10/10/83

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
Philadelphia Electric Company	)	Docket Nos. 50-352
	)	50-353
(Limerick Generating Station, Units 1 and 2)	)	
	)	

LEA'S REPLY TO APPLICANT AND STAFF RESPONSE
TO SEVERE ACCIDENT RISK ASSESSMENT CONTENTIONS

Preliminary Statement

By order of September 21, 1983 ("Memorandum and Order Regarding Parties' Motion for Clarification and Schedule for Prehearing Conference"), the Board granted LEA the opportunity to reply in writing to the answers of the Applicant and NRC Staff on the admissibility of LEA's proposed contentions regarding the environmental assessment of the impact of severe accidents. This document constitutes LEA's reply.

LEA will first address the general concerns regarding admissibility of the contentions. Integrated with this discussion is LEA's reply regarding Contention SARA-6. Discussion of other contentions follows, *seriatim*.

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SUMMARY OF ARGUMENT

The National Environmental Policy Act (NEPA)[1] and Commission regulations[2] require a detailed consideration of alternatives to a proposed licensing action. The Staff is well aware of ongoing research into measures to mitigate the impact of severe accidents, such as filter venting of containments, and other more reliable containment heat removal subsystems. Research on filter venting and other improved containment heat removal systems has been considered by the Commission in a proposed policy statement to be sufficiently advanced to require the consideration of cost-effectiveness of such measures in future construction permit applications.[3]

To date, the environmental review of the Limerick licensing proposal has not considered any alternative to prevent or mitigate the impact of severe accidents. LEA's contention SARA-6 contends that NEPA and Commission regulations implementing NEPA require some consideration in the licensing process of at least those alternatives of which the staff is aware through Commission sponsored research and its review of the facility.

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[1] 42 U.S.C. §§4321 et.seq.

[2] 10 C.F.R. §§51.23 (c)

[3] 48 Fed. Reg. 16020

LEA's other SARA contentions specify certain inadequacies of the environmental risk analysis performed for NEPA purposes.

General Concerns Regarding Admissibility

The Applicant presents three arguments against admissibility of LEA's SARA contentions:

(1) NEPA imposes no additional safety requirements in this proceeding.

(2) No mitigative alternative to PECO's proposal for licensing is "feasible" or required.

(3) The individual contentions are "without basis and specificity". [1]

Applicant's summary of its argument, "The Nuclear Regulatory Commission Has No Duty To Augment Its Safety Requirements Under the Auspices of NEPA", demonstrates that the Applicant misses LEA's point, and ignores relevant Commission regulations.

While the obligations of an agency under NEPA are "essentially procedural", (Vermont Yankee Nuclear Power

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[1] Applicant's responses to the individual contentions constitute nothing more than attempts to attack the merits of the contentions. As this Board is well aware, such attempts at this stage of the proceeding are improper (Houston Lighting and Power Co., (Allens Creek), 50-446, 11 NRC 542 (1980)), notwithstanding Applicant's concealing each argument on the merits in the veil of an attack on the "basis" for the contention.

Corp. v. Natural Resources Defense Council, Inc. 435 US 519, 558 (1978)), the purpose of those "essentially procedural" obligations is to compel the agency to take a "hard look" a the environmental consequences of its actions (Cf. Kleppe v. Sierra Club, 427 US 390, 410 n. 21 (1976), and to require it to undertake a "thorough study and a detailed description of alternatives" to the proposed action. Monroe County Conservation Council, Inc. v. Volpe, 472 F2d. 693, 697-98 (2d Cir., 1972).[1] This requirement for a thorough study and a detailed description of alternatives has been described as "the linchpin of the entire impact statement." Id.[2]

NEPA requires more than merely the full disclosure of environmental consequences and project alternatives.

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1 The National Environmental Policy Act requires an agency to include in every report on proposals for major federal actions significantly affecting the quality of the human environment a detailed statement on "alternatives to the proposed action". 42 U.S.C. §4332(2)(c)(iii).

2 See, 40 CFR §1502.14 (1980), considering the comparison of alternatives to be at the "heart of an environmental impact statement". 10 CFR §51.23(d) requires that the Commission be guided by these CEQ guidelines.

An agency is not free to disclose adverse consequences of a project, and alternatives to reduce those consequences and then completely ignore them. NEPA requires their "full consideration in agency decision-making". Natural Resources Defense Council, Inc. v. Grant, 355 F. Supp. 280 (1973) [Emphasis added]. While an agency need not elevate environmental concerns over other appropriate considerations (Strycker's Bay Neighborhood Council v. Karlen, 444 US 223 (1980)), an agency is not thereby relieved of its obligation to consider alternatives to reduce environmental impact.[1] The detail of the consideration must be sufficient to show that the agency made a good faith effort to consider the values NEPA seeks to protect by explaining fully the agency's course of inquiry, analysis and reasoning. Philadelphia Council of Neighborhood Organizations v. Coleman, 437 F. Supp. 1341 (E.D. Pa. 1977) aff'd without opinion, 578 F. 2d 1375 (3d Cir., 1977).

The concept of "alternatives" for NEPA purposes is indeed "bounded by some notion of feasibility", Vermont

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1 Indeed, in Strycker's Bay, the trial court had found the agency's analysis of alternatives as "thorough and exhaustive", and the Court of Appeals had conceded that the agency had given consideration to the alternatives. At 226-227.

Yankee Nuclear Power Corp. v. Natural Resources Defense Council, Inc., 435 US 519 (1978). However, "the concept of 'alternatives' is an evolving one, requiring the agency to explore more or fewer alternatives as they become better known and understood." Id. At 552-553 [Emphasis added] .

The obligation to consider alternatives is not imposed solely by NEPA, but is specifically required by the Commission's regulations:

> The draft environmental impact statement will include a preliminary cost-benefit analysis which considers and balances the environmental and other effects of the facility and the alternatives available for reducing or avoiding adverse environmental and other effects...
>
> While satisfaction of Commission standards and criteria pertaining to radiological effects will be necessary to meet the licensing requirements of the Atomic Energy Act, the cost/benefit analysis will, for the purposes of NEPA, consider the radiological effects of the facility and alternatives.
>
> 10 CFR §51.23(c) [Emphasis added] .

Significantly, neither Applicant or Staff responses to SARA-6 even mention these Commission regulations implementing NEPA. LEA believes that these straight-forward regulations and NEPA's clear language mandate some consideration of mitigative alternatives.

Yet rather than conceding the point, both Staff and Applicant oppose the contention. Applicant's arguments consist largely of counsel's assertions of the "infeasibility" of alternatives which have not yet even been examined for Limerick. Staff's filing professes ignorance of a matter which is the subject of an enormous research program sponsored by the Commission. See, Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation, 48 Fed. Reg. 16014, 16017-8.

The thrust of SARA-6 is simply that NEPA requires that the environmental review for Limerick reflect a consideration of the Commission's intensive investigation of mitigative alternatives and an application of that investigation to the Limerick facility. This is a contention which addresses the threshold legal question of whether some consideration of mitigative measures is required. Both Applicant and Staff in opposing the contention appear to contend that NEPA does not require the crossing of that threshold, and that the Commission is free to completely ignore even preliminary insights and findings produced by research into mitigative measures.

PECO specifically argues that (1) no mitigative alternatives are required under Commission regulations, and (2) no mitigative alternatives are "feasible" for

purposes of NEPA consideration.

These arguments improperly prejudge the outcome of any analysis of the alternatives for Limerick, and ignore relevant Commission regulations, particularly 10 CFR §50.109, 10 CFR Part 50, Appendix A, and 10 CFR §51.23(c).

Applicant's suggestions that the Commission lacks the regulatory authority to examine mitigative measures and direct their implementation if warranted by public risk, completely ignore 10 CFR §50.109, and the discussion in Appendix A concerning "additional criteria".[1]

The "backfitting" regulations of §50.109 specifically authorize the Commission to require design changes "if it finds that such action will provide substantial, additional protection which is required for the public health and safety or the common defense and security". The Commission is further authorized to require a construction permit holder to "submit such information concerning the addition or proposed addition...or the modification or proposed modification, of structures, systems, components of a facility as it deems appropriate." 10 CFR §50.109.

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1 SARA-6 is a NEPA contention, and does not assert that implementation of mitigative measures is necessarily required. The decision to implement mitigation measures which have been considered as NEPA requires is left to the Commission's descretion. However, LEA's point is that Commission regulations amply authorize such a decision.

Further, the Commission's regulations specifically contemplate that "there will be some water cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions..." Part 50, App. A.

While LEA has consistently taken the position that Limerick is a special case, it has also consistently argued that the Commission's regulations themselves contemplate the identification of and appropriate treatment of such special cases.

Whether _in fact_, mitigative alternatives to PECO's licensing proposal are _required_ for implementation at Limerick is a conclusion which must be reached in the course of examining those alternatives, to determine the level of risk posed, and the opportunities for risk reduction offered by alternatives. Applicant's assertions of "remoteness" and "infeasibility" merely prejudge the outcome of the analysis. The _Indian Point_ licensing board has recognized that such matters cannot be "prejudged". With respect to contentions advocating the necessity for filtered vented containments or separate containment struc-

tures for relief of excess pressure at the Indian Point facilities, the Licensing Board rejected the licensees' arguments that prior to admitting the contentions, the Board is first required to determine that a significant risk exists without those measures:

> "We do not believe that the Commission intended that prior to admitting a contention advocating a safety measure, we should find that a significant risk surely exists without such safety measure. <u>We believe such a finding should reflect the outcome of this litigation rather than its starting point...</u>" <u>In Re Consolidated Edison Co. of New York</u> (Indian Point) <u>Dockets 50-247, 286-SP</u>, 16 NRC 1629 (1982) at 1634. [Emphasis added].

