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A REVIEW OF LIGHT-WATER REACTOR SAFETY STUDIES. VOLUME 3 OF THE FINAL REPORT ON HEALTH AND SAFETY IMPACTS OF NUCLEAR, GEOTHERMAL, AND FOSSIL-FUEL ELECTRIC GENERATION IN CALIFORNIA

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Author

Nero, A.V.

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Energy and Environment Division

A Review of Light-Water Reactor Safety Studies

A.V. Nero and M.R.K. Farnaam

January 1977

Lawrence Berkeley Laboratory University of California/Berkeley

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Energy and Environment Division
Lawrence Berkeley Laboratory
University of California
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Vol. 1: "Health and Safety Impacts of Nuclear, Geothermal, and Fossil-Fuel Electric Generation in California: Overview Report," by the entire staff, Lawrence Berkeley Laboratory Report LBL-5924. Includes "Executive Summary" for the project.

Vol. 2: "Radiological Health and Related Standards for Nuclear Power Plants," by A.V. Nero and Y.C. Wong, Lawrence Berkeley Laboratory Report LBL-5285.

Vol. 3: "A Review of Light-Water Reactor Safety Studies," by A.V. Nero and M.R.K. Farnaam, Lawrence Berkeley Laboratory Report LBL-5286.

Vol. 4: "Radiological Emergency Response Planning for Nuclear Power Plants in California," by W.W.S. Yen, Lawrence Berkeley Laboratory Report LBL-5920.

Vol. 5: "Control of Population Densities Surrounding Nuclear Power Plants," by A.V. Nero, C.H. Schroeder, and W.W.S. Yen, Lawrence Berkeley Laboratory Report LBL-5921.

Vol. 6: "Health Effects and Related Standards for Fossil-Fuel and Geothermal Power Plants," by G.D. Case, T.A. Bertolli, J.C. Bodington, T.A. Choy, and A.V. Nero, Lawrence Berkeley Report LBL-5287.

Vol. 7: "Power Plant Reliability-Availability and State Regulation," by A.V. Nero and I.N.M.N. Bouromand, Lawrence Berkeley Laboratory Report LBL-5922.

Vol. 8: "A Review of Air Quality Modeling Techniques," by L.C. Rosen, Lawrence Berkeley Laboratory Report LBL-5998.

Vol. 9: "Methodologies for Review of the Health and Safety Aspects of Proposed Nuclear, Geothermal, and Fossil-Fuel Sites and Facilities," by A.V. Nero, M.S. Quinby-Hunt, et al., Lawrence Berkeley Laboratory Report LBL-5923.

A REVIEW OF LIGHT-WATER REACTOR SAFETY STUDIES

ABSTRACT

This report summarizes and compares important studies of light-water nuclear reactor safety, emphasizing the Nuclear Regulatory Commission's Reactor Safety Study, work on risk assessment funded by the Electric Power Research Institute, and the Report of the American Physical Society study group on light-water reactor safety. These reports treat risk assessment for nuclear power plants and provide an introduction to the basic issues in reactor safety and the needs of the reactor safety research program. Earlier studies are treated more briefly. The report includes comments on the Reactor Safety Study. The manner in which these studies may be used and alterations which would increase their utility are discussed.

TABLE OF CONTENTS

Abbreviations used in the report

AEC - Atomic Energy Commission

APS - the American Physical Society

BWR - boiling-water reactor

ECC(S) - emergency core cooling (systems)

EPA - Environmental Protection Agency

EPRI - the Electric Power Research Institute

LOCA - loss-of-coolant accident

LOFT - loss-of-fluid test

LWR - light-water reactor

MWe - megawatts electric

MWt - megawatts thermal

NRC - Nuclear Regulatory Commission

PWR - pressurized-water reactor

RSR - Division of Reactor Safety Research (Nuclear Regulatory Commission)

1. INTRODUCTION

Nuclear power constitutes one of the most important energy sources available to us in the near future. However, dependence on nuclear power requires the handling of substantial amounts of material that is substantially different from the materials employed in other energy technologies: much of the material of nuclear power can decay radioactively. Examination of the potential for this material to have significant impacts on the public health is important, both because of the quantity of radioactive material handled in a nuclear power system and because of the availability - in certain instances - of large amounts of energy to initiate a release of radioactivity or to aid its dispersal.

A complete commercial nuclear power system consists not only of reactor power stations, but also of support facilities, in particular those which produce the fuel for insertion into the reactors and those which handle or process the fuel after removal from the reactor. Insofar as impacts on the public health are concerned, each type of facility may be characterized by its routine radioactive emissions and by its potential for large accidental releases. Both routine and accidental potentials have received a substantial amount of attention. It is the potential for accidental release that is the focus of the present discussion. Of the nuclear facilities which are now prevalent, power reactors themselves have drawn most attention as potential sites for accidental releases. This is natural, not only because of the concentration of radioactivity at the reactor sites, but because of the availability of large amounts of energy in reactors and associated equipment. The reactor* makes energy available for production of electricity, and this same energy may become involved in a radioactive release. However, it is understood that physical laws do not allow the energy to become available for a cataclysmic instantaneous release characteristic of a bomb.

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*Note that, for the sake of precision, we have distinguished the reactor, the device which produces thermal energy, from the nuclear power plant, which has electrical energy as its output.

A number of important studies of the safety aspects (actually, the potential risk) of nuclear reactor power plants have been performed. The purpose of this report is to discuss those studies. The studies of particular interest are those which deal with light-water reactor (LWR) power plants, either of the pressurized-water reactor (PWR) or boiling-water reactor (BWR) variety. These are the types of power plants that are presently being operated or constructed in the United States for the commercial generation of electric power. Other types of reactors are used elsewhere, and a variety of types, such as the liquid-metal fast breeder reactor, are under development or investigation. (See Ref. 1 for a description of some of the more important reactor types.) Because of the prevalence of light-water reactors in this country, most studies of overall nuclear reactor safety have focussed on this type, and it is these studies which are discussed in this report.

The rate of construction of LWR power plants has a direct bearing on the overall public risk from the nuclear power system. As of June 30, 1976, a total of 60 nuclear power plants, with a total generating capacity of 41,000 megawatts, were authorized to operate in the United States, representing 8% of total installed electrical capacity. An additional 178 (with 196,000 megawatts capacity) were under construction, on order, or planned.[2] Further, it is anticipated that additional light-water reactors will be built, contributing to a total light-water reactor generating capacity of up to 800,000 megawatts or 800 gigawatts* by the year 2000.[3] (A gigawatt, which equals 1000 megawatts, is a natural unit of generating capacity, since the nuclear power plants currently being ordered have approximately one gigawatt of electrical output). The total public risk is related to the number of LWRs built. However, the relationship may not be simple since LWRs designed in the future may have features which differ significantly, from a safety point of view, from the reactors currently operating. Reactor safety studies have emphasized reactors with features similar to those presently operating or being built, and their results may not be directly applicable to future LWRs. However, these studies have some value with respect to future reactors, since many of the systems will be similar to those of current reactors and, furthermore, the methodologies used in these studies may be applicable to new designs.

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*Our examination of reactor safety studies does not treat the question of whether this number of reactors can or should be built.

Reactor safety studies may have two basic purposes, closely related. The first is to assess the risk presented to the public by reactors as presently designed and operated. Assessment of this risk may be used in two different fashions; one is to understand the overall risk that the nuclear power system presents to the entire society and another is to point out the risks imposed on the population near to a specific power plant. For example, risk assessment may be applied in the process of making overall decisions on the development of nuclear power. On the other hand, risk assessment may be applied in choosing a site for a particular power plant. These uses may be related, respectively, to the overall acceptability of nuclear power to society and to the acceptability of a particular plant to the population which would surround that plant.

The second basic purpose of reactor safety studies, sometimes inextricable from the first, is to point out those areas of reactor design or operation where improvements could significantly decrease the risk posed by nuclear power plants. Examination of the safety aspects of a particular reactor or reactors in general can be part of the design or licensing process. Such examination routinely takes place in these processes, but there are, in addition, cases of independent studies of the overall safety of nuclear reactors. These studies can examine safety systems or safety research programs for their effectiveness or they can systematically examine the contribution of failure modes to the impact of nuclear power on the public. Either type of study (and it is sometimes hard to distinguish the two) can be used as a basis for improvements in nuclear power plants.

It is also worth noting that the same approach and specific techniques that are used in studies of the safety of nuclear reactors may be applied to other types of facilities. These include the other facilities in the nuclear fuel cycle, examples of which are fuel fabrication plants, fuel reprocessing plants, and waste disposal facilities, and also facilities for other types of energy production. However, most attention has been directed to the specific case of nuclear power plants and we will restrict our attention to this case.

A further specification must be made. It is not our intention to discuss studies which have been aimed at very narrow aspects of reactor systems or studies of the safety of particular reactors. Thus, for example, we do not discuss studies of corrosion of steam generator tubes or design of neutron flux measurement instrumentation. Further, we do not discuss the safety analysis reports presented in support of license applications for specific nuclear power plants. Instead, we concentrate on studies intended to treat generically the overall safety of light-water reactor power plants. These studies are not large in number, unless one includes the innumerable comments on the primary studies, but the few existing studies treat many of the important aspects of the safety of nuclear reactors.

For the most part, studies of overall nuclear reactor safety have been prompted by the need for information in the public arena on the risk presented by nuclear power. Recently, studies have tended to examine, in addition, steps to be taken to reduce this risk through improvements in the design and operation of power plants. Keeping in mind that risk assessment and risk reduction are the two purposes of reactor safety studies, we can still distinguish various studies by their approaches to these questions. Some distinguishable approaches are:

1. Some version of a possible or "maximum possible" accident may be examined. This would postulate the release of some fraction of the radionuclide inventory of a nuclear power plant, then make some assumptions about population distributions and radionuclide transport and about the effect of exposures to radiation. These assumptions may be combined to yield some selection of: near-term deaths or illness, long-term deaths, illness, or genetic effects, and property damage.

2. Instead of adopting the "postulated" accident approach above, a study may attempt to examine on some basic mechanistic grounds the probability and effects of various types of accidents, thereby actually assessing the risk in terms of the same categories as just mentioned: deaths, illness, and so on. A useful examination of the risk from nuclear power based on this approach requires a much more detailed understanding of the design and operation of nuclear power plants and of the physical processes of release and transport than is needed in a postulated accident approach.

3. A study may examine the safety systems and procedures associated with nuclear power plants with the intention of suggesting steps to be taken to reduce the overall risk from such power plants. Such suggestions may refer either to proposed reactor safety research programs or to possible changes in systems or procedures in plants now being planned. The examination of safety systems and procedures may rely on qualitative "engineering judgment"

or may involve a detailed methodology yielding quantitative results. It is often difficult, in any given study, to separate this process of searching for possibilities for risk reduction from the general process of risk assessment, emphasized in the two approaches above.

A common feature of all such studies of reactor safety is that they are predictive, rather than retrospective. We have had no experience with large accidental releases of radioactivity. Furthermore, by comparison with what the future appears to hold, our experience with the routine operation of commercial nuclear power plants is not large. All our experience of about three hundred years of large reactor operation will be equalled on an average yearly basis toward the end of this century, according to many projections.

Until recently, a study performed 20 years ago for the Atomic Energy Commission (AEC) by Brookhaven National Laboratory stood as the primary basis for our understanding of the potential risk associated with the operation of nuclear power plants. Because this study, entitled "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Plants" (AEC Report, WASH-740),[4] adopted approach 1 above, it did not attempt to assess the overall risk from such power plants. Rather, it dealt with the consequences of postulated accidental releases of radioactivity. More recent studies have dealt with the detailed questions implicit in the second and third approaches above, and these are the studies emphasized in this report. Specifically, several important studies or groups of studies have been performed, to some extent concurrently, since 1972. Their results have been published within the last two years. The studies emphasized in this report are:

1. The Reactor Safety Study,[5] performed under the direction of Prof. Norman C. Rasmussen for the Atomic Energy Commission and the Nuclear Regulatory Commission. Also referred to by its AEC report number, WASH-1400, this report attempts to assess the overall risk presented by the first 100 nuclear reactors constructed, by a detailed examination of possible accident sequences and their corresponding probabilities and consequences. As such, this study uses the second approach described above.

2. Studies on probabilistic analysis funded by the Electric Power Research Institute and performed by Science Applications, Inc.[6] To a large extent, these studies do not stand as an independent assessment of the overall risk from nuclear power plants, but rather extend the methodologies utilized in WASH-1400.

3. The Report to the American Physical Society by the Study Group on Light-Water Reactor Safety.[7] This study took a broader approach to examining accident causes and consequences, but did not use the probabilistic methods of WASH-1400; it also examined the light-water reactor safety research program and made recommendations aimed at reducing the risks associated with LWR power plants. The APS report also comments on certain aspects of the 1974 draft of WASH-1400.

