class 4 (no mitigation) =	--
TOTAL	<u>178</u> person-rem/ reactor-year

5B. Drywell Head Flooding

This modification requires flooding of the drywell head. It could potentially mitigate those accidents that result in leakage through the drywell head (refer to Table 1).

## Potential Benefit:

Class 1 (high pressure) leakage =	113 person-rem/reactor-year
Class 1 (low pressure) leakage =	13 person-rem/reactor-year
Class 3 (high pressure) leakage =	13 person-rem/reactor-year

Because of uncertainty in containment performance for high pressure core melt accidents a 50% effectiveness is again assumed. Also a pool DF of only 3 was assumed for assessing the effectiveness of this SAMDA.

Potential Benefit:

Class 1 high pressure (50% mitigation, DF-3) =	38
Class 1 low pressure (DF-3) =	8
Class 3 high pressure (50% mitigation, DF-3) =	4
TOTAL	<u>50</u> person-rem/year

6. Enhanced Reactor Vessel Depressurization

Enhanced reactor vessel depressurization will have very little impact on the plant risk estimates unless used in conjunction with other SAMDAs. This is because even with the reactor vessel depressurized the containment is predicted to fail early (within 3 hours) so that there is little attenuation of the source term during this time period using WASH-1400 methods.

However, some of the SAMDAs considered above that were assumed to be only effective for 50% of the high pressure accidents will be more effective when coupled with depressurization. For the purpose of this analysis, all Class 1 sequences were assumed to be at low pressure, but Class 3 sequences were assumed to be high pressure events.

6A. In Conjunction with 3B

Potential Benefit:

Class 1 all low pressure (100% mitigation) =	193
Class 2 (Factor of 10) =	70
Class 3 high pressure (50% mitigation) =	39
Class 4 (No mitigation) =	--
TOTAL	<u>302</u> person-rem/reactor-year

6B. In Conjunction with 5A

Potential Benefit:

Class 1 all low pressure (DF-3) =	129
Class 2 (100% mitigation) =	80
Class 3 high pressure (50%, DF-3) =	27
Class 4 (No mitigation) =	--
TOTAL	<u>236</u> person-rem/reactor-year

6C. In Conjunction with 5B

Potential Benefit:

Class 1 all low pressure (DF-3) =	84
Class 3 high pressure (50%, DF-3) =	4
TOTAL	<u>88</u> person-rem/reactor-year

6D. In Conjunction with 4

Potential Benefit:

Class 1 all low pressure =	193
Class 3 high pressure (no mitigation)	--
TOTAL	<u>193</u> person-rem/reactor-year

7. Diesel-Driven Low Pressure Reactor Makeup Water System

This SAMDA can potentially prevent core damage for those accident sequences in which the reactor vessel is depressurized and all other ways of injecting water have been lost. This SAMDA is therefore potentially of benefit for some Class 1 and Class 2 sequences. It will be of benefit for Class 2 sequences provided it can continue to operate after the pool becomes saturated and the containment fails.

Potential Benefit:

Class 1 high pressure (no mitigation) =	--
Class 1 low pressure (100% mitigation) =	20
Class 2 (100% mitigation) =	80
Class 3 (no mitigation) =	--
Class 4 (no mitigation) =	--
TOTAL	<u>100</u> person-rem/reactor-year

8. Alternate Low Pressure Reactor Makeup Water System

This SAMDA is similar to SAMDA 7 but has the additional capability of depressurizing some of the Class 1 accident sequences so that core damage can be prevented for a larger fraction of this accident class. The potential benefit is 193 and 80 person-rem per reactor year from Class 1 and Class 2 sequences, respectively.

9. Secondary Containment Improvement in DF

This SAMDA would be effective for those accidents that result in leakage. Mitigation of these failure modes by drywell head flooding was addressed in SAMDA 5.B. and in SAMDA 6C (with enhanced reactor vessel depressurization). A DF of 3 was assumed for the flooding SAMDA. A similar benefit would be expected from an improved secondary containment DF.