Similarly, PECO's arguments that no mitigative alternatives need be considered because NEPA does not require consideration of "remote and speculative" alternatives (PECO pleading, p. 9-11) prejudge the outcome of an examination of alternatives. <u>A fortiori</u>, one must look at what alternatives exist before one determines them to be "unfeasible". The prematurity at this stage of the proceeding of PECO's arguments on the merits is well illustrated by <u>Houston Lighting and Power Co</u>., (Allens Creek Nuclear Generating Station) Docket 50-446, ALAB-590, 11 NRC 542 (1980) (reversing the Board, and ordering it to admit a contention that a 256 square mile biomass farm was an environmentally

preferable and viable alternative to the .lens Creek facility).

Obviously, the mitigative measures contemplated by LEA's contention are nowhere so "remote and speculative" as the 256 square mile "biomass farm" that the Appeal Board held to sufficiently delineate an "alternative" for purposes of a NEPA litigation contention.

Among the mitigative measures contemplated by the contention are a filter venting of the containment. This alternative, which Applicant characterizes as so remote and speculative that it need not even be examined for NEPA purposes, is being required in a simplified form by France's Commissariat a l'Energie Atomique, for implementation within 2 years at all French nuclear facilities. See, "Inside N.R.C.", Vol. 5, No. 18, (September 5, 1983).

In the United States, programmatic analysis of vent-filtered containments is well under way. See, e.g., NUREG/CR-1029, SAND 79-1088, "Program Plan for the Investigation of Vent-Filtered Containment Conceptual Designs for Light Water Reactors" (Sandia Labratories, October 1979).

Another alternative which the contention suggests and

which is being investigated are various options for core retention. <u>See</u>, <u>e.g</u>., NUREG/CR-2155, SAND 81-0416, "A Review of the Applicability of Core Retention Concepts to Light Water Reactor Containments", (Sandia Laboratoreis, September, 1981).

As a result of its ongoing research program into mitigation strategies, the Commission has already determined that some of the mitigation alternatives which PECO argues are too remote and speculative for NEPA consideration are sufficiently feasible to warrant detailed cost-effectiveness analysis:

> "In future CP applications for both pressurized water reactors [PWRs] and boiling water reactors [BWRs], filtered-vented containment systems or a variation of such systems, should be provided if these yield a cost-effective reduction in risk." Proposed Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation, 48 Fed. Reg. 16014 (April 13, 1983) at 16019.

So, too with more reliable containment heat removal:

> "The Staff is studying the need for more reliable subsystems for containment heat removal...as possible alternatives to filtered venting for prevention of gradual overpressurization failure of the containment building. The cost-effectiveness of this alternative should be considered in the design of plants for new CP application." <u>Id</u>., at 16020.

These references establish an adequate factual basis for LEA's contention that adequate cost-effectiveness review is possible for NEPA consideration of these mitigative alternatives for the <u>Limerick</u> facility.

PECO complains at length about uncertainties associated with probabilistic risk assessment techniques, and asks this Board to conclude (prematurely, at this early stage of determining contention <u>admissibility</u>, without any record upon which such a conclusion could be reached) that "it is presently inappropriate, and as a practical matter, impossible to utilize these [PRA] techniques to impose additional safety requirements for this facility under the auspices of NEPA". PECO Pleading, at p. 6.

This prejudgment has been rejected in the Commission's proposed policy statement:

> "In sum, considering the experience with risk assessments thus far made, we conclude that the cost-effectiveness of risk reduction measures can be studies through PRA. Although there are limitations due to the many uncertainties associated with the use of PRA, the Commission considers it to be a valuable adjunct to the established regulatory process and the NRC's nuclear safety regulations in 10 CFR, Chapter I". Proposed Policy Statement, at 16016.

It may, or may not be, upon a consideration of alternatives, their cost, and their potential for risk reduction, that construction has proceeded so far that no mitigative measures will be cost-effective for Limerick.

That, however, is a conclusion which can only be reached by the Staff after it has considered the relevant factors.[1]

In contrast to the Applicant's prescience of the results of an alternatives analysis, the Staff scarcely

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[1] Applicant also argues that "only alternatives which are feasible in the time frame of facility licensing need be considered", but offers absolutely no authority for such a proposition. To the contrary, NEPA requires a consideration of alternatives which are available in a time frame which is "meaningfully compatible with the time-frame of the needs to which the underlying proposal is addressed". See, Vermont Yankee, supra, at 551, citing Natural Resources Defense Council v. Morton, 458 F2d 827, 837-838 (1972). In this case, the operating license for Limerick is for a duration of 40 years and the facility is expected to meet a need for power over that time period. LEA submits that NEPA requires some consideration of alternatives which would be available within that 40 year period.

seems to know what LEA is talking about at all, despite the fact that mitigative measures for plants including Limerick are the focus of intensive scrutiny of Commission technical research and regulatory debate.

The Staff response to SARA-6 complains that "no risks are identified at all" by LEA, and "no alternatives and mitigative measures are proposed" and that LEA is "not raising a contention but calling for an unspecified Staff study". (NRC Staff, Pleading, p.8)

Let us recall that we are discussing contentions filed in response to a document that for two volumes discloses the risk of severe accidents at Limerick and which has been filed in this proceeding.

The "risk" to which the contention refers is obviously the risk of severe accidents as disclosed by SARA, and by the BNL report to which the LEA's filing specifically referred.

As far as "proposing" alternatives and mitigative measures is concerned, any such proposal is premature in view of the Staff's refusal to concede the threshold legal question to which the contention is directed. In any event, the Staff is well aware of the existence of mitigative

alternatives for Limerick, having sponsored a preliminary analysis of such measures. See, e.g., NUREG/CR-2666, Chapter 7, "Further Considerations of Mitigative Features for Specific Plants: Limerick", PWR Severe Accident Delineation and Assessment (attached).

It appears, in fact, that while the Staff is contesting the NEPA necessity for an alternatives analysis, it is conducting just such an analysis for other purposes.

Subsequent to the original filing of LEA's SARA contentions, pursuant to the Freedom of Information Act, LEA sought disclosure by the Commission of non-public records relating to mitigation features for Limerick. The Commission finally responded to this request on October 3, 1983, providing to LEA's counsel information which heretofore was unavailable to LEA. That information clearly establishes that the Staff has received significant information concerning mitigation strategies for Limerick:

> c. Summary to date: For Mark II containment as exemplified by the Limerick Plant, mitigation requirements (functions) have been identified, including containment heat removal, core residue capture and retention without concrete attack, and (if ATWS events are to be mitigated) some kind of venting system. Candidate components to fulfill these requirements have been selected for preliminary conceptual design and cost estimation. Separate

cost figures will be generated for
1) Plants before construction begins,
2) Plants built but not yet in operation, and 3) Operational plants.

d. Plans for next period: Complete preliminary designs and assessments for Limerick, and begin final design of selected version.

"Monthly Project Status Report, September 15, 1983, Contract NRC-03-83-092", by R&D Associates, p. 4.

A copy of some of the information provided to LEA pursuant to its request is attached hereto, and provides additional factual bases for SARA-6.

This information concerning feasibility of mitigation alternatives which is available to the Staff is surely sufficient to "require reasonable minds to inquire further" (Vermont Yankee, supra, at 554), into this alternative for NEPA consideration. In fact, the "unspecified Staff study" (NRC Staff, Pleading, p.8) which LEA has contended is required for NEPA purposes is apparently being performed for other Commission purposes.

The Staff knows exactly what analysis is required. The question is whether what the Staff already knows must be considered in this licensing proceeding for purposes of complying with the NEPA requirement of a detailed consideration of alternatives.

SARA-1 and 2

Both the Applicant and Staff object to these contentions on the grounds that they were previously submitted as PRA contentions and denied by this Board.

LEA resubmitted these contentions as SARA contentions, in spite of the Board's previous ruling, because LEA read that ruling to hinge primarily on the Board's decision not to litigate PRA methodology for safety purposes. LEA did not request reconsideration of the Board's ruling on these two contentions due to that fact.

LEA has not attempted to "repackage" the contentions by making them more specific, since it would have been very difficult to do so without reproducing large portions of the Brookhaven report (NUREG/CR-3028); the Board is thus faced with a straightforward decision as to whether or not the contentions can be resubmitted for a different purpose than they were originally submitted. If they were denied the first time due largely to the Board's decision regarding litigation of PRA methodology, there is no problem with their reconsideration in a SARA (NEPA) context. If the Board is not convinced, based on LEA's statements regarding specificity, that the contentions are sufficiently specific to admit, they will again be denied.