These studies constitute an important contribution to our understanding of the risks from nuclear power and represent some basis, although incomplete, for societal decisions on nuclear power, whether on the broad question of the extent of development of nuclear power or on narrower question of how to make improvements in the safety of nuclear reactors. Section 2, the bulk of the present report, is devoted to an examination of these studies. Other studies which have been performed and the studies which are now being pursued are discussed in Section 3. Comments on the three above studies are also included in this section. Section 4 treats the question of how such studies may be used in judging or improving the safety of nuclear reactors.

Before beginning our discussion of specific studies of reactor safety, it is worth commenting briefly on the general safety design of reactor power plants, at least to the extent of mentioning the primary safety systems. The first action taken in response to an abnormality is to shut down the nuclear chain reaction by insertion of a set of control rods. One means of regulating the power level of a reactor during operation is to use such rods, which contain neutron absorbers and are used to rob the chain reaction of enough neutrons to maintain the reaction rate, and hence the energy production rate, at the desired level. A special set of fast-acting shutdown control rods is present to shut down the reactor under emergency conditions. Insertion of these control rods does not guarantee that all is secure, because even after shutdown of the chain reaction a substantial amount of heat is produced in the reactor core, primarily as a result of decay of radioactive species produced during the course of the chain reaction. Because of this continuing generation of heat, cooling of the fuel assemblies must be continued to prevent melting of the fuel and subsequent release of radioactivity from the reactor vessel. Under ordinary conditions, the same cooling systems that are used for heat removal during operation of the reactor could be used for cooling subsequent to shutdown, but additional systems exist to assure availability of cooling.

The emergency core cooling system (ECCS) is such a system and is one of the "engineered safety features"* designed to prevent or mitigate releases of radioactivity. Other engineered safety features are, for example, containment sprays and heat removal systems (designed to remove, respectively, radioactivity and heat from the containment atmosphere) and the containment itself, either primary or secondary (designed to isolate the reactor system from the general environment).

Once an event occurs to initiate a possible accident sequence, these features are designed to prevent core melting or, should that occur, to limit the release of any radioactivity. Any quantitative analysis of release probabilities must take into account the probable success or failure of these systems, as well as the probability of the initiating event itself. Moreover, more general approaches to the question of reactor safety, even if qualitative rather than quantitative, often address the question through an examination of the engineered safety features' design adequacy and operational reliability. More detailed descriptions of reactor systems and safety systems are contained in references 1, 5, and 7 and, additionally, in safety analysis reports submitted in support of licensing applications.

Two concepts deserve special mention before proceeding. The first, that of a "common mode" failure, refers to multiple systems failure resulting from a single more fundamental (either component or human) failure. Such events may circumvent any redundancy incorporated into the safety systems design. The second notion to be mentioned is the possibility of alternative points of view in modeling reactor-related phenomena. A "realistic" model would use the best available information to arrive at a result which represents the physical situation as well as possible. However, where large uncertainties exist, it is often prudent to make "conservative" assumptions, which take a pessimistic view at points of uncertainty, in an attempt to assure that the results err in such a way as to protect the reactor and the public. This distinction is particularly important for the models used to evaluate ECCS performance.

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*The "engineered" safety features, which are added onto the basic reactor concept, are to be distinguished from "intrinsic" safety features which are characteristic of the basic concept itself. An example of the latter is the fact that, should cooling water be lost from the reactor vessel in a light-water reactor, the chain reaction would be shut down because water was no longer available to moderate neutrons to energies where they have a high probability of causing fission.

References for Section 1

1. A. V. Nero, "A Guidebook to Nuclear Reactors", Lawrence Berkeley Laboratory Report LBL-5206, May 1976 (available from NTIS).

2. U. S. Energy Research and Development Administration, July 1976.

3. U. S. Energy Research and Development Administation, "A National Plan for Energy Research, Development, and Demonstration...", USERDA report ERDA-76-1, 1976.

4. "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants", USAEC Report WASH-740, 1957 (available from NTIS).

5. "Reactor Safety Study: An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants", U. S. Nuclear Regulatory Commission Report WASH-1400 (NUREG-75/014), in 9 volumes, October 1975 (available from NTIS).

6. "Summary of the AEC Reactor Safety Study (WASH-1400)", EPRI 217-2-1, April 1975; "Generalized Fault Tree Analysis for Reactor Safety", EPRI 217-2-2, June 1975; "Critique of the AEC Reactor Safety Study (WASH-1400)", EPRI 217-2-3, June 1975; "Probabilistic Safety Analysis", EPRI 217-2-4, July 1975; "User's Guide for the Wam-Bam Computer Code", EPRI 217-2-5, January 1976. All of these are Electric Power Research Institute Reports, prepared by Science Applications, Inc., and available from NTIS.

7. "Report to the American Physical Society by the Study Group on Light-Water Reactor Safety", Reviews of Modern Physics 47 Supplement No. 1, S1-S124 (Summer, 1975).

2. STUDIES EXAMINED IN DETAIL

2.1 The Reactor Safety Study, WASH-1400

2.1.1 General Background and Objective

Prior to 1970, no study of the safety of light-water reactors had attempted to perform a quantitative analysis of the probability of accidents and the size of releases. Since such an analysis is the first step in a quantitative assessment of the overall risk from nuclear reactors, it is fair to say that no such assessment had been performed by the beginning of this decade. However, there had been substantial interest in such an assessment, largely because of the growing public awareness of the possibility of large accidental releases of radioactivity. Moreover, because of developments in recent years in the areas of decision analysis and reliability analysis, it appeared that techniques were available to attempt a quantitative assessment of the risk from nuclear power.

As a result, the Atomic Energy Commission began a study, under the direction of Norman C. Rasmussen, Professor of Nuclear Engineering at Massachusetts Institute of Technology, whose main purpose was to attempt such a quantitative assessment, using these new techniques. Although there was some lack of confidence in the reliability of these techniques for analysis of low-probability events, it was felt that their applicability should be tried. The study began in 1972, lasted 3 years, and expended approximately $4 million, making it the most substantial study of overall reactor safety to date. The study was designed to examine the risk from the current generation of light-water reactor power plants. It chose an existing PWR (Surry 1, 778 MWe* output capacity) and an existing BWR (Peach Bottom 2, 1065 MWe*) as sources for the detailed engineering information necessary for the quantitative evaluation of accident probabilities and release sizes. Sets of meteoreological conditions and population distributions typifying 68 reactor sites (at which the first 100 reactors were being operated or constructed) were then used in the calculation of population exposures (resulting in a spectrum of death and

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*These were the largest reactors of their types, about to begin operation as the study commenced.

disease effects) and property contamination (resulting in loss of value). More details are given below. The important point to be made here is that the study was intended to make a realistic estimate, based on actual physical and engineering understanding, of the risks to the public from accidents in nuclear power plants of the type now in use, which are based on pressurized-water reactors (PWRs) and boiling-water reactors (BWRs). The study then went on to compare these risks with those presented by other accidents, either man-made or natural. This attempt at a realistic estimate distinguishes this study from previous analyses, such as that of WASH-740,[2] which attempted to make a conservative estimate by postulating releases and calculating their effects in a rather simple manner.

A draft report on the study was issued in August 1974, and the final report (WASH-1400) was issued in October 1975. In each case, the documentation consisted of a main report, supported by ten appendices documenting in detail the technical work of the study. A very short and, to some extent, simplistic "executive summary" was added in each case. The final report added an eleventh appendix discussing the comments which were received on the draft report and the extent to which the final report responded to these comments. See Table 2-1.

2.1.2 Summary of Methodology and Results of WASH-1400

The approach employed in WASH-1400 is based on a straightforward understanding of the concept of risk. That is, possible accident sequences must be identified, and the probability of occurrence and list of consequences associated with each possible sequence must be calculated. The probabilities may be combined with the consequences to obtain the total risk associated with one or many nuclear power plants.

The original intention of the study was to adapt reliability techniques, based on "fault tree" analysis, to find the probability of failure of reactor safety systems. It was found that developments in this area alone could not satisfactorily treat the complex sequences possible in reactor safety analysis, and that it was first necessary to apply a systematic methodology simply to identify the possible sequences. The latter methodology, previously applied largely in the area of decision analysis, develops "event trees" to identify the accident sequences which may result from initiating events known to be possible. The combination of event trees and fault trees could then be used

TABLE 2-1 WASH-1400 APPENDICES*

Appendix I-Accident Definition and Use of Event Trees.

This appendix contains a description of event tree methodology as used in the study and its role as the principal tool in defining complex accident sequences. It also contains a discussion of the potential accidents explored in the study and presents the event trees used. See Chapter 3.

Appendix II-Fault Trees.

Methodologies used in constructing and quantitatively assessing fault trees are presented along with the results of the quantification of the fault trees used in this study. Individual reports describing the fault tree evaluation of the plant systems analyzed are also presented. See Chapter 4.

Appendix III-Failure Data.

This appendix contains a compendium of data sources and data used in the quantitative evaluation of fault trees and event trees. See Chapter 4.

Appendix IV-Common Mode Failures.

The techniques used in the study to analyze the possible contributions of common mode failures to overall risk assessment are summarized. See Chapter 4.

Appendix V-Quantitative Results of Accident Sequences.

The probabilities of occurrence combined with the radioactive releases for the accidents defined in Appendix I are presented. Also included is the ordering of accident sequences to identify those sequences that are the major contributors to the various sizes of releases. See Chapter 5.

Appendix VI-Calculations of Reactor Accident Consequences.

The model used for predicting the dispersion of radioactivity in the environment is presented, together with the models for predicting the results of this dispersion in terms of fatalities, injuries, long term health effects, and property damage. See Chapter 5.

Appendix VII-Release of Radioactivity in Reactor Accidents.

The factors affecting the magnitude of the release of radioactivity from fuel under various conditions determined by the accident sequences are presented, as are the transport and removal mechanisms that affect the releases of radioactivity from the facility. See Chapter 5.

Appendix VIII-Physical Processes in Reactor Meltdown Accidents.

The various engineered safety feature interactions as defined by the accident sequences are described. Included are predictions of core and containment behavior, along with times of fuel melting, times and modes of containment failure, and the interactions of molten fuel and cladding with water and concrete. See Chapters 3 and 4.

Appendix IX-Safety Design Rationale for Nuclear Power Plants.

A discussion of the safety design rationale currently used for pressurized and boiling water reactors is presented. It includes a discussion of the barriers to the release of radioactivity and their design bases, a discussion of potential accident initiators in nuclear power plants, and the features provided to mitigate the effects of these accident initiators.

Appendix X-Safety Design Adequacy of Nuclear Power Plants.

A study of the extent to which safety design requirements in regard to seismic and accident environments have been fulfilled in the actual engineering design of the plants. See Chapter 5.

Appendix XI-Analysis of Comments on the Draft WASH-1400 Report.

This appendix contains a discussion of the comments received as a result of the draft report.

*This listing is taken from the main report of WASH-1400, and the chapters referred to are in the main report.

to calculate the probability of occurrence of the identified accident sequences. Additional information to be associated with the various accident sequences is the quantity and mode of radioactive release. In WASH-1400, what is referred to as task 1 includes the identification of accident sequences, calculation of their probabilities, and association of an appropriate release mode with each. This task required a preponderance of the study's effort.

The second task was to calculate the consequences of each release mode. The primary elements required in this task were a meteorological model to simulate dispersion, meteorological data to supply the model, population distributions around nuclear power plants, a model for calculating exposures, and one for calculating resulting effects on health and property. The output from this task was a set of consequences to be associated with each release mode.

The final task was to combine the results from the first two tasks, i.e., the probability associated with each accident type and the consequences of each type, to obtain the overall risk from nuclear power plants. The study then went on to compare these risks with other risks to which we are subject.

The manner in which the study performed these tasks is discussed below.

2.1.2.1 Accident Sequence Identification and Calculation of Associated Probabilities and Release Quantities

Identification of possible accident sequences generally proceeded as follows:

1. The location and sizes of all sources of radioactivity in the power plant were determined.

2. The ways in which overheating of the fuel could occur were examined in order to identify events which could initiate sequences leading to such overheating and safety systems which would tend to prevent or limit these sequences.

3. Event trees were constructed to lay out methodically the possible sequences of events; for those sequences which led to releases, the mode of release was identified.