TABLE 1  
 Person-rem/year Within 50 Miles As a Function of  
 Accident Class and Failure Mode  
 Assuming FES Results with Modified Class 1 Frequency

Accident Class	Overpress/Overtemp Failure			H Burn	Leakage		Total
	Drywell	Wetwell Airspace	Wetwell Pool		With SGTS	Without SGTS	
Class 1 (High Pressure)	52	2	Neg <sup>1</sup>	4	2	113	174
Class 1 (Low Pressure)	6	Neg	Neg	1	Neg	13	19
Class 2	40	36	4	Neg	NCM <sup>2</sup>	NCM	80
Class 3	33	30	3	Neg	Neg	13	79
Class 4	9	8	1	Neg	Zero	Zero	18
Total	140	76	8	5	2	139	370

1. Negligible

2. No Core Melt

TABLE 2  
 Land Area for Long-Term Interdiction (m<sup>2</sup>/year)  
 As a Function of Accident Class and Failure Mode  
 Assuming FES Results with Modified Class 1 Frequency

Accident Class	Overpress/Overtemp Failure			H Burn	Leakage		Total
	Drywell	Wetwell Airspace	Wetwell Pool		With SGTS	Without SGTS	
Class 1 (High Pressure)	7	Neg	Neg	6	Neg	243	256
Class 1 (Low Pressure)	1	Neg	Neg	1	Neg	27	29
Class 2	95	85	10	Neg	NCM	NCM	190
Class 3	47	43	5	1	Neg	26	122
Class 4	17	15	3	Neg	Zero	Zero	35
Total	167	143	18	8	--	296	632

TABLE 3  
 Early Fatalities (per year) As a Function of  
 Accident Class and Failure Mode  
 Assuming FES Results with Modified Class 1 Frequency

Accident  Class	Overpress/Overtemp Failure			H Burn	Leakage		Total
	Drywell	Wetwell Airspace	Wetwell Pool		With SGTS	Without SGTS	
Class 1* (High Pressure)	Zero*	Zero*	Zero*	Neg	Zero	Neg	Neg
Class 1 (Low Pressure)	Zero	Zero	Zero	Neg	Zero	Neg	Neg
Class 2	Zero	Zero	Zero	Neg	NCM	NCM	Zero
Class 3*	Zero*	Zero*	Zero*	Neg	Zero	Neg	Neg
Class 4	1(-4)	7(-5)	1(-5)	Neg	Zero	Zero	1.8(-4)
Total	1(-4)	7(-5)	1(-5)	Neg	Neg	Neg	1.9(-4)

\* The base case results in NUREG/CR-3028 did not calculate any early fatalities for Class 1 and Class 3 accidents because of the assumed warning time (4 hours) before fission product release. It was noted in NUREG-1068 that for high pressure core meltdown accidents it is possible for the containment to fail at the time the core debris penetrates the reactor vessel. If this were to occur then the warning time for evacuation would be shorter than assumed in NUREG/CR-3028 and some early fatalities would be predicted for Class 1 and 3 sequences.

## APPENDIX B - STAFF ESTIMATES OF COST OF SAMDAs FOR LIMERICK

This Appendix provides a general discussion of the cost elements contributing to the staff's estimates of the costs of SAMDAs for Limerick.

1. General Inflation

The RDA results<sup>1</sup> were prepared in early 1984 (1983-1984 dollars) whereas the PECO estimates<sup>2</sup> were developed in mid 1989 (1989 dollars). Assuming implementation of a mitigation system is approved, work would likely commence in 1990 or beyond. Costs should be expressed in 1990 dollars. For PECO's estimates the impact is negligible. However, RDA's estimates should be adjusted upward by 25 percent based on actual and projected changes in the GNP Implicit Price Deflator between 1984 and 1990.

2. Replacement Energy Costs

Replacement energy cost penalties are potentially a dominant cost factor for backfits to existing power reactors. In NUREG/CR-4012<sup>3</sup> the staff estimates incremental costs on the order of \$500,000 for each day one of the Limerick units is out of service in the 1990 timeframe.

The RDA study notes that replacement energy costs have not been factored into their analysis although for several of the modifications the authors do acknowledge the need for plant downtime.

The PECO study assumes that for each mitigation system a portion of the construction activity will require the reactor to be shut down.