SARA-3

Applicant objects to this contention alleging that LEA

> erroneously assumes that the entire population between 10 and 25 miles from the facility will be relocated. There is no basis given by LEA for asserting that such total evacuation would be necessary or was assumed in the model. In fact, the model simply assumed that the portion of the populace which could be affected by deposition from the plume would be relocated once the plume had passed.

PECO Pleading, p. 21.

This objection impermissibly addresses the merits, not the basis of the contention, and is based upon counsel's assertions nowhere supported by any matter appearing in SARA.

In any event, the objection misses the point of the contention. Even assuming the truth of Applicant's counsel's unsupported assertions that "only one, or at most a few of the 16 sectors at these distance would require relocation", the problem addressed by the contention still remains. SARA shows that the year 2000 population of sector SE is 680,330. (SARA,p. 10-33) The year 2000 population in the sector ESE is 505,011. (SARA, p. 10-33) If "only one or at most a few" of the sectors require rapid relocation,

the total population to be "rapidly relocated" still may easily exceed one million people. NRC Staff has understood the point: "LEA's point is that the area outside of the ten mile zone contains a high population density and asserts that for this reason the assumption in SARA, that people in the zone from ten to twenty miles from Limerick can be rapidly relocated, has an inadequate factual foundation". (NRC Staff Pleading, p. 5)

SARA-4

The Applicant's objection to this contention once again consists of a defense on the merits which properly belongs in hearing testimony, not in an objection to contention admissibility.

Contrary to Applicant's assertions that LEA "points to no better data which was overlooked in determining the best way to model evacuation" (PECO Pleading, p.22) the contention specifically points to the site-specific highway movement time of the Emergency Plan Appendix H evacuation time estimates.[1] Applicant complains about the fact that Appendix H "does not take into account the prompt notification system", and the impossibility of separating "preliminary evacuation notification time from preparation time which makes such data difficult to use in SARA" (PECO Pleading, p.23). But LEA never proposed the use of Appendix H's notification and preparation time

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1 While pointing to this data as "better", LEA does not necessarily propose its use until the errors delineated in LEA Contention VIII-6 are corrected.

estimates. The contention proposes using only the highway movement time component (a component independent of notification time) from which an evacuation speed can be calculated. Delay time (including preparation and notification time) is calculated separately from evacuation speed in the CRAC 2 evacuation model which SARA incorporates, and the proper evacuation speed can easily be entered as input in the CRAC 2 evacuation model.

With respect to the 3-hour notification time for seismically induced accidents, Applicant asserts that the low evacuation speed assumed "would compensate for any shift in notification time". (PECO Pleading, p.23) This assertion once again belongs in hearing testimony. However, LEA notes that it has earlier noted in this proceeding that health effects calculations are extremely sensitive to evacuation delay assumptions (See e.g., LEA Filing of April 12, 1983 (Specification of Conditionally Admitted Contentions, I-16b)), and the low evacuation speed would not necessarily compensate.

SARA-5

Applicant objects to SARA-5, alleging that "many of the consequences which are alleged to be omitted are, in fact, directly or indirectly included in SARA. LEA's complaint seems to be that instead of or in addition to expressing consequences in terms of dollars, SARA should state effects in terms of other units of measure, e.g., acres of land or job losses". (PECO Pleading, p.24)

To the contrary, LEA contends that NEPA requires a disclosure of what the consequences will be - not their obsfucation of the consequences by disclosing only their economic impact. Acceptance of Applicant's reasoning would permit an EIS to express all environmental impacts solely in economic terms. Such descriptions would conceal, not disclose, environmental impacts.

The impacts should be stated in terms that describe the impact - if 10 square miles of land will be permanently interdicted due to contamination, this impact should be so stated.

While Applicant provides a list of types of impact that are "included in CRAC 2 runs" (PECO Pleading, p.25) they are not set forth in SARA. When LEA's counsel requested certain results printouts from the CRAC 2 runs for SARA, Applicant's counsel informed him that the only runs done were those set forth in SARA.

If Applicant is willing to include in SARA complete CRAC 2 final results printouts, LEA's contention as to matters (3) - (6) would be satisfied.

With respect to health effects, while SARA does consider total thyroid nodules (SARA, p.12-17), the contention seeks disclosure of the total of <u>all</u> non-fatal cancers (not merely thyroid nodules) and other health effects known to be associated with radiation exposure such as sterility and genetic effects. Applicant does not even address these omissions.

With respect to items in (14), Applicant admits that SARA does not consider them "individually", and that the CRAC 2 runs which "implicily "consider them rely upon national averages. Once again, only the economic impact is considered; further, " national averages" do not reflect site-specific nature of the mineral and water resources and scenic and aesthetic resources referred to.

With respect to the compensation cost of health effects, LEA sees no reason why these costs should not be treated similarly to, and included within, the CCDF curves of "ex-plant costs" set forth in SARA. SARA's different treatment of such costs obscures their significance, contravening NEPA's "full disclosure" purposes.

SARA-6

LEA's reply to objections to this contention are integrated into its general discussion regarding admissibility of SARA contentions.

LEA corrects, for the record, a typographical error in SARA-6. At page 20 of its SARA contentions, the last line should read:

Hampshire, (Seabrook Units 1 and 2), 6 NRC 33 at 83.

It previously read:

Hampshire, (Seabrook Units 1 and 2), 5 NRC 33 at 83.

SARA-7

Both the Applicant and Staff rely on the Commission's Safety Goal Policy Statement of March 14, 1983, for the proposition that sabotage cannot be considered as part of the risk assessment for NEPA purposes.

LEA believes that a distinction should be made between the analysis needed for setting an absolute safety goal and the analysis that could be done for NEPA purposes. Sholly lays out in his discussion the information available for making such a risk assessment for NEPA. LEA believes that it is the duty of both the Staff and Applicant to make the best assessment presently possible of the risk of sabotage, recognizing that such an analysis will carry with it uncertainties sufficient to justify excluding it from the safety goals at this time.

Applicant also objects to this contention on the basis that Mr. Sholly has not been shown to be an expert in the field of probabilistic risk assessment. LEA notes for the record that in the recent special investigation at Indian Point, Mr. Sholly was recognized by Mr. Gleason, the Chair for that investigation, to be qualified to testify with respect to a wide variety of matters concerning nuclear safety. He was in fact qualified in that proceeding to testify as an expert on consequence analysis.

Both Staff and Applicant object to SARA-7(b). LEA notes that this contention was submitted as a PRA contention previously (I-23c), and was held in abeyance by the Board for possible refiling as a SARA contention to whatever degree necessary. It was reworded as resubmitted in order to comply with the Board's order that parties with similar contentions attempt to agree on wording prior to submission -- the City of Philadelphia brought into discussions with LEA a contention regarding errors of commission. That requirement necessitated keeping the wording of joint contentions simple.

Respectfully submitted,

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CERTIFICATE OF SERVICE

I hereby certify that the foregoing LEA's Reply to Applicant and Staff Response to Severe Accident Risk Assessment Contentions was served on October 10, 1983, by first-class mail, postage prepaid, upon the following*:

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CHAPTER 7. FURTHER CONSIDERATION OF MITIGATION FEATURES FOR SPECIFIC PLANTS: LIMERICK

As part of the review and analysis of the Probabilistic Risk Assessment (PRA) for Limerick [1], the question of mitigation features was considered. This consideration is divided into two sections. In Section 7.1, mitigation is considered within the context of the PRA as submitted. Because of possible changes due to updating, inconsistencies and further analysis of containment behavior, Section 7.2 considers mitigation in terms of alternative scenarios. In both cases, the discussion is qualitative. Section 7.3 has concluding remarks.

7.1 THE LIMERICK PRA

The dominant accident sequences identified in the Limerick PRA were grouped into four classes, as shown in Table (7.1). Note that Classes II and IV involve containment failure prior to core melt, and Classes III and IV involve transients with loss of scram function (ATWS). The containment event trees identified eleven potential failure modes, which were eventually combined into seven failure modes, as shown in Table (7.2). The frequencies of each of these seven failure modes for the four accident classes are shown in Table (7.3). Although Table (7.3) implies that 28 release categories could have been used in the consequence analysis, only five were used. These are shown in Table (7.4), which also indicates how the containment failure modes were combined into the five release categories.