The first step revealed, not surprisingly, that the bulk of the radioactivity in the plant is in the core and the spent fuel storage pool. Typical distributions are shown in Table 2-2. It is interesting to note that the table distinguishes the radioactivity stored in the fuel-cladding gap from that in the fuel itself, a useful distinction since the gap radioactivity is

TABLE 2-2 (Table 3-1 of WASH-1400)

TABLE 3-1 TYPICAL RADIOACTIVITY INVENTORY FOR A 1000 MWe NUCLEAR POWER REACTOR

Location	Total Inventory (Curies)			Fraction of Core Inventory		
	Fuel	Gap	Total	Fuel	Gap	Total
Core[(a)]	8.0×10^9	1.4×10^8	8.1×10^9	9.8×10^{-1}	1.8×10^{-2}	1
Spent Fuel Storage Pool (Max.)[(b)]	1.3×10^9	1.3×10^7	1.3×10^9	1.6×10^{-1}	1.6×10^{-3}	1.6×10^{-1}
Spent Fuel Storage Pool (Avg.)[(c)]	3.6×10^8	3.8×10^6	3.6×10^8	4.5×10^{-2}	4.8×10^{-4}	4.5×10^{-2}
Shipping Cask[(d)]	2.2×10^7	3.1×10^5	2.2×10^7	2.7×10^{-3}	3.8×10^{-5}	2.7×10^{-3}
Refueling[(e)]	2.2×10^7	2×10^5	2.2×10^7	2.7×10^{-3}	2.5×10^{-5}	2.7×10^{-3}
Waste Gas Storage Tank	-	-	9.3×10^4	-	-	1.2×10^{-5}
Liquid Waste Storage Tank	-	-	9.5×10^1	-	-	1.2×10^{-8}

(a) Core inventory based on activity 1/2 hour after shutdown.

(b) Inventory of 2/3 core loading; 1/3 core with three day decay and 1/3 core with 150 day decay.

(c) Inventory of 1/2 core loading; 1/6 core with 150 day decay and 1/3 core with 60 day decay.

(d) Inventory based on 7 PWR or 17 BWR fuel assemblies with 150 day decay.

(e) Inventory for one fuel assembly with three day decay.

more easily released and, in some cases, is the primary release (for example, in cases with no core melting). Based on the observed distribution of radioactivity in the plant, the study concluded that the most serious public consequences would arise from melting of fuel in the core or the storage pool. The study later goes on to show that accidents involving the storage pool do not contribute noticeably to the total risk from the plant. For this reason, although the methodology discussed here is applicable to such accidents, the rest of our discussion will be directed at the reactor system itself.

Under normal operating conditions, a balance is achieved between production of heat in the fuel and removal of heat by the coolant. Fuel overheating therefore occurs if the capacity of the heat removal system decreases below that required by the circumstances or if the heat production rate increases above the normal rate enough to exceed the available heat removal capacity. Although the distinction between these conditions is not unambiguous, it leads to two general categories of conditions which may lead to core melting and/or radioactive release. The first is the <u>loss-of-coolant accident</u> (LOCA), in which - due to a break in the reactor cooling system - the rate of loss of the cooling water is so great that the system inventory cannot be maintained by the makeup system. The second category is that of <u>transients</u>, which decrease the heat removal rate of the cooling system below the heat production rate or which cause the reactor power to increase beyond the heat removal rate. (Strictly speaking, the decreases in removal rate include LOCAs.) Transients and the events which cause LOCA are the events which initiate accident sequences. Identification of these initiating events is an obviously important part of accident analysis.

The specific LOCA initiating events analyzed in the study were:

1. Large pipe breaks (6" to approximately 3 feet equivalent diameter)
2. Small to intermediate pipe breaks (2" to 6" equivalent diameter)
3. Small pipe breaks (1/2" to 2" equivalent diameter)
4. Large disruptive reactor vessel ruptures
5. Gross steam generator ruptures
6. Ruptures between systems that interface with the reactor coolant system

The term reactor "transient" applies to any significant deviation from normal operating values of the important operational parameters of the reactor. Such transients may arise from operator error or equipment failure. Transients which extend parameters beyond the operating range of the reactor control system will cause shutdown of the reactor. From the point of view of safety analysis, the transient itself, or the subsequent shutdown (trip) and decay heat removal systems, are of interest to the extent that they can contribute to increases in core power, decreases in coolant flow, or increases in reactor coolant system pressure. Such changes can lead to damage to the fuel or to the primary coolant system boundary, and the analyses of the study emphasized transients which could lead to such damage.

Potential transients were categorized as either anticipated (likely) or unanticipated (unlikely). Most transients fall into the first category, and power plants experience about 10 such occurrences each year (including some planned shutdowns). The relatively low probability (unanticipated) transients were eliminated from detailed consideration since their contribution to the overall risk was small by comparison with higher probability anticipated transients with similar consequences. In a similar way, the class of anticipated transients was reduced, for assessment purposes, to those involving loss of offsite power and loss of plant heat removal systems.

These transients and the LOCA initiating events listed above served as starting points for <u>event trees</u> used to identify accident sequences. Each branch point in the tree corresponds to an applicable engineered safety feature or function which is assumed either to operate successfully or to fail, thereby allowing only dual branches.* Incorporating into the tree all safety systems which pertain to the initiating event defines a tree which includes all the accident sequences which may follow that event. For each of the tree endpoints corresponding to failure, the failure mode implied by that accident sequence was identified.

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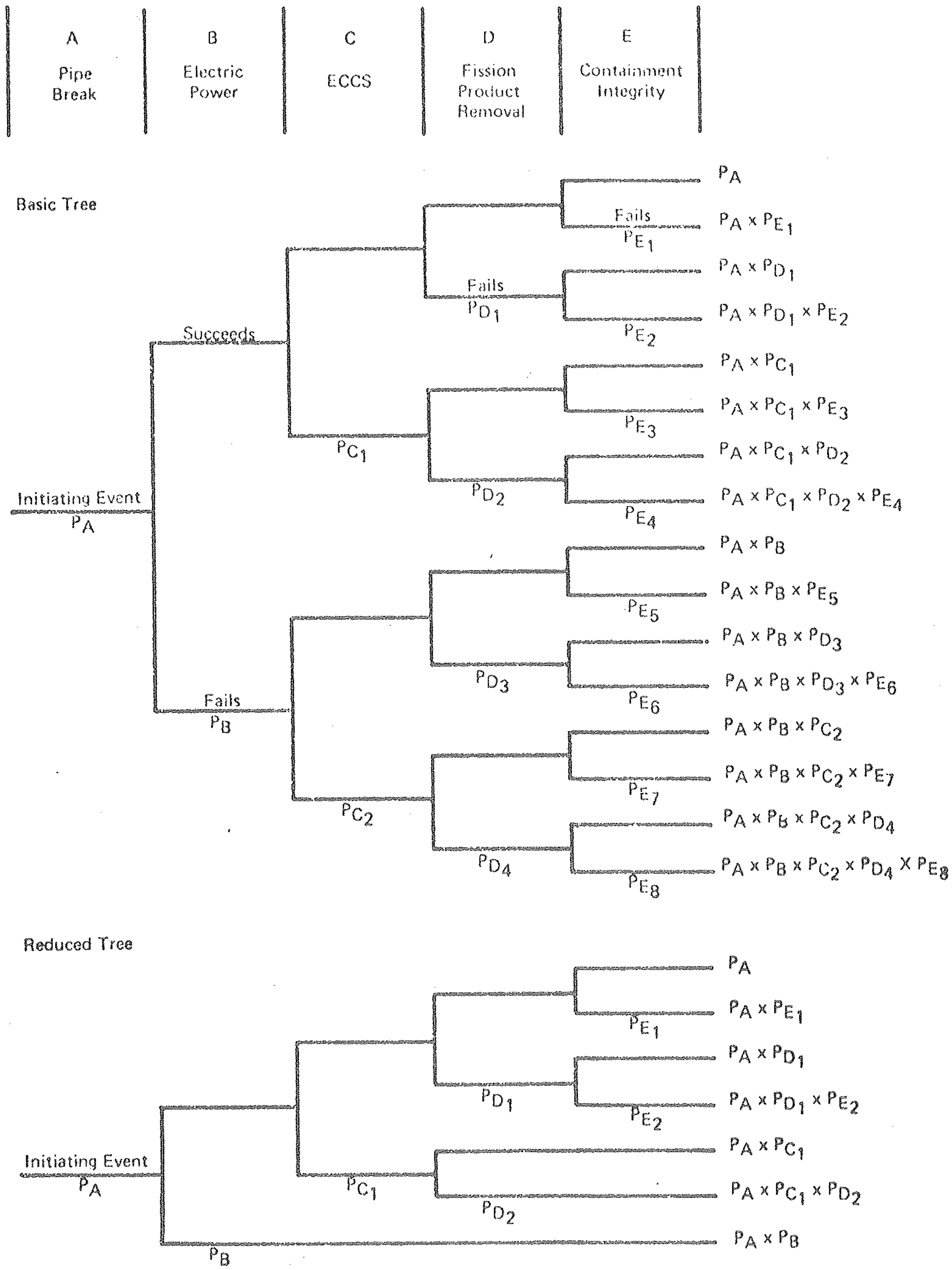
*Evaluation of probabilities on the basis of event trees in which only dual branch points are permitted depends on knowledge of probabilities of system success or system failure. In many systems this requires determination of the fraction of (typically redundant) equipment which must be operable to assure successful function of the system. Where failure of the system contributed in an important way to the risk, further analysis was performed to remove unwarranted conservatism.

These event trees can not, of course, be developed without identifying the safety systems which may be called into action in the course of each accident. For example, during a pipe-break-(PB)-initiated LOCA leading to failure, the engineered safety systems would be called on to perform the following functions:

1. Reactor trip (RT) or shutdown to stop the nuclear chain reaction
2. Emergency core cooling (ECC) to remove decay heat from the core, preventing fuel melting
3. Post accident radioactivity removal (PARR) to remove any released radioactivity from the containment atmosphere
4. Post accident heat removal (PAHR) to remove decay heat from within the containment, preventing containment overpressure
5. Containment integrity (CI) to prevent radioactivity within the containment from escaping into the atmosphere.

In certain cases, branch points in the event tree may be associated, not only with explicit safety systems, but also with other systems, such as the plant electrical systems. However, this does not affect the manner of construction of the trees, an example of which is given in Figure 2-1. In general, if the number of systems which may be called on <u>after</u> the initiating event is N, then the total number of accident sequences which that event may initiate is 2 to the Nth power (2^N), and these will all be visible on the event tree. However, because of interdependence between some of the systems and because of certain simplifying assumptions, the number of branches in the tree may be reduced, as shown in the figure. For example, failure of electrical power implies failure of other systems, thereby requiring that the sequence end in failure. Likewise, failure of the ECC systems leads to core melting. (Core melt is always assumed to lead to failure of the containment. Moreover, another simplification which should be mentioned is that any fuel melting is assumed to imply complete melting of the core.)

Identification of initiating events and of the reactor systems which may be called into action as a result of these events was the second step listed at the beginning of this section. The third step was the construction of the event trees used to define the accident sequences. Completion of the third step required identification of the containment failure modes for these accident sequences leading to failure. Because there were often many factors affecting the failure mode, this identification proceeded by developing contain-



Note - Since the probability of failure, P, is generally less than 0.1, the probability of success (1-P) is always close to 1. Thus, the probability associated with the upper (success) branches in the tree is assumed to be 1.

FIGURE 4-4 Simplified Event Trees for a Large LOCA

FIGURE 2-1 (Figure 4-4 of WASH-1400)

ment event trees, actually extensions of the event trees just described.

Once event trees were developed, quantitative assessment of accident probabilities and release modes could take place. Referring again to Fig. 2-1, associated with each initiating event and each safety system is a probability of occurrence and failure. As indicated in the figure, if these individual probabilities are known, then probabilities for each end point of the tree may be calculated. Further, for a sequence ending in failure, the detailed sequence, including the containment event tree, may be used to determine the magnitude and time sequence of the corresponding radioactive release. Thus it was within the framework of these event trees that quantitative determination of the probability and magnitude of radioactive releases took place.

Calculation of accident probabilities required the probabilities associated with initiating events and safety systems. The first were based on available failure rate data, but most of the system failure rate probabilities were determined with fault tree techniques, since these are suited to the analysis of complex systems. In a sense, a fault tree is constructed in a fashion that is the reverse of the event tree construction process. For a fault tree, one asks how the system of interest can fail when a demand is made that it operate. Based on knowledge of the particular system, the fault tree analyst may identify several possible failures which may lead to a system failure. Depending on whether these failures can independently cause system failure or whether some combination of them is required, they are connected to "system failure" by a logical "OR" or a logical "AND", respectively. Each of these failures, in turn, may arise from a number of causes. This process continues down to the level where failure rate data is available In developing the trees, consideration was given to intrinsic component failures, human factors, and test and maintenance. Once the tree was developed, failure rate data could be assigned and the total system failure probability could be calculated in accordance with the logic of the fault tree. This completed the information required by the event trees for calculation of the accident sequence probabilities.