However, in most instances the downtime is projected as 13 weeks in duration and is assumed to be accommodated during normally scheduled outages. However, for three of these options, incremental outages of about 1, 2, and 5 months are projected and for these options replacement energy costs are included in their cost estimate. For these options, this cost element is the major contributor to the cost differential observed between PECO and RDA. In the staff's view, PECO's inclusion of replacement energy costs under these select circumstances is reasonable, particularly since most downtime has been assumed to be accommodated within scheduled outages.

Select adjustments to RDA system costs were made in the staff's cost estimates. The systems impacted and bases are indicated in the notes to Table 1. Essentially, the staff adopted the incremental downtime reported by PECO but applied the NRC daily replacement energy cost penalty of \$500,000 vs PECO's own estimate of \$850,000 per day. Nevertheless, for these select systems, the addition of replacement energy costs constituted the dominant adjustment to the RDA cost estimates.

PECO estimates that any one of the modifications will require a construction period of from about 1 to 2 years. The staff cautions that if Limerick 2 operations delayed pending installation of one of these mitigation systems, replacement energy cost penalties on the order of hundreds of millions of dollars would be incurred.

\*Enhanced Drywell Spray System, Water-Cooled Rubble Bed, Dry Crucible.

### 3. Labor Installation Costs

NRC's generic cost methodology recognizes a dramatic fall off in labor productivity when the work environment shifts from a new construction environment to a completed or operating reactor.<sup>4,5,6</sup> Worker productivity is affected by access and handling constraints, congestion and interference, radiation environments, manageability considerations, removal activities, and security constraints. For example, an outage activity performed in containment at an operating reactor, which best characterizes a good deal of the work proposed here, requires over three times the manpower requirements of comparable work in a new construction environment, based on NRC generic cost estimating assumptions.<sup>7</sup>

The staff's review of the RDA report suggests that their costs have not been adjusted adequately to account for this. The cost differences for reactors in the design stage (Case A) vs. operating reactors (Case C) are minimal, and since costs under Case C allow for "...radiation protection, draining of equipment, etc."<sup>8</sup> it is likely that no adjustment has been made for lower labor productivity. The PECO report, on the other hand, acknowledges the inclusion of labor productivity adjustments<sup>9</sup> and clearly, its labor cost category is consistently significantly higher than RDA's.

PECO's higher labor cost estimates are also consistent with NRC's inclusion of learning curve factors in its generic cost methodology.<sup>10</sup> If it is the first or second time industry will be performing these activities, which appears likely for much of the work proposed here, labor costs are estimated to be 2.5 to 3.6 times higher than for activities that have been performed by industry 3 or more times. For these reasons the higher labor costs embedded in PECO's estimates appear more reasonable. Consequently, the labor installation cost component for the RDA systems was adjusted upward by a factor of 6 to account for NRC generic cost labor productivity and learning curve effects.

### 4. Engineering

The NRC's generic cost estimate for engineering effort for complex modifications to operating reactors consists of a 25 percent cost factor to be applied to the direct construction cost.<sup>11</sup> Wide variability in this cost factor is acknowledged. For example, a much larger engineering cost factor is to be expected for relatively minor structural/system changes where engineering analysis is required. Alternatively, large modifications involving primarily off-the-shelf items are likely to require a minimal amount of engineering as a percentage of the direct cost.

Both RDA and PECO include engineering effort in their overall cost estimates. RDA assumes engineering constitutes 12 percent of the direct labor and material costs.<sup>12</sup> PECO's engineering cost is significantly higher. For the more expensive mitigation systems, PECO's "engineering" cost category typically ranges in the mid to high 30 percent range as a percentage of direct costs. For the less expensive options, the engineering effort typically approaches and exceeds 100 percent of the direct construction cost. Additional engineering effort associated with the PECO Nuclear Engineering Department and Field Engineering are included in their overall



estimates. These engineering efforts are embedded in their "station/owner" cost category.

The staff's cost estimate modifies both RDA's and PECO's engineering cost based on a 25 percent cost factor applied to the direct construction cost.