For the consideration of mitigation, it is important to note the following. Release category OXRE combines all the in-vessel and ex-vessel steam explo-

TABLE 7.1. Accident Sequence Classes

Generic Accident Sequence Designator	Physical Basis for Classification	System Level Contributing Event Sequence
Class I (C1)	Relatively fast core melt; containment intact at core melt and at low pressure	Transients involving loss of inventory makeup, small LOCA events involving loss of inventory makeup
Class II (C2)	Relatively slow core melt due to lower decay heat power; containment failed prior to core melt	Transients or LOCAs involving loss of heat removal, inadvertent SRV opening accidents with inadequate heat removal capability
Class III (C3)	Relatively fast core melt; containment intact at core melt, but at high internal pressure	Transients involving loss of scram function and inability to provide coolant makeup, large LOCAs with insufficient coolant makeup transient with loss of heat removal and long term loss of inventory makeup
Class IV (C4)	Relatively fast core melt; containment fails prior to core melt due to over-pressure	Transients involving loss of scram function and loss of containment heat removal or all reactivity control, but which have coolant makeup capability

TABLE 7.2. Containment Failure Modes

CONTAINMENT FAILURE MODES

Designator	Description
α	Steam explosion in vessel
β	Steam explosion in containment
μ'	H_2 explosion induced containment failure
μ	H_2 deflagration sufficient to cause containment overpressure failure
δ	Overpressure small leaks (A_R = 0.05 ft^2)
γ	Overpressure failure (A_R = 2.0 ft^2) Release through drywell
γ'	Overpressure failure (A_R = 2.0 ft^2) Release through wetwell break
γ''	Overpressure failure (A_R = 2.0 ft^2) Wetwell pool drained
ζ	Overpressure, large leak (A_R = 0.2 ft^2)
$\zeta\varepsilon$	Overpressure, large leak, SGTS failure (A_R = 0.2 ft^2)
$\delta\varepsilon$	Overpressure, small leak, SGTS failure (A_R = 0.05 ft^2)

TABLE 7.3

SUMMARY -- GENERIC ACCIDENT SEQUENCE/RELEASE PATH COMBINATIONS*

CONTAINMENT FAILURE MODE \ CLASS	CLASS I	CLASS II	CLASS III	CLASS IV	TOTAL PROBABILITY BY CONTAINMENT FAILURE MODE
α	1.2×10^{-8}	9.6×10^{-10}	1.1×10^{-9}	1.3×10^{-10}	1.5×10^{-8}
β,μ'	2.5×10^{-8}	1.9×10^{-9}	2.2×10^{-9}	2.5×10^{-10}	2.9×10^{-8}
γ,μ	3.2×10^{-6}	2.5×10^{-7}	2.8×10^{-7}	6.4×10^{-8}	3.8×10^{-6}
γ'	2.8×10^{-6}	2.1×10^{-7}	2.4×10^{-7}	5.6×10^{-8}	3.3×10^{-6}
γ''	3.1×10^{-7}	2.4×10^{-8}	2.7×10^{-8}	6.3×10^{-9}	3.7×10^{-7}
$\zeta\varepsilon,\delta\varepsilon$	9.7×10^{-7}	7.5×10^{-8}	8.5×10^{-8}	2.5×10^{-11}	1.1×10^{-6}
ζ,δ	5.2×10^{-6}	4.0×10^{-7}	4.6×10^{-7}	2.5×10^{-11}	6.1×10^{-6}
TOTAL PROBABILITY BY CLASS	1.2×10^{-5}	9.6×10^{-7}	1.1×10^{-6}	1.3×10^{-7}	1.5×10^{-5}

* Reproduced from Table 3.5.14 of the Limerick PRA

TABLE 7.4. Release Categories Used in the Limerick PRA

Release Categories	Containment Failure Modes
$C4\gamma$	γ' - Class IV
$C4\gamma'$	$\gamma\mu$ - Class IV
$C4\gamma''$	γ'' - Class IV
OXRE	α - Class I, Class II, Class III, Class IV, $\beta\mu'$ - Class I, Class II, Class III, Class IV.
OPREL	$\gamma\mu$ - Class I, Class II, Class III γ' - Class I, Class II, Class III
Not Allocated to Any Release Category	γ'' - Class I, Class II, Class III, $\xi\varepsilon,\delta\varepsilon$ - All Classes
	ξ,δ - All Classes

sions, and hydrogen detonation failure modes. Release category OPREL combines drywell and wetwell overpressurization failures for Classes I, II and III. Over 50% of the total probability is not allocated to any release category. Included in this percentage are overpressurization failures of the wetwell (assuming that the suppression pool drains) for Classes I, II and III, and all leakage failures.

However, based on the five release categories, Philadelphia Electric Company (PECO) [2] was able to produce the risk as a function of release category These are shown in Tables (7.5) and (7.6). The calculations carried out by PECO show that, with reference to latent fatalities as a consequence measure, the total risk is distributed among the five release categories in the following way:

Release Category	Percentage of Total (latent risk)
$C4\gamma$	1.49
$C4\gamma'$	2.35
$C4\gamma''$	0.34
OXRE	4.02
OPREL	91.79

By the use of Table (7.4), it is possible to redistribute the risk directly among failure modes and classes. The result is shown in Table (7.7). This table can be used as the basis for a discussion on risk mitigation. It is apparent that more than 90% of the risk is associated with overpressure failure (with either wetwell or drywell break). Therefore, strategies for mitigation have to address this failure mode.

TABLE 7.5. ACUTE FATALITIES

(TOTAL RISK)

RELEASE CATEGORY	PROBABILITY (from Table 7.3)	NORMALIZED MEANS (from PECO)	MEANS BY CATEGORY	PERCENTAGE OF TOTAL
C4γ	5.6 (-8)*	3.556	0.199 (-6)	8.5
C4γ'	6.4 (-8)	11.02	0.705 (-6)	29.9
C4γ"	6.3 (-9)	75.78	0.477 (-6)	20.3
OXRE	4.35 (-8)	22.37	0.973 (-6)	41.3
OPREL	6.98 (-6)	-	-	-
		TOTAL	2.35 (-6)	100

* 5.6 (-8) = 5.6×10^{-8}

TABLE 7.6. LATENT FATALITIES

(TOTAL RISK)

RELEASE CATEGORY	PROBABILITY (from Table 7.3)	NORMALIZED MEANS (from PECO)	MEANS BY CATEGORY	PERCENTAGE OF TOTAL
C4γ	5.6 (-8)*	93.26	0.052 (-4)	1.49
C4γ'	6.4 (-8)	127.8	0.082 (-4)	2.35
C4γ"	6.3 (-9)	182.8	0.012 (-4)	0.34
OXRE	4.35 (-8)	332.0	0.14 (-4)	4.02
OPREL	6.98 (-6)	46.38	3.196 (-4)	91.79
		TOTAL	3.48 (-4)	100.0

*5.6 (-8) = 5.6 x 10^{-8}

Before considering the possibility of backfitting a Filtered-Vented Containment System (FVCS), however, it is necessary to know the possible role that containment sprays can have in case of accident. Containment overpressure is due to both steam and condensible gases. If, as is usually the case, steam is the dominant contributor to overpressure, then containment sprays would effectively reduce containment pressure in certain accident sequences. However, because of the uncertainties on the implementation of containment sprays under postulated severely degraded core conditions, their effect was not evaluated in the Limerick PRA. It would, therefore, be interesting to quantify the mitigation effect that the sprays can have in some sequences. Consideration should also be given to the possibility of upgrading their performance in order to cope with the severe environmental conditions in case of accident.

If the sprays do not seem to reduce the risk significantly, then a FVCS can be considered as a mitigation feature.

Work done at Sandia for Mark I and Mark III containment by A.S. Benjamin and F.T. Harper [3], has shown that, despite the differences between the two containments, basically the same filtered-venting strategies are thought to be effective in reducing the total risk significantly.

From the viewpoint of mitigation by a FVCS, Table (7.7) can be reorganized in the following way: if the γ and μ containment failure modes' contribution are decoupled by the use of the containment event trees, the following results:

TABLE 7.7. Total Risk Contributors

Containment Failure Mode	Class: I	II	III	IV	
α	1.11	0.09	0.10	0.01	1.31
β, μ'	2.30	0.18	0.20	0.02	2.70
γ, μ	42.09	3.29	3.68	2.35	51.41
γ'	36.83	2.76	3.16	1.49	44.24
γ''	--	--	--	0.34	0.34
	82.33	6.32	7.14	4.21	100.00

From Table 7.5, it is clear that release category OXRE is an important contributor to acute fatalities, even though OPREL has the highest probability. This occurs because the Limerick PRA consequence analysis determined that OPREL would result in zero early fatalities. Since this analysis is under review by the NRC Staff and their consultants, the discussion here will focus on mitigation of the γ and γ' failure modes, using latent fatalities as the measure of risk. If acute fatalities are excluded, it would appear that overpressurization of drywell or wetwell for Class I accidents is the largest contributor to risk.

Release Category	Percentage of Total
A - non-mitigable failure modes: α, β, μ', μ	5.86
B - mitigable*, no ATWS: γ, γ' (classes I, II)	83.34
C - mitigable* ATWS: γ, γ', γ'' (classes III, IV)	10.80
	100.00

Therefore, a low-volume venting strategy can potentially eliminate contribution B and the risk reduction factor is $\sim$6.

On the other hand, a combined low-volume/high-volume strategy, similar to the one proposed by Benjamin for a Mark I BWR, can potentially eliminate contributions B and C and the risk reduction factor is $\sim$17. This combined strategy may be necessary because ATWS events tend to give higher loadings over short periods of time.

Category A, above, assumes that a hydrogen burn will fail containment with a probability of one. Since this is conservative, the risk reduction factor may even be greater.

*Note that for Class II and IV, the containment fails prior to core-melt. Hence only a vent might be required.

7.2. LIMERICK MITIGATION

The Limerick PRA [1] expressed ideas about how vessel failure occurs and how subsequent dispersal of the core materials led one to the conclusion that diaphragm ~~diagram~~ failure fairly early in time would lead to immediate containment failure. Most of the dangerous decay products were assumed to have been sparged by the suppression pool before vessel failure. Currently it is thought that vessel failure will lead to core materials entering the suppression pool within the pedestal. It is argued that the suppression pool will act as a heat sink of sufficient magnitude to condense the initial large amounts of steam. The result is a slow overpressurization and containment failure, be it either the γ or the γ' mode. It is not clear why the γ'' mode is considered improbable in the Limerick PRA. The risk is not very different because of the early blowdown through the SRVs and resultant sparging by the suppression of the more dangerous constituents. What the fraction of the more dangerous decay products that are transported through the SRVs should be is addressed elsewhere.