The accuracy of calculated probabilities depends directly on the accuracy of the basic failure rates. These rates were determined based on the most nearly applicable experience, primarily from similar applications in non-nuclear industry. Failure rates were altered to take account of conditions peculiar to nuclear power plants. Furthermore, a test of the extent to which design specifications were met was made by reviewing a sample of components. In any case, the failures rates input to the probability calculations had large uncertainties, typically factors of 3 to 10. These uncertainties were propagated through the probability calculations using standard statistical methods, based on the "Monte Carlo" technique. It was found that the overall system probabilities, with their associated uncertainties, were precise enough to be useful in risk assessment.

Although we have discussed the construction of event trees and fault trees in rather general terms, it is clear that their utilization in any situation of interest requires analysts with a good understanding of the systems being analyzed. For event trees, the ordering of the relevant systems as they are developed in the tree requires an understanding of how the reactor is designed to operate under accident conditions and how the various systems depend on one another. A similar understanding is required in the development of fault trees, where the problem is made even more difficult by the great complexity of the fault trees. In fact, because there is a good deal of freedom in the manner in which a particular tree is developed and hence in the resulting structure, care must be taken to avoid duplication of the same physical situation.

More importantly, the identification of system interdependencies is made difficult by the complexity of the analysis. To some extent, this difficulty may be overcome by labelling procedures in the development of the trees that are designed for later computer identification of common components and operations. However, human attention must ultimately be relied upon for the basic identification of interdependencies, and their corresponding "common mode" failures. Such failures can significantly increase the overall probability of failure since more than one system can be caused to fail by a single intrinsic failure. The accuracy of any failure analysis may depend critically on the identification of common mode failures, and a substantial portion of the effort of the study was devoted to identification of such failures. Attention was given to such possibilities throughout the various stages of probability modeling and quantification.

It is important to note that not all possible event trees and fault trees were fully developed. In many cases, approximate treatment of a particular sequence indicated that it could not contribute significantly to the overall risk. An iterative process involving successive improvements in the definition of failure probabilities, the incorporation of system interactions, and detail of physical process descriptions was used to identify those sequences having a significant effect in the risk assessment. Only those sequences which could contribute significantly were given detailed treatment. Furthermore, even in these cases, the analyst developing a fault tree must, at some point, truncate his development, excluding those possibilities which do not significantly affect the total failure rate. A check on such truncation was provided in sensitivity studies.

The magnitude of releases to the environment was determined for the accident sequences identified in the event tree analysis discussed above. These were grouped into several "release categories" (9 for the PWR and 5 for the BWR) which were used in the actual risk assessment as discussed below. Each of these categories is associated with a particular type and magnitude of release; included in these characteristics are composition, timing, and release point.

We must comment on modes of release. In all cases of core meltdown, and this includes all the release categories except PWR 8 and 9 and BWR 5, failure of containment is assumed to occur eventually. Failure can occur in two manners: the core can melt through the bottom of the containment before any other breach of containment occurs, or such a breach may occur first. The failure mode significantly affects the point at which radioactivity is released, as well as the timing and magnitude. Melt-through was predicted to occur 1/2 to 1 day after accident initiation, and in such a case - because of the time available for radioactive decay, washout in the containment, and so on, and because of the subsequent filtering action through soil - the total amount of radioactivity effectively released to the environment is relatively small. On the other hand, rupture of the containment, say through overpressure due to carbon dioxide generated from decomposing concrete or through projectiles from a "steam explosion", could permit earlier release and circumvent filtering, thus allowing the possibility of very large releases of radioactivity. Finally, we should note that the study concluded that the categories that did not involve core meltdown did not significantly affect the risk.

The major factors affecting release magnitudes are: the amount and isotopic composition of the radioactivity released from the core, the amount of the radioactivity that is removed within the containment, and the containment failure mode. The time dependence of these factors, and the physical conditions associated with them, were determined within the framework of the event trees discussed above. For the differing sequences, this information, and the relevant experimental data on movement of radioactivity out of the core and within the containment, were used by a computer code which calculated the quantity of each of 54 biologically significant radioisotopes released to the environment. These yields were calculated for approximately 60 "key" accident sequences, selected from approximately 1000 identified sequences, but whose timing and physical processes during the accident were characteristic of the large majority of identified sequences. Many of the accident sequences involve similarity in core melting, radioactivity removal processes, and containment failure modes. On the basis of results from the code, it was then possible to group all identified sequences into the several release categories mentioned above, which are characterized by composition, timing, and release point. For each category, the sum of the calculated probabilities (discussed above) of the associated accident sequences yields a probability for that release category. This is the final information sought in this first step of the risk assessment, i.e., for a series of releases categories, values for the probability and magnitude of releases.

Drawing on figures from the main report, we indicate the basic results from this step in the risk assessment. The dominant (i.e., high probability) accident sequences are grouped by release category in tables 2-3 (PWR) and 2-4 (BWR). Using the associated keys, the codes for the indicated sequences maybe deciphered. As a convenience, a brief description of the major accident sequences contributing to each release category is given in table 2-5. In tables 2-3 and 2-4, it is important to note that, in order to allow for the fact that release quantities (and therefore categories) may be estimated incorrectly for any given accident sequence, each sequence was assumed to have a 10% probability of occurring in the adjacent release categories. (The release categories are ordered roughly by the size of the release.) This is the major reason that the total probability for any release category differs from the sum of the probabilities of the accident sequences assigned to that release category.

Table 2-6 summarizes information on the release categories, including total probabilities, timing, and release magnitudes. (The equivalent information for accidents not involving the core is also given.) From the point of view of risk, the categories of most importance are those, roughly, whose product of total probability and release magnitude (particularly for iodine and

TABLE 2-3 (Table 5-2 of WASH-1400)

TABLE 5-2 PWR DOMINANT ACCIDENT SEQUENCES vs. RELEASE CATEGORIES

RELEASE CATEGORIES	Core Melt							No Core Melt	
	1	2	3	4	5	6	7	8	9
LARGE LOCA A	AB-α 1×10^{-11} AF-α 1×10^{-10} ACD-α 5×10^{-11} AG-α 9×10^{-11}	AB-γ 1×10^{-10} AB-δ 4×10^{-11} AHF-γ 2×10^{-11}	AD-α 2×10^{-8} AH-α 1×10^{-8} AF-δ 1×10^{-8} AG-δ 9×10^{-9}	ACD-β 1×10^{-11}	AD-β 4×10^{-9} AH-β 3×10^{-9}	AB-ε 1×10^{-9} AHF-ε 1×10^{-10} ADF-ε 2×10^{-10}	AD-ε 2×10^{-6} AH-ε 1×10^{-6}	A-β 2×10^{-7}	A 1×10^{-4}
A Probabilities	2×10^{-9}	1×10^{-8}	1×10^{-7}	1×10^{-8}	4×10^{-8}	3×10^{-7}	3×10^{-6}	1×10^{-5}	1×10^{-4}
SMALL LOCA S_1	S_1B-α 3×10^{-11} S_1CD-α 7×10^{-11} S_1F-α 3×10^{-10} S_1G-α 3×10^{-10}	S_1B-γ 4×10^{-10} S_1B-δ 1×10^{-10} S_1HF-γ 6×10^{-11}	S_1D-α 3×10^{-8} S_1H-α 3×10^{-8} S_1F-δ 3×10^{-8} S_1G-δ 3×10^{-8}	S_1CD-β 1×10^{-11}	S_1H-β 5×10^{-9} S_1D-β 6×10^{-9}	S_1DF-ε 3×10^{-10} S_1B-ε 2×10^{-9} S_1HF-ε 4×10^{-10}	S_1D-ε 3×10^{-6} S_1H-ε 3×10^{-6}	S_1-β 6×10^{-7}	S_1 3×10^{-4}
S_1 Probabilities	3×10^{-9}	2×10^{-8}	2×10^{-7}	3×10^{-8}	8×10^{-8}	6×10^{-7}	6×10^{-6}	3×10^{-5}	3×10^{-4}
SMALL LOCA S_2	S_2B-α 1×10^{-10} S_2F-α 1×10^{-9} S_2CD-α 2×10^{-10} S_2G-α 9×10^{-10} S_2C-α 2×10^{-8}	S_2B-γ 1×10^{-9} S_2HF-γ 2×10^{-10} S_2B-δ 4×10^{-10}	S_2D-α 9×10^{-8} S_2H-α 6×10^{-8} S_2F-δ 1×10^{-7} S_2C-δ 2×10^{-6} S_2G-δ 9×10^{-8}	S_2DG-β 1×10^{-12}	S_2D-β 2×10^{-8} S_2H-β 1×10^{-8}	S_2B-ε 8×10^{-9} S_2CD-ε 2×10^{-8} S_2HF-ε 1×10^{-9}	S_2D-ε 9×10^{-6} S_2H-ε 6×10^{-6}		
S_2 Probabilities	1×10^{-7}	3×10^{-7}	3×10^{-6}	3×10^{-7}	3×10^{-7}	2×10^{-6}	2×10^{-5}		
REACTOR VESSEL RUPTURE - R	RC-α 2×10^{-12}	RC-γ 3×10^{-11} RF-δ 1×10^{-11} RC-δ 1×10^{-12}	R-α 1×10^{-9}				R-ε 1×10^{-7}		
R Probabilities	2×10^{-11}	1×10^{-10}	1×10^{-9}	2×10^{-10}	1×10^{-9}	1×10^{-8}	1×10^{-7}		
INTERFACING SYSTEMS LOCA (CHECK VALVE) - V		V 4×10^{-6}							
V Probabilities	4×10^{-7}	4×10^{-6}	4×10^{-7}	4×10^{-8}					
TRANSIENT EVENT - T	TMLB'-α 3×10^{-8}	TMLB'-γ 7×10^{-7} TMLB'-δ 2×10^{-6}	TML-α 6×10^{-8} TKQ-α 3×10^{-8} TKMQ-α 1×10^{-8}		TML-β 3×10^{-10} TKQ-β 3×10^{-10}	TMLB'-ε 6×10^{-7}	TML-ε 6×10^{-6} TKQ-ε 3×10^{-6} TKMQ-ε 1×10^{-6}		
T Probabilities	3×10^{-7}	3×10^{-6}	4×10^{-7}	7×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}		
(Σ) SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORY									
MEDIAN (50% VALUE)	9×10^{-7}	8×10^{-6}	4×10^{-6}	5×10^{-7}	7×10^{-7}	6×10^{-6}	4×10^{-5}	4×10^{-5}	4×10^{-4}
LOWER BOUND (5% VALUE)	9×10^{-8}	8×10^{-7}	6×10^{-7}	9×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}	4×10^{-6}	4×10^{-5}
UPPER BOUND (95% VALUE)	9×10^{-6}	8×10^{-5}	4×10^{-5}	5×10^{-6}	4×10^{-6}	2×10^{-5}	2×10^{-4}	4×10^{-4}	4×10^{-3}

Note: The probabilities for each release category for each event tree and the Σ for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacent release category probability.

KEY TO TABLE 5-2 ON FOLLOWING PAGE

KEY TO PWR ACCIDENT SEQUENCE SYMBOLS

A - Intermediate to large LOCA.

B - Failure of electric power to ESFs.

B' - Failure to recover either onsite or offsite electric power within about 1 to 3 hours following an initiating transient which is a loss of offsite AC power.

C - Failure of the containment spray injection system.

D - Failure of the emergency core cooling injection system.

F - Failure of the containment spray recirculation system.

G - Failure of the containment heat removal system.

H - Failure of the emergency core cooling recirculation system.

K - Failure of the reactor protection system.

L - Failure of the secondary system steam relief valves and the auxiliary feedwater system.

M - Failure of the secondary system steam relief valves and the power conversion system.

Q - Failure of the primary system safety relief valves to reclose after opening.

R - Massive rupture of the reactor vessel.

S_1 - A small LOCA with an equivalent diameter of about 2 to 6 inches.

S_2 - A small LOCA with an equivalent diameter of about 1/2 to 2 inches.

T - Transient event.

V - LPIS check valve failure.

α - Containment rupture due to a reactor vessel steam explosion.

β - Containment failure resulting from inadequate isolation of containment openings and penetrations.

γ - Containment failure due to hydrogen burning.

δ - Containment failure due to overpressure.

ϵ - Containment vessel melt-through.