#### 5. Regulatory and Procedural Costs

In the staff's view, the RDA study attempts to quantify only the most direct costs associated with the proposed mitigation systems. In reality, physical modifications of this nature are likely to necessitate numerous regulatory/licensing and procedural requirements. For example, the issuance of new technical specifications, rewriting of procedures and training manuals, training sessions for operators and supervisors, issuance of detailed documentation and analytical reports, and extensive interfaces with the NRC are all likely to materialize if one of these mitigation systems is adopted. The RDA report does not include any costs for these activities. PECO captures most of these costs under its "regulatory" cost category. These regulatory costs range from about 1 percent to 5 percent of the total cost for the various options under consideration, and were based on 25 percent of PECO and Bechtel engineering and home office costs.<sup>13</sup> In absolute dollars these regulatory costs range from about \$0.15 million to \$1 million per reactor. The PECO estimates included additional cost allowances for training related activities that in some instances exceed \$0.5 million. In the staff's view an allowance for these factors is not unreasonable and are an appropriate addition to a comprehensive cost estimate. The staff's cost estimates modified RDA's costs by incorporating allowances for regulatory/licensing and procedural requirements. An estimate of \$0.5 million was derived from NRC's generic cost estimating<sup>14</sup> methodology and was incorporated in RDA's overall cost calculation unless PECO identified lower costs for a comparable system. In those circumstances, PECO's lower estimates for regulatory and training requirements were adopted by the staff.

#### QA/QC, O&M, Land, Profit, Insurance

The RDA study includes no allowance for QA/QC, O&M costs, land costs, profit (assuming contractors perform part or all of the work), or liability insurance. The RDA authors, in recognition of comments that their estimates were unrealistically low performed a sensitivity analysis on one of their baseline estimates. Adding allowances for just land costs and QA/QC caused their baseline cost to increase by a factor of 1.75.<sup>15</sup> In the staff's view, most of these factors are either already accounted for by the staff's earlier adjustments [e.g., engineering factor of 25% includes an allowance for QA/QC], or are sunk costs that are not incremental to the mitigation system [e.g., land]. However, O&M costs are a legitimate cost of all physical modifications. For example, maintenance, cleaning, testing, and inspection of the new hardware will be required over its assumed 40 year life. The present worth cost of this stream of expenditure is included in the PECO estimates. An allowance of either \$50,000 or \$100,000 has been added to the RDA estimates.

#### 6. AFUDC

Allowance for funds used during construction captures the interest paid on monies expended during the life of the project. PECO's estimates include this item which typically constitutes between 8 percent and 14

percent of the total cost, and for two of the mitigation systems analyzed exceeds \$10 million of the total cost.

The staff recognizes that AFUDC is a real cost to the utility, but disallows it for value-impact analysis purposes. In a value impact context all future costs are subject to present worth considerations and discounting. PECO's inclusion of AFUDC acknowledges that the monies will be expended over time, but these same cost streams have not been discounted in the PECO analysis. Assuming PECO's cost of money is reasonably commensurate with the discount rate would minimize the importance of the distinction between AFUDC and present worth considerations.

References for Appendix B

1. NUREG/CR-4025, "Design and Feasibility of Accident Mitigation Systems for Light Water Reactor," R & D Associates, August 1985.
2. "Cost Estimate for Severe Accident Mitigation Design Alternatives -- Limerick Generating Station for Philadelphia Electric Company," Bechtel Power Corporation, June 22, 1989.
3. NUREG/CR-4012, Volume 2, "Replacement Energy Costs for Nuclear Electricity-Generating Units in the United States: 1987-1991," Argonne National Laboratory, January 1987.
4. NUREG/CR-4627, Revision 1, "Generic Cost Estimates -- Abstracts from Generic Studies for use in Preparing Regulatory Impact Analyses," Science and Engineering Associates, February 1989.
5. NUREG/CR-5138, "Validation of Generic Cost Estimates for Construction-Related Activities at Nuclear Power Plants," Science and Engineering Associates, May 1988.
6. NUREG/CR-4546, "Labor Productivity Adjustment Factor," Science and Engineering Associates, March 1986.
7. NUREG/CR-5138, pp. 8, 14.
8. RDA, P. 2-10.
9. Bechtel, P. 36.
10. NUREG/CR-5138, pp. 17-19.
11. NUREG/CR-4921, "Engineering and Quality Assurance Cost Factors Associated with Nuclear Plant Modification," United Engineers and Constructors, April 1987.
12. RDA, p. 2-10.
13. Bechtel, p. 36.
14. NUREG/CR-4627, Rev. 1.
15. RDA, pp. A-20 to A-22.