The earlier sequence that leads to containment failure caused by diaphragm failure does not lend itself easily to mitigation. The diaphragm must be protected. To do this, we might consider modifying the region under the reactor vessel, if existing drains do not supply a sufficient path, so that the core materials enter the suppression pool without failing containment. This might be accomplished by replacing part of the concrete floor with an easily melted metal cover. Comments about mitigation of an event such as this would be the same as for the current scenario which is discussed below.

The slow overpressurization that is believed to follow core material entering the suppression pool could be mitigated by heat removal from the containment. This could be accomplished by a low volume flow vent-filtered system, a heat pipe or a containment spray system. Heat pipes or a vent filter have been discussed elsewhere in this report for application to PWRs. The BWR Mark II containment is similar to a PWR when cavity flood precedes vessel failure.

A spray system could mitigate slow overpressurization accidents, providing suppression pool cooling is possible. To increase reliability, a system could easily be devised that could be driven from outside the containment, using equipment brought to the site from elsewhere (portable pumps, diesels, etc.). Some consideration should be given to augmenting the spray system as an alternative method of mitigation. It should be kept in mind that, if water is added to the containment, it must eventually be removed. Some sort of a closed loop heat exchange process seems to be the most desirable approach.

PECO and their consultants [4] give arguments as to why slow pressurization will follow direct entry of the molten core material into the suppression pool. The PECO view is that very rapid condensation will lead to very little pressure increase due to steam generation and no communication of the wet well with the dry well. Questions have been raised regarding the efficiency of the condensation process. Calculations, assuming zero condensation, lead to very early containment failure. It is our opinion that the condensation rate will not be infinitely fast. The high rate of steam generation--recall the PWR steam spike--will overwhelm heat pipes. One is led to consider a high volume vent-filter system or sprays of very high capacity (if operated in a timely manner). This area needs a great deal more attention before conclusive statements can be made.

Whether or not the core debris bed will remain coolable within the pedestal needs to be determined. If there is any doubt, a rubble bed within the pedestal may be desirable. Suitable flow passages in the pedestal wall at the suppression pool floor level are needed. At this time it is not clear that they exist. Debris bed dryout is discussed in NUREG 0850, and the rubble bed and its design are discussed by Swanson [5].

Further consideration should be given γ'' failure of the containment because it would make mitigation much more difficult. The highly contaminated suppression pool water would flood surrounding areas and could create no great difficulties. Mitigation would require prevention of the γ'' mode. As this mode of containment failure is potentially the most dangerous, it deserves serious attention.

7.3 Concluding Remarks

The PRA and its ammendments, as submitted by PECO did not give realistic or best estimate scenarios. Rather, some conservative as well as controversial assumptions were made. One can only consider mitigation in the context of realistic, best estimate accident and containment failure scenarios.

* It is concluded that the next step should be a better definition of these scenarios so that mitigation strategies can be developed if proved to be necessary.

REFERENCES

1. Philadelphia Electric Company, "LIMERICK Generating Station Probabilistic Risk Assessment, 1981.

2. Letter, Treavor Pratt to W.E. Kastenberg, August 1982.

3. A.S. Benjamin, F.T. Harper and P. Cybulskis, "Risk Assessment of Filtered-Vented Containment Options for a BWR Mark I Containment", Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, September, 1981.

4. Summary of Meeting Between PECO and Consultants and NRC and Consultants, 3 Sept. 1982, to be published.

5. D.G. Swanson, "Core Melt Materials Interaction Evaluation", Annual Progress Report for April 1980 to March 1981, ASAI Report No. 81-001.

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

DOCKETED
USNRC

October 3, 1983

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Charles W. Elliott, Esquire
Brose and Poswistilo
1101 Building
Easton, PA 18042

IN RESPONSE REFER
TO FOIA-83-550

Dear Mr. Elliott:

This is in response to your letter dated September 13, 1983 in which you requested, pursuant to the Freedom of Information Act, drafts of Volume 2 of NUREG-0850 and documents regarding the feasibility of measures or design features to mitigate the consequences of a core melt accident at Limerick or any other reactor of the GE Boiling Water Mark I, II, or III Design.

During a September 23, 1983 telephone conversation with Mr. Stephen Isaacs of my staff you, (1) eliminated all Mark III design material from the request, and (2) were informed that there is no draft or manuscript of NUREG-0850, Volume 2.

The following documents address mitigating features in Mark-II containments:

1. NUREG/CR-3028, "A Review of the Limerick Generating Station Probabilistic Assessment."

2. NUREG/CR-2666, "PWR Severe Accident Delineation and Assessment."

3. NUREG/CR-3299, "Core Melt Materials Interactions Evaluation," Final Report (in publishing).

Enclosed are extracts of the first two monthly reports from the NRC Technical Assistance Contractors, R&D Associates.

NRC staff state that additional information requested in Item 2 was supplied in response to a previous FOIA request, FOIA-83-432. A copy of that request and response is enclosed.

Other NUREG's which are relevant to this request are:

1. NUREG/CR-2182 Vol. I & II - "Station Blackout at Browns Ferry Unit One-Iodine and Noble Gas Distribution and Release" (September 1982)

2. NUREG/CR-2572 Vol. I & II - "SBLOCA Outside Containment at Browns Ferry Unit I - Accident Sequence Analysis" (November 1982)

3. NUREG/CR-2973 "Loss of DHR Sequences at Browns Ferry - Unit I - Accident Sequence Analysis" (May 1983)

These reports are available by contacting the NRC Public Document Room (PDR), 1717 H Street, NW, Washington, DC 20555.

This completes action on your request.

Sincerely,

J. M. Felton, Director
Division of Rules and Records
Office of Administration

Enclosures: As stated

MONTHLY PROJECT STATUS REPORT
September 15, 1983

Report No. RDA-MR-127300-003
Period Covered: July 30 through August 31, 1983
Name of Program: Severe Accident Mitigation Systems
Contract Number: NRC-03-83-092
Start Date: June 27, 1983 Completion: 27 months

SECTION A: Technical Summary of Project Status

Technical progress has been good and active work continues on Tasks 1-5. (Task 6 is done only on specific assignments). As agreed to at the kick-off meeting (7/12/83), the first specific containment type to be studied in Task 3 is the Mark II BWR, as exemplified by the Limerick plant. We have been requested to make a special rapid response on our study of this type plant by January 1, 1984. This mini-study will include assessments based on Task 1, 2, 3, and 4 activities, and thus represents the first priority of activity for all tasks.

During August substantial progress was made on reviewing Mark II containments and their failure modes, including preliminary assessment of dominant risks, possible mitigation schemes, and a preliminary value/impact range. Several types of mitigation systems were selected for preliminary design and costing. These were chosen on the basis that systems having unequivocable modes of performance should be the standard against which other systems, having greater operational uncertainty, should be compared. These comparisons, and the final selection of a system for design, are expected to be accomplished during October. An addendum to this monthly report provides a summary of preliminary value/impact assessment being performed under Task 4 for the Mark II mini-study. This assessment provides an estimate of the range of costs that can be supported for mitigation, (including not only equipment but procedures and outage costs). Another addendum is a draft version of the cumulative mitigation requirements for the Limerick plant. It is by no means final at this time.

The conference on LWR severe accident evaluating was attended by W.E. Kastenberg and I. Catton. A meeting of BNL, NRC, Purdue and RDA was accomplished during the conference. A report on the meeting will be forwarded under separate cover. At the request of Dr. J. Meyer, I. Catton attended a meeting on MFlICR. His report on the meeting will also be forwarded under separate cover.

For Tasks 1 and 2 the process of data collection is well along, outlines have been made of the topical reports, and some text preparation has begun. A special list of all known mitigation proposals, devices, and systems was prepared and delivered to NRC; a copy is attached. The report outline for Task 3 is currently being revised, and procedures and outlines for Tasks 4 and 5 are being prepared.

The Master Plan outline for the project is under way, and is expected to be completed during the current period. NRC has designated GESSAR as the second of the three types of containment to be studied in Task 3. Plans are being made to visit the Perry plant during the current period.

For the purposes of our contract "mitigation" has been defined as those actions, devices, components and systems that deal with events and their consequences *after* the core is melted. This definition is quite straightforward except in the case of failure to scram. For a BWR under certain conditions (Class IV) a large steam spike could rupture the containment long before core-melt occurred. Similarly, there are some sequences (Class II) wherein the containment fails prior to core melt, but on a long time scale. We have requested written instructions on how to interpret these special cases. It is feasible to include a mitigation system for an ATWS event (probably a vent), but it has been our impression that prevention of an ATWS would be more cost-effective than mitigation, and that the Limerick plant had already been modified to reduce such events to a non-dominant risk level (the so-called 3-A fix). The significance of ATWS to mitigation will be clarified in the near future.