KEY TO TABLE 5-2

TABLE 2-4 (Table 5-3 of WASH-1400)

TABLE 5-3 BWR DOMINANT ACCIDENT SEQUENCES OF EACH EVENT TREE vs. RELEASE CATEGORY

	Core Melt				No Core Melt
	RELEASE CATEGORIES				
	1	2	3	4	5
LARGE LOCA DOMINANT ACCIDENT SEQUENCES (A)	AE-α 2×10^{-9} AJ-α 1×10^{-10} AHI-α 1×10^{-10} AI-α 1×10^{-10}	AE-γ′ 3×10^{-8} AE-β 1×10^{-8} AJ-γ′ 2×10^{-9} AI-γ′ 2×10^{-9} AHI-γ′ 2×10^{-9}	AE-γ 1×10^{-7} AJ-γ 1×10^{-8} AI-γ 1×10^{-8} AHI-γ 1×10^{-8}	AGJ-δ 6×10^{-11} AEG-δ 7×10^{-10} AGHI-δ 6×10^{-11}	A 1×10^{-4}
A Probabilities	8×10^{-9}	6×10^{-8}	2×10^{-7}	2×10^{-8}	1×10^{-4}
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S_1)	S_1E-α 2×10^{-9} S_1J-α 3×10^{-10} S_1I-α 4×10^{-10} S_1HI-α 4×10^{-10}	S_1E-γ′ 4×10^{-8} S_1E-β 1×10^{-8} S_1J-γ′ 7×10^{-9} S_1I-γ′ 7×10^{-9} S_1HI-γ′ 6×10^{-9}	SE-γ 1×10^{-7} S_1J-γ 3×10^{-8} S_1I-γ 4×10^{-8} S_1HI-γ 2×10^{-8} S_1C-γ 3×10^{-9}	S_1GJ-δ 2×10^{-10} S_1GE-δ 2×10^{-10} S_1EI-ε 1×10^{-10} S_1GHI-δ 2×10^{-10}	
S_1 Probabilities	1×10^{-8}	9×10^{-8}	2×10^{-7}	2×10^{-8}	
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S_2)	S_2J-α 1×10^{-9} S_2I-α 1×10^{-9} S_2HI-α 1×10^{-9} S_2E-α 5×10^{-10}	S_2E-γ′ 1×10^{-8} S_2E-β 4×10^{-9} S_2J-γ′ 2×10^{-8} S_2I-γ′ 2×10^{-8} S_2HI-γ′ 2×10^{-8}	S_2E-γ 4×10^{-8} S_2J-γ 8×10^{-8} S_2I-γ 9×10^{-8} S_2HI-γ 9×10^{-8} S_2C-γ 8×10^{-9}	S_2CG-δ 6×10^{-11} S_2GHI-δ 6×10^{-10} S_2EG-δ 3×10^{-10} S_2GJ-δ 6×10^{-10} S_2GI-δ 2×10^{-10}	
S_2 Probabilities	2×10^{-8}	1×10^{-7}	4×10^{-7}	4×10^{-8}	
TRANSIENT DOMINANT ACCIDENT SEQUENCES (T)	TW-α 2×10^{-7} TC-α 1×10^{-7} TQUV-α 5×10^{-9}	TW-γ′ 3×10^{-6} TQUV-γ′ 8×10^{-8}	TW-γ 1×10^{-5} TC-γ 1×10^{-5} TQUV-γ 4×10^{-7}		
T Probabilities	1×10^{-6}	6×10^{-6}	2×10^{-5}	2×10^{-6}	
PRESSURE VESSEL RUPTURE ACCIDENTS (R)		P.V. RUPT. 1×10^{-8} Oxidizing Atmosphere	P.V. RUPT. 1×10^{-7} Non-oxidizing Atmosphere		
P Probabilities	2×10^{-9}	2×10^{-8}	1×10^{-7}	1×10^{-8}	
SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORIES					
MEDIAN (50% VALUE)	1×10^{-6}	6×10^{-6}	2×10^{-5}	2×10^{-6}	1×10^{-4}
LOWER BOUND (5% VALUE)	1×10^{-7}	1×10^{-6}	5×10^{-6}	5×10^{-7}	1×10^{-5}
UPPER BOUND (95% VALUE)	8×10^{-6}	3×10^{-5}	8×10^{-5}	1×10^{-5}	1×10^{-3}

NOTE: The probabilities for each release category for each event tree and the Σ for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacent release category probability.

KEY TO TABLE 5-3 ON FOLLOWING PAGE

KEY TO BWR ACCIDENT SEQUENCE SYMBOLS

A - Rupture of reactor coolant boundary with an equivalent diameter of greater than six inches.

B - Failure of electric power to ESFs.

C - Failure of the reactor protection system.

D - Failure of vapor suppression.

E - Failure of emergency core cooling injection.

F - Failure of emergency core cooling functionability.

G - Failure of containment isolation to limit leakage to less than 100 volume per cent per day.

H - Failure of core spray recirculation system.

I - Failure of low pressure recirculation system.

J - Failure of high pressure service water system.

M - Failure of safety/relief valves to open.

P - Failure of safety/relief valves to reclose after opening.

Q - Failure of normal feedwater system to provide core make-up water.

S_1 - Small pipe break with an equivalent diameter of about 2"-6".

S_2 - Small pipe break with an equivalent diameter of about 1/2"-2".

T - Transient event.

U - Failure of HPCI or RCIC to provide core make-up water.

V - Failure of low pressure ECCS to provide core make-up water.

W - Failure to remove residual core heat.

α - Containment failure due to steam explosion in vessel.

β - Containment failure due to steam explosion in containment.

γ - Containment failure due to overpressure - release through reactor building.

γ' - Containment failure due to overpressure - release direct to atmosphere.

δ - Containment isolation failure in drywell.

ϵ - Containment isolation failure in wetwell.

ζ - Containment leakage greater than 2400 volume per cent per day.

η - Reactor building isolation failure.

θ - Standby gas treatment system failure.

KEY TO TABLE 5-3

TABLE 2-5 - APPROXIMATE DESCRIPTION OF RELEASE CATEGORIES

PWR Release Categories

1 - steam explosion, failed containment spray and heat removal systems (possibly causing overpressure in containment); explosion ruptures reactor vessel and containment

2 - failure of core cooling systems, followed by failure of containment spray and heat removal systems and failure of the containment through overpressure

3 - overpressure failure of containment due to failure of containment heat removal systems; containment failure is followed by core melting, venting through containment

4 - failure of core-cooling and containment spray after LOCA, and failure of containment isolation

5 - failure of core-cooling after LOCA (but containment spray works), and failure of containment isolation

6 - failure of core cooling and containment spray, but integral containment until melt-through

7 - failure of core-cooling, but containment spray works and containment is integral until melt-through

8 - large pipe break LOCA with failure of containment isolation, no core melt

9 - large pipe break LOCA, no melt

BWR Release Categories

1 - steam explosion

2 - transient event with subsequent failure of heat removal and failure of containment due to overpressure - leading to core melt

3 - transient event with failure of scram or heat removal, failure of containment before or after core melt

4 - core melt, containment leakage (not failure)

5 - large pipe break LOCA, no melt

Table 2-6 (Tables V 2-1 and V 2-2 of WASH-1400)

TABLE V 2-1 SUMMARY OF ACCIDENTS INVOLVING CORE

RELEASE CATEGORY	PROBABILITY per Reactor-Yr	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	WARNING TIME FOR EVACUATION (Hr)	ELEVATION OF RELEASE (Meters)	CONTAINMENT ENERGY RELEASE (10^6 Btu/Hr)	FRACTION OF CORE INVENTORY RELEASED[a] Xe-Kr	Org. I	I	Cs-Rb	Te-Sb	Ba-Sr	Ru[b]	La[c]
PWR 1	$9x10^{-7}$	2.5	0.5	1.0	25	520[d]	0.9	$6x10^{-3}$	0.7	0.4	0.4	0.05	0.4	$3x10^{-3}$
PWR 2	$8x10^{-6}$	2.5	0.5	1.0	0	170	0.9	$7x10^{-3}$	0.7	0.5	0.3	0.06	0.02	$4x10^{-3}$
PWR 3	$4x10^{-6}$	5.0	1.5	2.0	0	6	0.8	$6x10^{-3}$	0.2	0.2	0.3	0.02	0.03	$3x10^{-3}$
PWR 4	$5x10^{-7}$	2.0	3.0	2.0	0	1	0.6	$2x10^{-3}$	0.09	0.04	0.03	$5x10^{-3}$	$3x10^{-3}$	$4x10^{-4}$
PWR 5	$7x10^{-7}$	2.0	4.0	1.0	0	0.3	0.3	$2x10^{-3}$	0.03	$9x10^{-3}$	$5x10^{-3}$	$1x10^{-3}$	$6x10^{-4}$	$7x10^{-5}$
PWR 6	$6x10^{-6}$	12.0	10.0	1.0	0	N/A	0.3	$2x10^{-3}$	$8x10^{-4}$	$8x10^{-4}$	$1x10^{-3}$	$9x10^{-5}$	$7x10^{-5}$	$1x10^{-5}$
PWR 7	$4x10^{-5}$	10.0	10.0	1.0	0	N/A	$6x10^{-3}$	$2x10^{-5}$	$2x10^{-5}$	$1x10^{-5}$	$2x10^{-5}$	$1x10^{-6}$	$1x10^{-6}$	$2x10^{-7}$
PWR 8	$4x10^{-5}$	0.5	0.5	N/A	0	N/A	$2x10^{-3}$	$5x10^{-6}$	$1x10^{-4}$	$5x10^{-4}$	$1x10^{-6}$	$1x10^{-8}$	0	0
PWR 9	$4x10^{-4}$	0.5	0.5	N/A	0	N/A	$3x10^{-6}$	$7x10^{-9}$	$1x10^{-7}$	$6x10^{-7}$	$1x10^{-9}$	$1x10^{-11}$	0	0
BWR 1	$1x10^{-6}$	2.0	2.0	1.5	25	130	1.0	$7x10^{-3}$	0.40	0.40	0.70	0.05	0.5	$5x10^{-3}$
BWR 2	$6x10^{-6}$	30.0	3.0	2.0	0	30	1.0	$7x10^{-3}$	0.90	0.50	0.30	0.10	0.03	$4x10^{-3}$
BWR 3	$2x10^{-5}$	30.0	3.0	2.0	25	20	1.0	$7x10^{-3}$	0.10	0.10	0.30	0.01	0.02	$3x10^{-3}$
BWR 4	$2x10^{-6}$	5.0	2.0	2.0	25	N/A	0.6	$7x10^{-4}$	$8x10^{-4}$	$5x10^{-3}$	$4x10^{-3}$	$6x10^{-4}$	$6x10^{-4}$	$1x10^{-4}$
BWR 5	$1x10^{-4}$	3.5	5.0	N/A	150	N/A	$5x10^{-4}$	$2x10^{-9}$	$6x10^{-11}$	$4x10^{-9}$	$8x10^{-12}$	$8x10^{-14}$	0	0

(a) A discussion of the isotopes used in the study is found in Appendix VI. Background on the isotope groups and release mechanisms is found in Appendix VII.

(b) Includes Mo, Rh, Tc, Co.

(c) Includes Nd, Y, Ce, Pr, La, Nb, Am, Cm, Pu, Np, Zr.

(d) A lower energy release rate than this value applies to part of the period over which the radioactivity is being released. The effect of lower energy release rates on consequences is found in Appendix VI.

TABLE V 2-2 SUMMARY OF ACCIDENTS NOT INVOLVING CORE[a]

ACCIDENT	PROBABILITY OF OCCURRENCE per Reactor-Yr	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	ELEVATION OF RELEASE (M)	EQUIVALENT FRACTION OF CORE INVENTORY RELEASED[c] Xe-Kr	Org. I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La
LOSS OF COOLING IN SFSP	$<10^{-6}$ [b]	5	5	40	10^{-1}	$7x10^{-4}$	$7x10^{-4}$	10^{-3}	$6x10^{-4}$	10^{-4}	$2x10^{-4}$	$2x10^{-5}$
DROPPED SHIPPING CASK	$6x10^{-7}$	10	10	0	$3x10^{-7}$	10^{-11}	10^{-11}	$1x10^{-7}$	$3x10^{-10}$	$5x10^{-11}$	∼0	∼0
REFUELING ACCIDENT	10^{-3}	<1	<1	40	$9x10^{-5}$	$4x10^{-7}$	$4x10^{-7}$	$6x10^{-9}$	$2x10^{-11}$	$3x10^{-11}$	∼0	∼0
WASTE GAS STORAGE TANK RUPTURE	10^{-2}	<1	<1	0	$2x10^{-4}$	10^{-9}	10^{-9}	∼0	∼0	∼0	∼0	∼0
LIQUID WASTE STORAGE TANK RUPTURE	10^{-2}	<1	<1	0	∼0	$8x10^{-8}$	$8x10^{-8}$	$6x10^{-8}$	$4x10^{-9}$	$5x10^{-11}$	$2x10^{-8}$	10^{-11}
EARTHQUAKE-INDUCED LOSS OF COOLING IN SFSP WITH LOSS OF AIR COOLING SYSTEM	$3x10^{-8}$	5	5	0	10^{-1}	$7x10^{-2}$	$7x10^{-2}$	10^{-1}	$6x10^{-2}$	10^{-2}	10^{-2}	10^{-3}

(a) PWR and BWR designs were examined, and the more severe accidents were selected as representative bounds for both.
(b) Estimated probability includes consideration of turbine- and tornado-generated missiles.
(c) Fractions of total core inventory 30 minutes after shutdown.

the alkalis) is largest.* As can be seen from tables 2-3, 2-4 (and 2-5), the release categories were usually dominated by probabilities contributed by one or few accident sequences of each category. As a further note, single system failures were found to dominate the accident sequences which determined the release category probabilities, and single component failures, in turn, dominated the single system failure probability. Thus, common mode failures between components had little impact. On the other hand, in certain systems, common modes - usually arising from the same human source - were significant. Moreover, common mode considerations were important in defining accident sequences in the event tree analysis.

The probability information on release categories is displayed in Figs. 2-2 (PWR) and 2-3 (BWR). The total probabilities for core melt-down may be obtained by summing the probabilities of the corresponding release categories. The results are 6×10^{-5} per reactor-year for PWRs and 3×10^{-5} for BWRs, yielding an average for the first 100 reactors of about 5×10^{-5} per reactor-year.

In addition to internal causes, the study considered the importance of releases induced by external causes. Earthquakes, tornadoes, floods, aircraft impacts, and missiles from the steam turbine were treated on an approximately quantitative basis; they were not treated in great detail because it was concluded that they did not contribute risks comparable to internal causes. Sabotage was not considered quantitatively; the study does, however, conclude that the probability of successful sabotage is "low".

Based on its systematic approach to accident identification and quantification, as well as sensitivity studies and other checks, the study expressed a high degree of confidence in its probability and release magnitude results and in the effectiveness of its effort to include all accident sequences with important contributions to risk. See section 3 for comments of others.

2.1.2.2 Calculation of Consequences Corresponding to Identified Release Modes

The next step in the risk assessment of WASH-1400 was to use the information developed on characteristics of each of the release categories to calculate the probable consequences of each category. In addition to these release characteristics, the consequences depend on how the radioactivity is

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*See section 4.3.

0 0 0 0 4 5 0 8 4 7 5

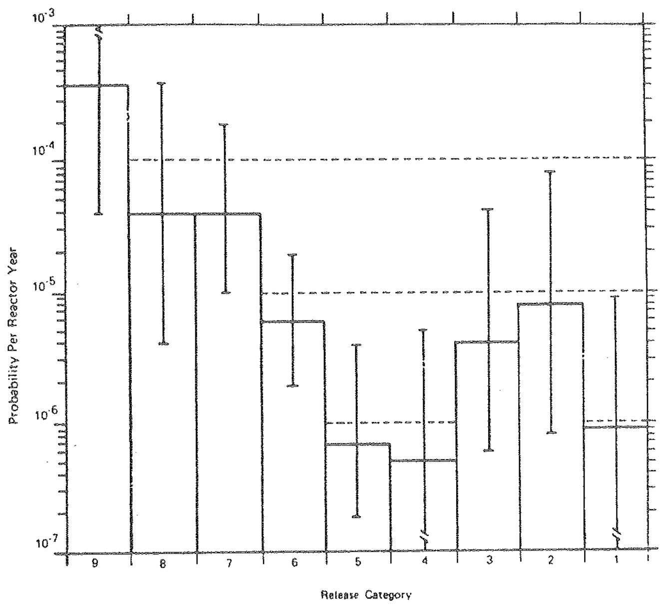

FIGURE 5-1 Histogram of PWR Radioactive Release Probabilities

FIGURE 2-2 (Figure 5-1 of WASH-1400)

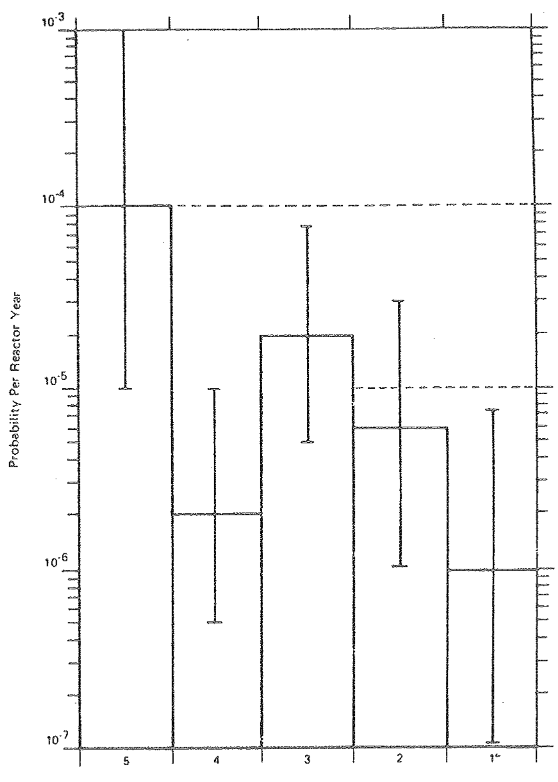

FIGURE 5-2 Histogram of BWR Radioactive Release Probabilities

FIGURE 2-3 (Figure 5-2 of WASH-1400)

dispersed in the environment, on the exposure of populations and property, and on the effects of this exposure. A calculational model was developed to evaluate these various dependencies and yield a set of consequences as a function of probability. Calculation of the consequences involves an atmospheric dispersion model, a population model, and a health effects and property damage model. Determination of the corresponding probabilities requires combination of the release probabilities, the probabilities for weather type, and the probabilities implicit in various population distributions.

Calculation of atmospheric dispersion was based on a Gaussian plume model, with modifications. The required inputs to this model were the mode of release (including information on point and time of release and on the amount of radioactivity and heat released) and the weather conditions during the release. The output of the model was the radioactivity concentrations (by isotope) in the air and on the ground as a function of both time and distance from the power plant.

The release mode data were developed as discussed in the previous section. The model characterized weather as one of six stability classes. The weather data used were hourly meteorological records, covering a period of a year, from six sites which typify the locations of the first 100 nuclear power plants. For each site, 90 samples of data were taken as input sets, including stability category, wind velocity, and rainfall. The weather samples were chosen equally from each season and from night and day. The dispersion model used the release mode data and weather data, together with parameters needed for calculating decay and deposition of radioactivity and for making certain corrections, to yield the required radioactivity concentrations.

The population model used census data reduced to population density versus distance from the reactor for each of 16 equal sectors of 22.5°. Such data from the various reactors assigned to any one of the six site types (previously characterized by meteorological data) were assembled to form a composite population distribution corresponding to that type. For major releases, it was assumed that all people within 5 miles of the plant, and those within 25 miles downwind, would be evacuated according to a simple model, in order to reduce early exposure to individuals. Population exposures were calculated considering external dose from the passing cloud, internal dose from inhalation-deposition of the passing cloud, external dose from ground-deposited material, and internal dose from inhaled resuspended material and from subsequently ingested ground contamination.

Health effects and property damage may then be calculated using results from the dosimetric model in conjunction with some dose-response relationship. (Property damage would involve such a relationship, because judgments on evacuation and cleanup or denial of property depend on the effects of radiation exposure.) The effects calculated were early fatalities (occurring within one year), early illnesses (requiring medical treatment), and cancer deaths, thyroid illness, and genetic effects arising long after the accident (due to the overall population exposure). In addition, property damage due to radioactive contamination was calculated, including population relocation costs, as well as the cost of decontaminating or abandoning property.

Briefly, the calculated early fatalities were dominated by cancers resulting from exposures of bone marrow; for this exposure mode, 510 rads* was chosen as the 50% fatality dose. The largest portion of early illnesses occurred due to several thousand rads dose to the lung. Long-term deaths, illness, and genetic effects were calculated based on recommendations of the BEIR report,[3] but with significant reduction of latent cancer deaths on the basis of dose rate and magnitude dependencies.

The health effects and property damage calculational model did not yield a distinct spectrum of consequences for each release category. Rather, it completed the convolution, to obtain an overall relationship between the magnitude of consequences versus the probability of that magnitude. These results are discussed in the next section.

Before proceeding, it is worth noting that the model calculated consequences using a large number of distinct combinations of release mode, weather data, and population distribution. There were a total of 14 release categories and 6 typical sites, each with 90 meteorological samples and 16 composite population sectors. This yields over 100,000 possible combinations. It is understandable that separate consequences were not presented for each combination. However, as noted in sections 3 and 4, it would have been interesting to see the consequences broken down by release category.

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*The WASH-1400 main report comments that 510 rads was chosen as the 50% fatality dose for whole-body (rather than bone marrow) exposures, but this comment is imprecise.

2.1.2.3 Risk from Nuclear Power Plant Accidents

The final step in risk determination, the convolution of the probabilities of various accidents with their consequences, was actually performed by the consequence calculational code discussed in the previous section. In each case, the probability was a product of the release category probability, the weather probability, and the population distribution probability. (This assumes these individual probabilities are independent.) For each type of consequence (early fatalities, early illness, and so on), the results were displayed graphically as the total probability for consequences of a minimum magnitude versus that magnitude. In addition, consequences were stated in tabular form as minimum consequences of accidents of specified probability. Finally, the sum of magnitudes times probabilities was taken to yield the overall risk to society. All these consequences were stated, not only for a single reactor, but for a national system of 100 reactors. (In the latter case, it was assumed that values intermediate between the PWR and BWR results could be extrapolated to the 100 reactors.) It is the study's 100-reactor results that we now state.

Figures 2-4, 2-5, 2-6, 2-7, 2-8, 2-9, and 2-10 show, respectively, early fatalities, early illness, latent cancer fatalities, thyroid nodules, genetic effects, property damage, and relocation and decontamination areas. Similar results are shown in tabular form in tables 2-7 and 2-8, and average overall risk probabilities are given in table 2-9.

For the delayed effects (latent cancers, thyroid, nodules, and genetic effects), note that the results stated are effects per year after a given accident, and not the summed effect of the accident. For each accident considered, the early effects have their effect soon after the accident and their total number is stated unambiguously. However, the delayed effects occur over a long period, roughly 30 years, and - for a given accident - one may state either the total number of effects over that period or the rate at which they occur. The final WASH-1400 report chose the latter form, thereby reducing the size of the numbers stated by a factor of 30. This manner of presentation is consistent with the form used in studies of the relationship between radiation exposures and delayed effects, but may obscure the fact that the integrated number of latent effects often substantially exceeds the number of early effects. (See section 4.3.)

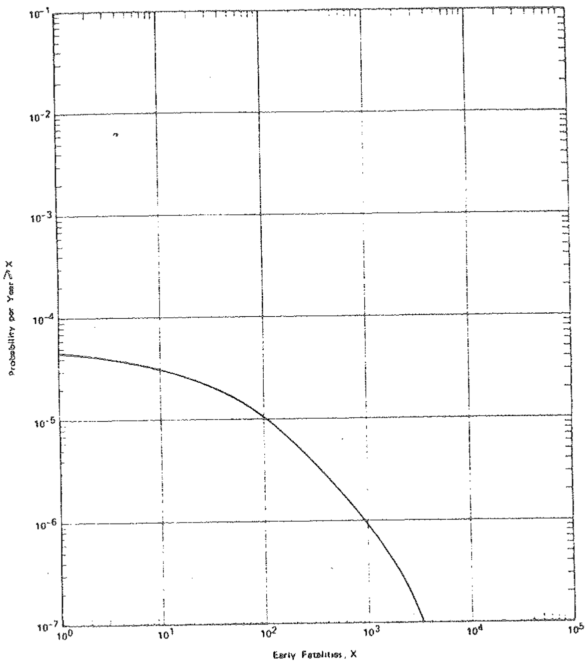

FIGURE 5-10 Probability Distribution for Early Fatalities per Year for 100 Reactors

Note: Approximate uncertainties are estimated to be represented by factors of 1/4 and 4 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

FIGURE 2-4 (Figure 5-10 of WASH-1400)

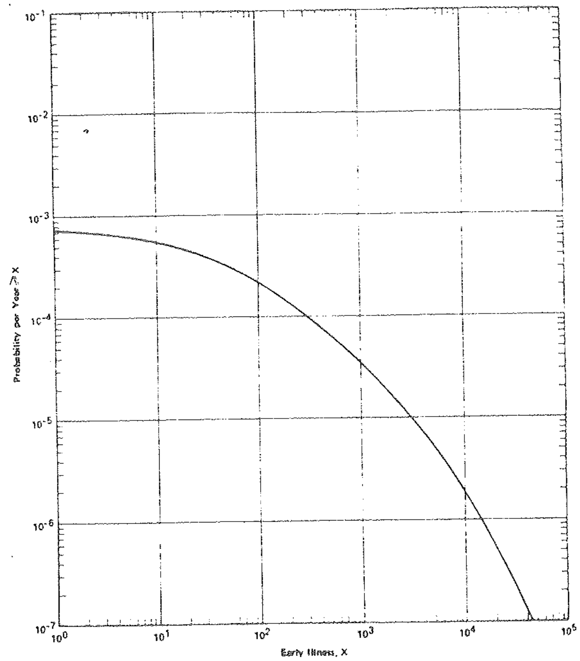

FIGURE 5-11 Probability Distribution for Early Illness per Year for 100 Reactors

Note: Approximate uncertainties are estimated to be represented by factors of 1/4 and 4 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