SECTION B: Technical Status by Tasks

Task 1. Survey of Containment Systems. This task comprises data gathering, categorization of dominant accident sequences, and evaluation of mitigation opportunities for the major types of reactor containment. The work will be organized and reported by containment type, *viz.* Mark II, Mark III, GESSAR, etc. Thus the work on Mark II containment will form a chapter both in the Mark II mini-study and in the topical report for Task 1.

a. Efforts completed: Assessment of Mark II containments and the severe accident research literature has been completed. Work on the other pressure-suppression types is well underway. Personal contact and telephone interviews have been made with research investigators in the field at

Sandia, Brookhaven, and Purdue.

b. Problems or delays: None.

c. Summary to date: Data collection mostly complete for pressure-suppression containments, well underway for other types. Review of accident sequences is complete for BWR's, just beginning for PWR's. Identification of mitigation opportunities is complete for Mark II containments, underway for other pressure-suppression types.

d. Plans for next period: Meetings have been scheduled with representatives of Sandia, Brookhaven, and Purdue to discuss and coordinate our survey with current research efforts. It is planned to finish source collection, finish assessment of Type I and Type II containments, and begin work on PWR containments. Text preparation will also begin.

Task 2. Survey of Mitigation Systems. a. This task will survey a wide range of concepts, proposals, devices, and systems for mitigating the consequences of severe accidents. These will be categorized into groups by function, and ranked according to feasibility, cost, etc. The specific devices, sub-systems, etc., to accomplish a given function, such as core-retention or heat removal, are designated as *components*. A selection of appropriate components will be made to form a mitigation *system* suitable for each major containment type as determined in Task 1.

a. Efforts completed: Tabulation of mitigation concepts applicable to Mark II containments has been completed. A complete list of all known suggestions for mitigation systems or components has been completed and submitted (see Addendum).

b. Problems or delays: None.

c. Summary to date: The information collection phase of this task is nearly complete. Assessment and ranking of the material is under way. An outline of the final report has been prepared and written text is being prepared. Information concerning mitigation components for Mark II plants has been transmitted to the other task efforts.

d. Plans for next period: Complete collection of data, continue assessment and ranking of mitigation components.

Task 3: Design and Feasibility. This task comprises selection of up to three major types of reactor containment with the approval of the Project Officer, establishing for each type the requirements for a mitigation system in view

of the dominant risks established in Task 1, and choosing a suitable combination of components as characterised in Task 2. After establishing by preliminary analysis in Task 4 that the functions chosen to be performed would likely have a suitable effect on overall risk and with enough design effort to show probable feasibility, a final selection of a mitigation system will be made with the approval of the Project Officer. Then a complete conceptual design, cost and feasibility assessment will be performed.

a. Efforts completed: Based on Task 1 results, the requirements for mitigating the residual risk in Mark II containments have been established, subject to some remaining uncertainty as to whether ATWS events are considered to have been prevented or are to be included in the mitigation category. Preliminary design and costing has been made for several possible mitigation systems (combinations of components). These are now undergoing evaluation before final selection and presentation to the Project Officer.

b. Problems or delays: None.

c. Summary to date: For Mark II containment as exemplified by the Limerick Plant, mitigation requirements (functions) have been identified, including containment heat removal, core residue capture and retention without concrete attack, and (if ATWS events are to be mitigated) some kind of venting system. Candidate components to fulfill these requirements have been selected for preliminary conceptual design and cost estimation. Separate cost figures will be generated for 1) Plants before construction begins, 2) Plants built but not yet in operation, and 3) Operational plants.

d. Plans for next period: Complete preliminary designs and assessments for Limerick, and begin final design of selected version.

Task 4: Value/Impact Analysis. This task will provide a quantitative assessment of the relative risk that can be averted by mitigating particular aspects of the containment failure. It will determine whether a proposed mitigation system is cost effective, and which components are most important.

a. Efforts completed: Collection of source documents for Value/Impact evaluation is virtually complete. The recent Boston meeting on Severe Accidents will provide additional material. Copies of the most important papers are already available. Previous Commission statements and action in the

Value/Impact evaluation is virtual y complete. The recent Boston meeting on Severe Accidents will provide additional material. Copies of the most important papers are already available. Previous Commissi n statements and action in the field has been reviewed incluing the PNL Value/Impact Handbook under development for RSR Division. A preliminary Value/Impact analysis for the Limerick plant was initiated using the BNL review of the Limerick PRA and the new PECO PRA. The results of this analysis are summarized briefly in the Addendum attached to this report.

b. Problems or delays: None.

c. Summary to date: Methodology for Value/Impact analysis of mitigation conceptual designs is being formulated, based on prior work in the field and the specific requirements of this task. A preliminary analysis for the Limerick plant has been completed.

d. Plans for next period: A working procedure for assessment will be put into preliminary operation for examination of Mark II mitigation schemes. This procedure will be discussed with NRC before final adoption.

<u>Task 5:</u> <u>Licensing Strategy Development.</u> This task will assist the NRC in utilizing the results of this project by developing suitable methodology and strategies for assessing and implementing severe accident mitigation policies.

a. Efforts completed: Collection and review is under way of suitable documents, reports, and prior action of the NRC, especially SECY-82-1-A and -82-1-B, and NUREG 0933.

b. Problems and delays: None.
c. Summary to date: Background and criteria under development.

d. Plans for next period: Continue development.

<u>Task 6:</u> <u>Consultation and Special Assignments.</u> No effort has been assigned by NRC to this task so far.

ADDENDUM TO MONTHLY REPORT

September 15, 1983

Value/Impact for Limerick Mitigation

In order to obtain a range of possible acceptable costs of mitigation for the Limerick plant, two estimates were made using the proposed trial goal of $1,000/man rem averted as follows:

ESTIMATE NO. 1

The first estimate uses the data contained in the BNL review (NUREG/CR-3028) of the Limerick PRA. Note that this data does not reflect external initiators. The Limerick PRA and the BNL review conclude that the majority of latent effects (and population dose) result from containment failure via slow overpressurization (Class I).

In Table 1, the contribution to latent fatalities for the dominant releases are shown as well as their frequency. The expected value is 0.174 latent deaths/year. Of these 0.166 latent deaths/year come from OPREL, the Class I sequences described above. Using a conservative conversion factor of 2×10^{-4} latent deaths/man rem (linear dose model, no threshold) and an allowed cost of $1,000/man rem, one obtains an upper limit cost of 33.6 million dollars.

ESTIMATE NO. 2

The second estimate uses the data contained in the new Limerick PRA which includes external events, but not the modifications to frequency and consequences contained in the BNL review. The new PRA indicates that fires and internally initiated sequences contribute about equally to latent effects (Class I slow overpressurization). Seismic events contributing to Class IV (ATWS events) are also important contributors to latent fatalities. Early fatalities still continue to be dominated by Class IV events (ATWS) but seismic vessel failure also becomes an important contributor.

Accidents initiated by fires fall into existing release categories. Random vessel failure and accidents initiated by earthquakes required new release categories.

The sequences which are affected by fires are QUX, QUV and QW, which increase the OPREL (Class I) release category significantly. The seismic initiators lead to case with various combinations of vessel and co ainment failure due to the seismic event itself. For those cases where the containment does not fail seismically mitigation is possible.

Using the data contained in the PRA, 220 man-rem/year can be averted if containment failure due to slow

overpressuriation (only) is eliminated (perfectly). At $1,000/man-rem and a 40 year life, the allowable cost is 8.8 million dollars.

DISCUSSION

The values estimated above (8-33 million) are ball-park estimates and are subject to change dramatically pending (a) the BNL review of the new PRA, and (b) new work on the Source term.

At present they are only for use as guidance in the design of mitigation. A target of 15 million dollars will be used for screening purposes.

TABLE I

Category	Frequency (per year)		Latent deaths		
OPREL	7.7×10^{5}	x	2.2×10^{3}	=	16.6×10^{-2}
R_{2a}	9.8×10^{-8}	x	2.1×10^{4}	=	20×10^{-4}
R_{2b}	2.1×10^{-8}	x	1.8×10^{4}	=	3.7×10^{-4}
R_{2c}	3.2×10^{-9}	x	1.8×10^{4}	=	0.58×10^{-4}
R_{2d}	4.3×10^{-7}	x	6.6×10^{3}	=	28×10^{-4}
C_4	1.4×10^{-7}	x	1.4×10^{4}	=	1.98×10^{-3}
$C_4\gamma'$	7.1×10^{-8}	x	1.4×10^{4}	=	9.98×10^{-4}
$C_4\gamma''$	7.1×10^{-8}	x	1.3×10^{4}	=	9.21×10^{-4}
					17.43×10^{-2}/yr

CUMULATIVE LIST OF MITIGATION REQUIREMENTS FOR MARK II

The dominant severe accident sequences for the Mark II Boiling Water Reactors result in a small number of final end states wherein the containment is breached. To consider mitigation of these end states, it is necessary to assess them in a cumulative fashion. Thus, if any of them result in a plant having a total electrical failure, then all mitigation schemes considered must work in this environment. If any of them will undergo failure by overpressure, then all systems must be able to protect against this failure, etc. On the other hand, if none of the dominant sequences include seismic failure of the containment structure, then we can assume that it is intact for purposes of mitigation, and so on. Following are a brief list of assumptions and policies on which a mitigation design might be based, followed by a tentative set of cumulative requirements for the Limerick plant, derived by assessment of its dominant failure modes. In C. are listed the end point conditions in the containment under the cumulative worst case accident sequences. A mitigation concept that could cope with these conditions would necessarily handle the lesser, more probable events.