FIGURE 2-5 (Figure 5-11 of WASH-1400)

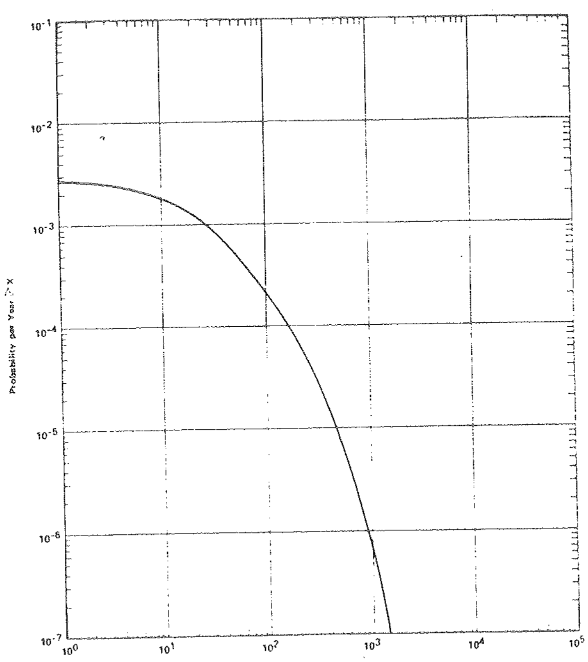

FIGURE 5-12 Probability Distribution for Latent Cancer Fatality Incidence per Year for 100 Reactors

Note: Approximate uncertainties are estimated to be represented by factors of 1/6 and 3 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

FIGURE 2-6 (Figure 5-12 of WASH-1400)

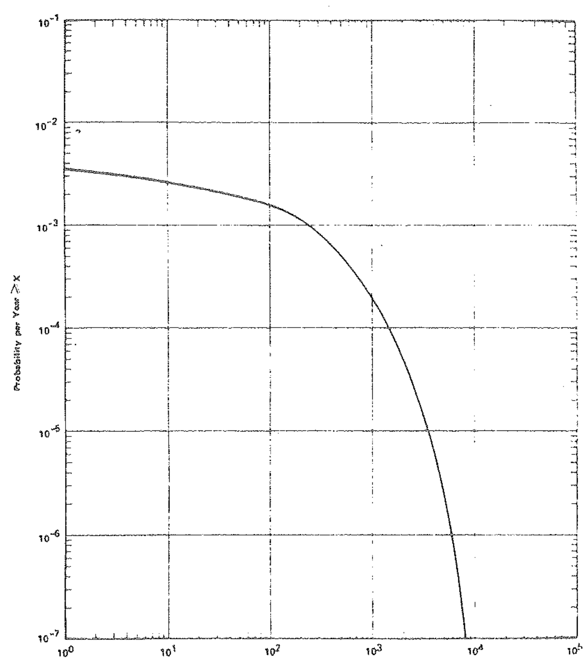

FIGURE 5-14 Probability Distribution for Thyroid Nodule Incidence per Year for 100 Reactors

Note: Approximate uncertainties are estimated to be represented by factors of 1/3 and 3 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

FIGURE 2-7 (Figure 5-14 of WASH-1400)

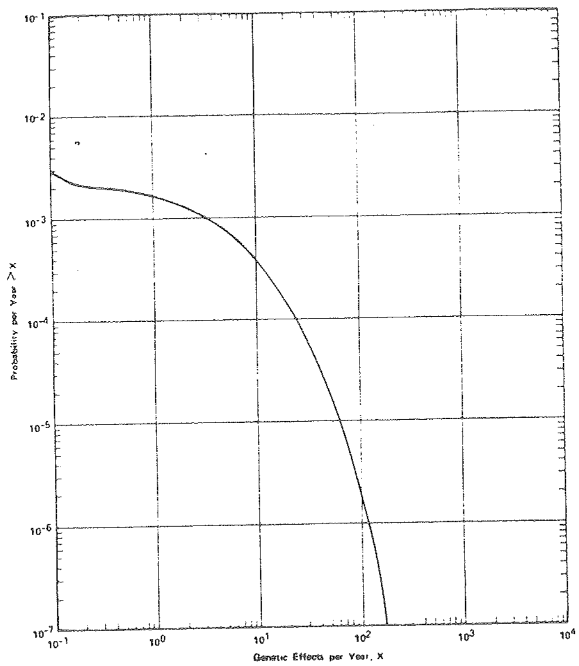

FIGURE 5-13 Probability Distribution for Incidence of Genetic Effects per Year for 100 Reactors

Note: Approximate uncertainties are estimated to be represented by factors of 1/3 and 6 on consequence magnitudes and by factors of 1/5 and 5 on probabilities.

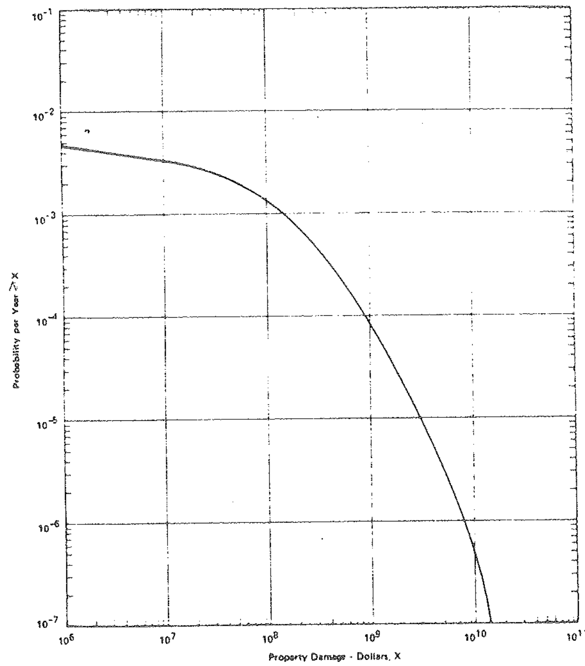

FIGURE 5-15 Probability Distribution for Property Damage per Year for 100 Reactors

Note: Approximate uncertainties are estimated to be represented by factors of 1/5 and 2 on consequence magnitudes and by factors of 1.5 and 5 on probabilities.

FIGURE 2-8 (Figure 5-13 of WASH-1400) FIGURE 2-9 (Figure 5-15 of WASH-1400)

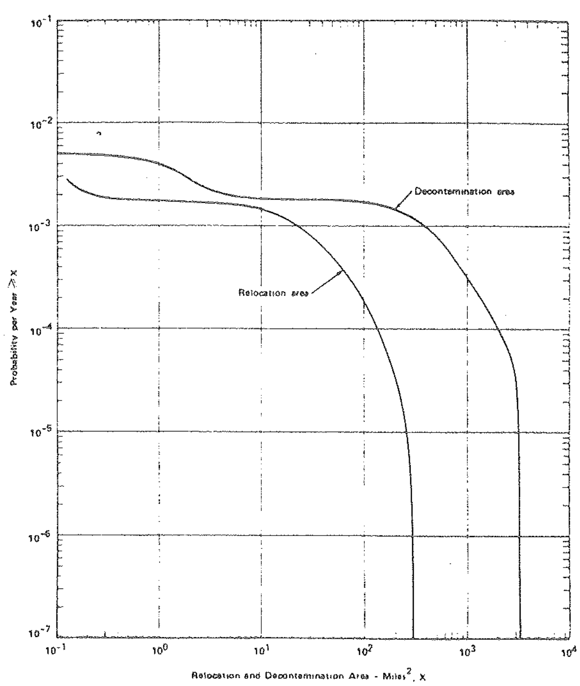

FIGURE 5-16 Probability Distribution for Relocation and Decontamination Area per Year for 100 Reactors

Note: Approximate uncertainties are estimated to be represented by factors of 1/5 and 2 on consequence magnitudes and by factors of 1.5 and 5 on probabilities.

FIGURE 2-10 (Figure 5-16 of WASH-1400)

TABLE 2-7 (Table 5-7 of WASH-1400)

TABLE 5-7 CONSEQUENCES OF REACTOR ACCIDENTS FOR VARIOUS PROBABILITIES FOR 100 REACTORS

	Consequences				
Chance Per Year	Early Fatalities	Early Illness	Total Property Damage $\$10^9$	Decontamination Area Square Miles	Relocation Area Square Miles
One in 200[(a)]	<1.0	<1.0	<0.1	<0.1	<0.1
One in 10,000	<1.0	300	0.9	2000	130
One in 100,000	110	300	3	3200	250
One in 1,000,000	900	14000	8	(b)	290
One in 10,000,000	3300	45000	14	(b)	(b)

(a) This is the predicted chance per year of core melt considering 100 reactors.
(b) No change from previously listed values.

TABLE 2-8 (Table 5-8 of WASH-1400)

TABLE 5-8 CONSEQUENCES OF REACTOR ACCIDENTS FOR VARIOUS PROBABILITIES FOR 100 REACTORS

	Consequences *		
Chance Per Year	Latent Cancer Fatalities[(b)] (per year)	Thyroid Nodules[(b)] (per year)	Genetic Effects[(c)] (per year)
One in 200[(a)]	<1.0	<1.0	<1.0
One in 10,000	170	1400	25
One in 100,000	460	3500	60
One in 1,000,000	860	6000	110
One in 10,000,000	1500	8000	170
Normal Incidence	17,000	8000	8000

(a) This is the predicted chance per year of core melt for 100 reactors.

(b) This rate would occur approximately in the 10 to 40 year period after a potential accident.

(c) This rate would apply to the first generation born after the accident. Subsequent generations would experience effects at decreasing rates.

*See text.

TABLE 2-9 (Table 5-6 of WASH-1400)

TABLE 5-6 APPROXIMATE AVERAGE SOCIETAL AND INDIVIDUAL RISK PROBABILITIES PER YEAR FROM POTENTIAL NUCLEAR PLANT ACCIDENTS[(a)]

Consequence	Societal	Individual
Early Fatalities[(b)]	3×10^{-3}	2×10^{-10}
Early Illness[(b)]	2×10^{-1}	1×10^{-8}
Latent Cancer Fatalities[(c)]	7×10^{-2}/yr*	3×10^{-10}/yr*
Thyroid Nodules[(c)]	7×10^{-1}/yr*	3×10^{-9}/yr*
Genetic Effects[(d)]	1×10^{-2}/yr*	7×10^{-11}/yr*
Property Damage ($)	2×10^{6}	——

(a) Based on 100 reactors at 68 current sites.

(b) The individual risk value is based on the 15 million people living in the general vicinity of the first 100 nuclear power plants.

(c) This value is the rate of occurrence per year for about a 30-year period following a potential accident. The individual rate is based on the total U.S. population.

(d) This value is the rate of occurrence per year for the first generation born after a potential accident; subsequent generations would experience effects at a lower rate. The individual rate is based on the total U.S. population.

*See text.

It is important to note that the uncertainties in the various results are large (factors of 3 or more), even though they are not stated in the tabular results or displayed on the graphical results. They are stated below the figures.

Some care should be taken in reading consequence magnitudes from these graphs and tables. In each case, the stated magnitude is the minimum effect of accidents included in that probability category. For example, from figure 2-4 (and table 2-7), the probability of having an accident with at least 110 fatalities is 10^{-5} per year (or once in 100,000 years). Table 2-9, which states overall risks, has summed the probabilities times the consequences (see also section 4).

2.1.2.4 Comparison with Other Risks

Having estimated the risks to society (and individuals) from nuclear power, the study went on to assemble historical data on risks from other sources, including both man-caused and natural events. Risks from early fatalities and property damage from these sources are displayed, along with those from 100 reactors, in figures 2-11, 2-12, and 2-13. Tables 2-10 and 2-11 show average incidence of fatalities, injuries, and economic loss for nuclear and non-nuclear accidents in the United States. In all these comparisons, the nuclear-contributed effects are found to be very small relative to other accidental risks. (However, note that figures 2-11 and 2-12 only compare "early" effects, which are much smaller than latent effects.)

2.1.3 Conclusions of the Study

The basic conclusion of the study leads directly from the comparison described in the previous section. The study concludes that the possible consequences of reactor accidents would be no larger and, in many cases, would be much smaller than those of non-nuclear accidents; and the likelihood of reactor accidents is much smaller than that of many non-nuclear accidents having similar consequences. As such, the risk to the public from nuclear power plant accidents is found to be comparatively small.