A. Ground-rules and Assumptions for Mitigation Design

1. When the behavior of the core or other material is in an uncertain or debatable situation, the uncertainty will be avoided by designing that situation out of the sequence. If this is not possible, the uncertainties will be reduced to the maxiumum amount possible.

2. Passive action will utilized wherever possible, but where it technically is impossible or unreasonably costly, a fully independent and redundant source of energy will be used.

3. If the containment is not overpressured, it is assumed that the water in the suppression pool will not escape, although it might drain partially into the secondary containment building to the level of the lowest penetration.

4. The containment will not fail to isolate, or changes will be made to insure isolation on demand.

5. An intact containment always presents less risk than an opened one.

6. A system that segregates and confines radioactive materials into a definite, enclosed region presents less human risk than one that spreads it over several regions.

B. Cumulative List of Mitigation Requirements.

1. If risk assessment indicates that early containment failure from an ATWS is a dominant (>1%) part of total risk, then a vent system of some type is required.

2. Reliable, redundant cooling of the containment is required even though there is no electric power.

3. The molten core debris must be provided an unequivocable pathway to a location where it can be retained and cooled indefinitely.

C. Assumed Initial Conditions at Time of Meltdown.

1. All electric power has been lost, both on-site and off-site.

2. The suppression pool has been heated by a turbine trip from full power. If an ATWS has occurred, the pool is saturated at the vent pressure.

3. The normal and emergency core-cooling systems are inoperative.

4. The emergency heat removal system is inoperative.

5. The core has boiled dry and is in the process of melting its way through the bottom of the vessel.

6. More than 50 tons of molten steel will accompany the core into the sub-vessel area.

7. All the zirconium in the core has reacted to form hydrogen.

After further refinement and developent of these ground rules and conditions, they will be used to specify the design of specific mitigation systems under Task 3.

PROPOSED SEVERE ACCIDENT MITIGATION SYSTEMS

The following list is fairly comprehensive as to types of proposed remedies, but does not attempt to include every variation, modification, and repetition within each type. No classification was made as to feasibility, effectiveness, or cost of the proposed systems.

I. CORE RETENTION DEVICES

1. Water-cooled crucible: A metallic container fitted with a water jacket and placed to intercept molten core material that has escaped from the reactor vessel or from the containment. In one version the crucible is retrofitted to an operating plant by tunneling below the basemat, and cooled by passive thermal siphons.

2. Flooded thoria rubble bed: A bed of refractory pebbles is placed on the floor of the reactor cavity, with water circulating through the bed.

3. Water-cooled refractory tiles: Similar to the pebble bed but consisting of interlocking tiles with cast-in water passages.

4. Pebble-bed covering cooling coils: A metallic piping system with a pumped water supply, placed in the bottom of the reactor cavity and covered with high-density refractory pebbles.

5. High-alumina cement covering cooling coils: A cast-in-place cement liner for the reactor cavity, with imbedded cooling coils.

6. Magnesium dioxide covering cooling coils: Cooling coils covered with interlocking magnesia refractory brock.

7. Zirconium dioxide covering cooling coils: As above with a different refractory brick.

8. Graphite covering cooling coils: Cooling coils covered with graphite or carbon brick. Sometimes with an outer cover of steel to prevent water contact.

9. Borax bath: A thick layer (12 ft) of borax bricks sealed in stainless steel, covering the bottom of the reactor cavity.

10. Heavy metal bath (lead, uranium, or copper): Cooling coils at the bottom of the reactor cavity, covered with a foot or so of lead bricks, or other metal. The lead will melt and transfer heat to the coils, but remain in place since it is denser than UO_2.

11. Iron oxide: A layer of iron oxide over cooling coils has been proposed, with the purpose of diluting the urania to lower its viscosity and increase volume and heat transfer surface.

12. Basalt concrete and basalt rubble bed: Basalt is soluble in molten urania, and the intention is to provide a dilution of the core material.

13. Sand core retention system: A very large mass of sand is provided below the containment building to absorb the heat of the core material and disperse it over a large volume.

14. Iron core retention system: A large mass of iron is provided to receive the core material and dissipate its heat.

15. Flooded cavity: Water is added to the reactor building to flood the entire cavity up to the vessel and even above it, in the hope that the core material will be kept dispersed enough to remain quenched.

16. <u>Other active cooling systems</u>: A number of special jackets and piping system in and around the reactor vessel have been proposed, with the intention of retaining the molten core within the reactor vessel.

II. OVERPRESSURE CONTROL FROM HYDROGEN OR HYDROGEN BURNING

1. <u>Oxygen exclusion</u>: The containment is operated with an atmosphere of nitrogen or carbon dioxide, or even vacuum.

2. <u>Oxygen removal</u>: Oxygen is removed from the containment when core damage is detected, using a combustion system or chemical absorbant.

3. <u>Oxygen dilution</u>: The oxygen content of the containment is diluted below the flammable limit with Halon gas, water fog or mist, foams, or sprays.

4. <u>Igniters</u>: Glow plugs or spark igniters are placed throughout the containment to burn hydrogen before it reaches an explosive concentration.

5. <u>Fans</u>: Rapid mixing of the containment air is proposed to prevent local accumulation of explosive hydrogen mixtures.

III. OVERPRESSURE CONTROL FROM ATTACK ON CONCRETE

1. <u>Special concrete composition</u>: The reactor cavity and basemat would be made with special concrete that does not release much noncondensible gas when attacked by core debris.

2. <u>Thin basemat</u>: Make the central part of the basemat thin to promote rapid escape from the containment building.

IV. OVERPRESSURE CONTROL BY VENTING THE CONTAINMENT BUILDING

1. <u>Non-filtered vent</u>: The containment is vented through a tall stack when it reaches a dangerous pressure.

2. <u>Vent to receiver</u>: The containment venting is connected to another large, closed building to provide a larger total expansion volume and greater cooling. A companion reactor containment has been proposed for this use. The receiver could also be an inflatable building or balloon, kept normally empty.

3. <u>Vent to a condenser-filter</u>: A large variety of condenser filter systems have been proposed, such as sand beds, gravel beds, water-sprayed gravel beds, scrubbers, gravel/sand, water pools, sand filters, charcoal filters, chemical scrubbers, all in various combinations.

V. OVERPRESSURE CONTROL BY CONTAINMENT HEAT REMOVAL

1. <u>Heat pipes</u>: Passive devices that absorb heat from vapor or pool space inside containment and release it externally through an evaporation-condensation exchange with an internal fluid.

2. <u>Modified heat pipes</u>: Heat pipes having separate liquid return passages, heat pipes with ganged penetrations, and variable gas-controlled heat pipes.

3. <u>Heat exchangers</u>: Standard cooling coils acting as condensers or pool coolers, with pumped external cooling fluid.

4. <u>Spray coolers</u>: Pumped sprays inside containment, combined with heat exchangers in the loop.

5. <u>Fan coolers</u>: Circulating fans combined with heat exchangers to increase thermal transfer from containment vapor space.

6. <u>Secondary suppression pool</u>: Provide a larger secondary suppression pool to increase heat capacity of system.

7. More reliable residual heat removal system: Increase the redundancy and ruggedness of the residual heat removal system.

VI. CONTAINMENT PROTECTION AGAINST MISSILES

1. Missile shields: Various structures designed to protect the containment penetrations or walls against flying debris or thrashing pipes inside the containment.

VII. SPECIAL CONTAINMENT STRUCTURES

1. Underground siting: Location of the containment vessel in an underground cavern or excavated pit, completely isolated from the external environment.

2. Berm shield: Partially underground containment building, protected by a earthen wall or berm. Sometimes a gravel bed is included for a filtered vent pathway.

3. Double containment: A second strong containment building surrounding the original containment has been proposed.

4. Strength improvements: For improving the pressure rating of an existing containment building, wrapping with wire, adding steel ribs, etc., have been proposed.

5. Increased volume: Increase the free volume of the containment building on new reactors.

6. Strengthen safety system: Make the essential safety systems more rugged by means of armor, bunkers, and heavier construction.

VIII. FISSION PRODUCT REMOVAL SYSTEMS

1. Containment spray systems: It has been proposed to decontaminate a containment building post accident with an elaborate spray system to wash down the interior with special solutions, and a treatment system to remove the contaminants from the solutions for use.

2. <u>Gas treatment system</u>: Provide a special recirculating treatment system to remove fission products from the containment gas volume.

IX. PORTABLE OR ADAPTIVE RESPONSES

1. <u>Pumps</u>: Use of portable pumps, fire trucks or fire boats to add water to the containment, keep a gravel vent bed wet, or for other requirements.

2. <u>Earthmovers</u>: Use of bulldozers, etc., to build up protective shields or berms around contaminated buildings, prevent flood erosion, etc.

R & D Associates
August 24, 1983

Monthly Project Status Report 15 August, 1983

Report No. RDA-MR-127300-001
Period Covered: June 27 through July 30, 1983
Name of Program: Severe Accident Mitigation Systems
Contract Number: NRC-03-83-092
Start Date: June 27, 1983 Completion: 27 months

SECTION A: Overall Summary of Project Status

Technical: In this first month of operation, active work has begun on Tasks 1-5. (Task 6 is done only on specific assignments). At NRC request, the first specific plant type to be studied is the Mark II BWR, as exemplified by the Limerick plant. We have been requested to make a special rapid response on our study of this type plant by January 1, 1984. Collection of data and reports is well underway, and the Limerick containment system has been visted. Besides the work on Limerick, present effort is directed at revisions and completion of the master plan for the project.

Financial: As of July 30, 1983, funds expended amounted to \$ 31,202 (6.5% of total). Funds obligated amounted to \$ 58,349 (12.1% of total).

SECTION B: Technical Status by Tasks

Task 1. Survey of Containment System. a. Efforts completed: A working outline of the final topical report has been developed, and a substantial portion of the source documents located and ordered. About third are already on hand. The significant failure modes of Type II containments have been identified, and mitigation requirements set out.
b. Problems or delays: None.
c. Summary to date: Data collection well underway, assessment just beginning.
d. Plans for next period: Finish assessment of Type II containments, complete source collection.

Task 2. Survey of Mitigation Systems. a. Efforts completed: Tabulation of mitigation concepts applicable to Mark II containments has been completed, and collection has begun of a complete list of all known suggestions for mitigation systems or components as a special assignment.
b. Problems or delays: None.
c. Summary to date: Information collection underway.
d. Plans for next period: Continue collection and assessment of mitigation literature.

Task 3: Design and Feasibility. Efforts completed: Preliminary assessment of feasibility has been made for several Type II mitigation concepts. These are to be given

a rough costing and then subjected to preliminary Value/Impact analysis before final selection.

b. Problems or delays: None.
c. Summary to date: Several Type II mitigation concepts are undergoing evaluation; others will be added.
d. Plans for next period: Complete preliminary designs and assessments, and begin final design of selected version.

Task 4: Value/Impact Analysis. a. Efforts completed Collection of source documents for Value/Impact evaluation is under way. Previous Commission statements and action in the field has been reviewed including the PNL Value/Impact Handbook under development for RSR Division.
b. Problems or delays: None.
c. Summary to date: Methodology for Value/Impact analysis of mitigation conceptual designs is being formulated, based on prior work in the field and the specific requirements of this task.
d. Plans for next period: A working procedure for assessment will be put into preliminary operation for examination of Type II mitigation schemes. This procedure will be discussed with NRC before final adoption.

Task 5: Licensing Strategy Development. a. Efforts completed: Collection and review is under way of suitable documents, reports, and prior action of the NRC, especially SECY 1-A and 1-B, and NUREG 0933.
b. Problems and delays: None.
c. Summary to date: Background and criteria under development.

Task 6: Consultation and Special Assignments. No effort assigned by NRC to this task so far.

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

AUG 25 1983

Mr. Steven Sholly
Technical Research Associate
Union of Concerned Scientists
1346 Connecticut Avenue, N.W.
Washington, DC 20036

IN RESPONSE REFER
TO FOIA-83-432

Dear Mr. Sholly:

This is in response to your letter dated July 28, 1983, in which you requested, pursuant to the Freedom of Information Act, documents produced by Sandia National Laboratory and/or their contractors under the NRC sponsored "Severe Accident Risk Reduction Program."

Appendix A is a listing of documents responsive to your request. These documents are being placed in the NRC Public Document Room in FOIA file folder 83-432 in your name.

This completes NRC's action on your request.

Sincerely,

J. M. Felton, Director
Division of Rules and Records
Office of Administration

Enclosure: As stated

UNION OF CONCERNED SCIENTISTS

1346 Connecticut Avenue, N.W. • S. 1101 • Washington, DC 20036 • (202) 296-5600

28 July 1983

FREEDOM OF INFORMATION ACT REQUEST
FOIA-83-432
Rec'd 8-1-83

Mr. J. M. Felton, Director
Division of Rules and Records
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Felton:

Pursuant to the Freedom of Information Act, please make available for public inspection and copying at the Commission's 1717 H Street Public Document Room copies of documents in the following categories:

A. All documents produced by Sandia National Laboratories and/or their contractors under the NRC-sponsored "Severe Accident Risk Reduction Program" (SARR). This request specifically includes draft reports, papers prepared for presentation at technical society meetings (e.g., ANS/ENS meetings), and memoranda communicating results anad conclusions of this work to the NRC. This request also specifically includes documents concerning value-impact and/or cost-benefit analyses of risk reduction measures analyzed in the SARR Program.

I recognize that this request may involve a large number of reports. At present, however, I am unable to refine the request further due to a lack of information on the NRC contract number, the identities of the Sandia researchers performing the work, or the identity of the NRC Staff Technical Monitor for the project. I am, however, willing to discuss this request with your staff to help avoid unnecessary document search efforts.

Should you or your staff have any questions regarding this request, please do not hesitate to contact me at UCS's Washington, D.C., office at 296-5600. Your cooperation in responding to this request is appreciated.

Sincerely,

Steven C. Sholly

Steven C. Sholly
Technical Research Associate

APPENDIX A

1. Evaluation of Severe Accident Safety System Value Based on Averted Financial Risks, SAND83-0443C.

2. Evaluation of the Sensitivity of Reactor Risks to Uncertainties, SAND83-0855C.

3. Severe Accident Risk Reduction Program, SAND82-2141C.

4. Ltr: Benjamin to Cunningham, August 30, 1982, w/following attachments:

   a. Value - Impact Analysis of Severe Accident Prevention and Mitigation Systems, 8/82.

   b. Risk Reduction Analysis of Severe Accident Prevention and Mitigation Systems, SAND82-1697C.

   c. Benjamin slides, Sept. 2, 1982.

5. Benjamin slides, LA ANS Mtg., June 9, 1982.

6. Ltr: Benjamin to J. B. Van Erp. January 29, 1983.

7. Risk and Systems Interaction Analysis of Severe Accident Prevention and Mitigation Systems, SAND82-0400A.

8. Ltr: Benjamin to E. N. Cramer, January 12, 1982.

9. Note: Hatch to Kolaczkowski, July 25, 1983, ASEP products expected by the SARR Program in FY83 and Beyond.

10. Ltr: Griesmeyer to distribution, July 22, 1983, MARCH Screening Sensitivity study.

11. General SARRP viewgraphs, 7/83.

12. SARRP Approach to Risk Benchmarking, May 19, 1983.

13. Ltr: Benjamin to Cunningham, April 20, 1983, SARRP Plan for the Accomplishment of Phase I objectives.

14. Presentations; NRC Management Review of Sandia Severe Accident Programs.

15. SARRP Phase I Report Outline, 9/82.

16. Design Considerations for Implementing a Vent - Filter System at the Barseback Nuclear Power Plant, Johansson et. al., August 1982, Chicago, ANS meeting.

APPENDIX A

17. Ltr: Benjamin to Cunningham, May 19, 1982, Severe Accident Uncertainty Analysis.

18. Presentation: NRC review of SARRP, etc., March 19, 1982.

19. Ltr: Benjamin to P. B. Bleiweis, January 29, 1982, review and comments on Bleiweis report.

20. SARRP presentation by A. S. Benjamin, October 21, 1981.

21. Cost-Benefit Considerations for Filtered-Vent Containment Systems, 17th DOE Nuclear Air Cleaning Conference.

22. Risk Assessment of Filtered-Vented Containment Options for a BWR Mark III containment, SAND82-0403C.

23. Probabilistic Risk Assessment of Filtered-Vent Containment Systems: Mark I BWR, Abstract from ANS Transactions, 1981 summer meeting.

24. Presentation by A. S. Benjamin on PRA of FVCS for Mark I BWR, 1981 ANS summer meeting.

25. Ltr: Benjamin to N. J. Diaz, circa November 1980, summary of paper for 1981 ANS summer meeting.

26. Filtered-Vent Containment Systems, IAEA-CN-39/103.

27. Presentation on FVCS, A. S. Benjamin, April 19, 1982.

28. FVCS Program - Grand Gulf Results, F. T. Harper, April 19, 1982.

29. Ltr: A. L. Carter, Holmes and Narver to Benjamin, January 13, 1982.

30. Ltr: Benjamin to M. W. First, Harvard, July 20, 1981.

31. Ltr: Benjamin to Rowsome, March 12, 1981.

32. Ltr: A. L. Carter to H. C. Walling, SNL, January 27, 1981, Cost Estimate of Toroidal Suppression Pool.

33. FVCS-BWR Mark I presentation by Benjamin, circa 1981-1982.

34. Issues affecting the Feasibility and Effectiveness of Vent-Filtered Containments, SANS79-1139A.

APPENDIX A

35. Program Plan for the Investigation of Vent-Filtered Containment Conceptual Designs for Light Water Reactors, NUREG/CR-1029, October, 1979.

36. Effect of Containment Venting on the Risk of LWR Meltdown Accidents, NUREG/CR-0138, June, 1978.

37. A Value-Impact Assessment of Alternate Containment Concepts, NUREG/CR-0165, June, 1978.

38. Value-Impact Comparison of Alternate Containment Designs, SAND77-1103C, November, 1977.