These conclusions of the study are coupled with a confidence that the study succeeded in identifying the significant accidents in nuclear power plants. This confidence is based on the systematic approach of the study in

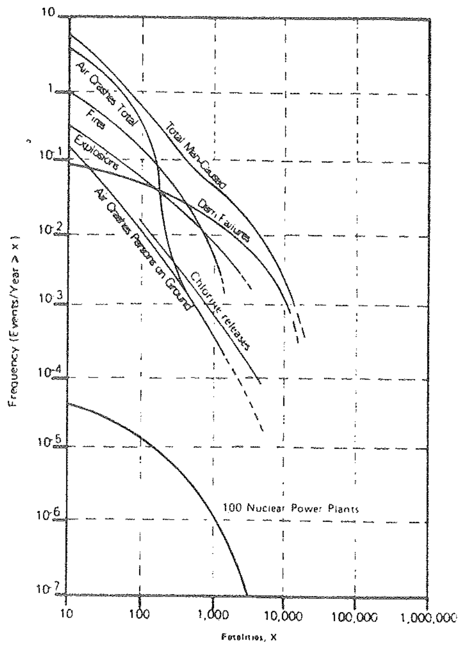

FIGURE 6-1 Frequency of Man-Caused Events Involving early Fatalities.

Notes 1 Fatalities due to auto accidents are not shown because data are not available for large consequence accidents Auto accidents cause about 50 000 fatalities per year

2. See section 6 4 for a discussion of confidence bounds applicable to the non nuclear curve See section 5.5 for the confidence bounds on the nuclear curve

FIGURE 2-11 (Figure 6-1 of WASH-1400)

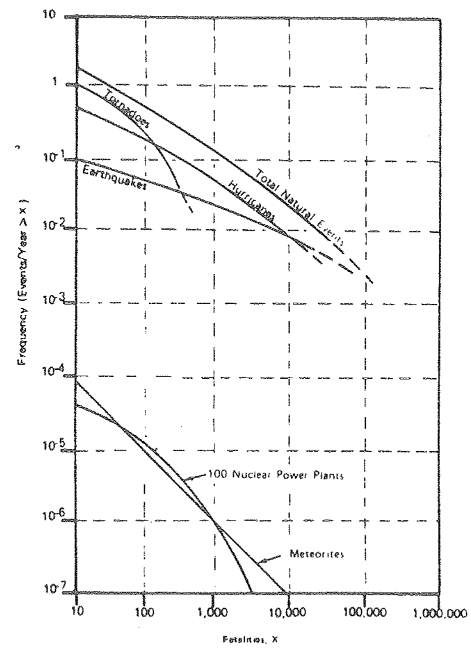

FIGURE 6-2 Frequency of Natural Events Involving early Fatalities.

Note See section 6.4 for a discussion of confidence bounds applicable to the non nuclear curve See section 5.5 for the confidence bounds on the nuclear curve.

Figure 2-12 (Figure 6-2 of WASH-1400)

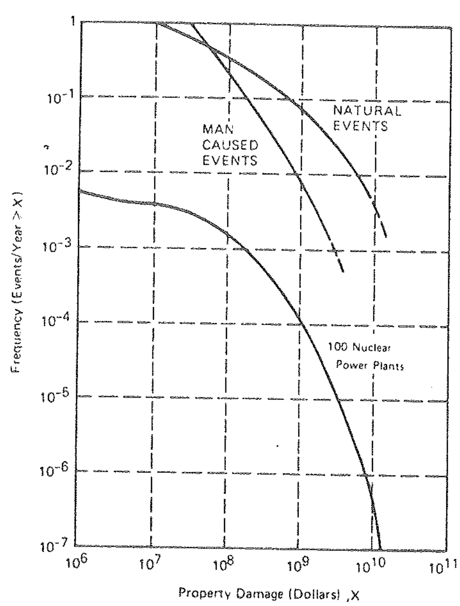

FIGURE 6-3 Frequency of Accidents Involving Property Damage

Notes 1. Property damage due to auto accidents is not included because data are not available for low probability events. Auto accidents cause about $15 billion damage each year.

2. See section 6.4 for a discussion of confidence bounds applicable to the non nuclear curve. See section 5.5 for the confidence bounds on the nuclear curve.

FIGURE 2-13 (Figure 6-3 of WASH-1400)

TABLE 2-10 (Table 6-6 of WASH-1400)

TABLE 6-6 ANNUAL ACCIDENT FATALITIES AND INJURIES IN THE U.S.

	Total United States		People Within 25 Miles of Nuclear Sites	
Accident Type	Fatalities	Injuries	Fatalities	Injuries
Automobile	55,000	5×10^6	4200	375,000
Falls	20,000	1×10^6	1500	75,000
Fire	7,500	0.3×10^6	560	22,000
Other	33,000	1.6×10^6	2500	120,000
TOTAL	115,000	7.9×10^6	8760	592,000
Reactor Accidents (for 100 plants from Table 5-6, Chapter 5)	7×10^{-2}*	1*	3×10^{-3}*	2×10^{-1}*

*These numbers appear to be in error. Total U.S. annual fatalities and injuries are, respectively, 2 and 20; these are obtained by multiplying the latent effects predictions (given as number per year per year in table 2-9) by an incidence period of roughly 30 years. Effects within 25 miles cannot be determined on the basis of the information stated in WASH-1400.

TABLE 2-11 (Table 6-7 of WASH-1400)

TABLE 6-7 U.S. ECONOMIC LOSSES FROM VARIOUS CAUSES

Source	Estimated Annual Losses (Millions of $)
Automobile Accidents (1970)	5,000
Fires (Property - 1970)	2,200
Hurricanes (1952-72 average)	500
Fires (Forest - 1970)	70
Tornadoes (1970)	50
Reactor Accidents from 100 plants (See Table 5-6, Chapter 5)	2

identifying sites of radioactivity, in utilizing a comprehensive understanding of the conditions under which radioactivity could be liberated from the fuel, and in applying techniques which - when used in a careful manner, with appropriate sensitivity tests - can yield quantitative results for the probability and magnitude of releases.

The detailed probability results are of some interest. For example, it was found that the large LOCA accident sequences contributed about a factor of 10 less to the probability of core melt than did the dominant contributors to core meltdown. In the case of the PWR, the major contributor to core melt probability was the small (pipe-break) LOCA, rather than LOCAs induced by large or intermediate breaks (see table 2-3). The major meltdown probabilities in the BWR were contributed by transient-initiated sequences in which the reactor shutdown system failed or the decay heat removal system failed. (See table 2-4.) (Although the study does not make a point of it, it is worth noting that these initiators do not necessarily contribute the bulk of the risk to the public. This is because the high-probability type meltdown is also the low release meltdown. For example, a scan of table 2-3 shows that the check valve failure sequence dominates the probability for causing a large release, and hence could be the major contributor to risk.*)

As noted before, the calculated average probability of meltdown is 5×10^{-5} per reactor-year. The study notes that, in approximately 2000 reactor-years of experience with commercial and military power reactors, no nuclear accident has affected the public, and goes on to suggest that this implies that the likelihood of accidents is less than 10^{-3} per reactor year. In addition, the fact that reactors of the type studied have not experienced small accidents, or elevated fuel temperatures is said to suggest meltdown probabilities much less than 10^{-3}, particularly in view of the general experience that large accidents are much less probable than small ones. These hints are consistent with the calculated value of 5×10^{-5} per reactor-year. (However, the suitability of this comparison has been questioned.*)

The study makes some closing remarks on its methodology. The purpose of the study was to achieve a realistic estimate of risks from reactor accidents, and participants in the study were confident that the basic risk assessment methodology, in particular the use of event and fault trees to find accident probabilities and to define containment failure modes, led to realistic results.

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*See discussion of section 4.3.

However, for the sake of simplicity, a conservative approach was taken in certain other areas, for example in calculation of the actual quantities of radioactivity released, in the treatment of plume rise and rain deposition, and in the allowance of steam explosions. Furthermore, the extrapolation of the design features of the two reactors chosen for detailed information was conservative in that newer reactors should be less likely to have accidents because of better design and greater experience.

The study is particularly careful to warn against immediate attempts to apply the risk assessment methodology to change reactor design in order to decrease accident probabilities. This warning is made in view of the fact that the methodological developments of the study were aimed toward overall risk assessment and may therefore not be useful or accurate for design analysis; in any case, the study concludes that more developmental effort is needed.

Lastly, the study indicates that future experimental and theoretical work on radioactive releases from molten fuel, on steam explosions, on plume behavior, and in methodological development would be useful in determining the degree of conservatism present in various assumptions made in the analysis. (It must be emphasized that this discussion is the study commenting on itself.)

2.1.4 Responses to Comments

A draft of the report WASH-1400 was released for public comment in August, 1974. Between that time and October 1975, the time of release of the final report, the Atomic Energy Commission and its successor in carrying out the study, the Nuclear Regulatory Commission, received comments from a variety of organizations and individuals. These comments ranged in generality from assessments of the overall study methodology to criticisms of very specific aspects of the study. To indicate explicitly its response to these comments, the final WASH-1400 report contains an eleventh appendix in which the major comments are mentioned and an indication is given both of the study's extent of agreement with the comments and of the manner in which such agreement resulted in changes in the results or in the report.

A large portion of the comments focussed on the two areas which - in general terms - constitute the major ingredients in WASH-1400's risk assessment: the probabilistic methodology and the calculation of consequences. In the first area, reviewers expressed doubts that all important accident sequences could be identified and, in particular, that those involving common mode failures could be found; furthermore, many comments indicated skepticism of the data base for quantification of the probabilities and, closely related, of the dependability of the fault tree methodology in yielding absolute numbers for the probability of failure, particularly where the probabilities are small. As indicated in the discussion of the previous sections, the authors of WASH-1400 express confidence that their systematic application of the methodology and their various checks provide assurance that the study results are accurate within the quoted uncertainties. They also point out that the probabilities of failure typically calculated from the fault trees are not as small as those resulting from previous applications of the fault tree methodology.

In the second area mentioned, that of consequences, each step of the consequence calculation has been criticized. These steps include the meteorological model, the population exposure model (including evacuation), and the health effects and propety damage model. The authors of WASH-1400 respond that the final results are based on a much improved consequences model (described in appendix VI). The greatest alteration in the consequences arises in the stated results for the latent health effects.

(This alteration is to some extent obscured by a change in the manner in which the results are presented. The draft report stated the total number of cancers, thyroid illness, and genetic effects induced by accidents of the stated probability, whereas the final report states each of these per year after the accident. The latter mode of reporting reduced the numerical value by a factor of 30.)

Additional comments were received in the more specific areas: probability of accident sequences, radioactive releases from accident sequences, emergency cooling functionability, reactor vessel rupture, large nuclear excursions, behavior of radionuclides in soil and water, core melt analysis, steam explosions, hydrogen combustion, data base, external forces, sabotage, scope, and design adequacy.

The study's assessment of the effect of changes to the draft report is worth quoting:

> "In general, the potential consequences predicted in the final report have increased over those predicted in the draft report. All predicted consequences, except one, were within the factors of 1/3 and 3 error bands of the values predicted in the draft report. The predicted average value of latent cancers increased by a factor of about 7,due principally to the error made in the weathering half life that was assigned for cesium decay in the draft report. This effect also increased the land area needing decontamination by 5 and that in which relocation is required by 10. Early illnesses were calculated on an organ by organ basis which increased the magnitude by a factor of 6. The rest of the changes were within the confidence bounds of the predictions of the draft report. The study believes that its current consequences model is conservative and that the potential consequences in the final report represent near upper bound limits for those consequences such as early effects, property damage and contaminated land areas... The above noted changes do not change the basic conclusion of the draft report that reactor risks are relatively small compared to other societal risks". (our emphasis added)

Evidence that these responses have not been entirely satisfactory to some reviewers of WASH-1400 is given by the appearance of subsequent criticism of the final report. These criticisms again center on the two general areas mentioned above, as well as on the presentation of the report and the public use which has been made of it. Some weaknesses of WASH-1400 were recognized by its authors or are apparent on independent reading. In subsequent sections of this report, a number of possible weaknesses will be mentioned, most often as part of the discussion of other studies, especially those which actually commented on WASH-1400. Particular attention will be given to how the WASH-1400 methodology might be improved or its results presented in a more useful manner.

We note finally that the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research has established a Probabilistic Analysis Branch, which is continuing the work begun during the Reactor Safety Study. Current directions of this work are indicated in section 3.2.

2.2 Studies of the Electric Power Research Institute

2.2.1 Introduction

The Electric Power Research Institute (EPRI), as the research organ of the electric utility industry, funds a major part of the research devoted to light-water reactor safety in this country. The bulk of this funding is devoted to specific aspects of reactor safety, but a portion of it is directed at more general evaluations, either to identify those areas where more research and development are needed or to assess the overall reliability of and risk from nuclear power.

Of particular interest from the point of view of this report is the risk-reliability assessment work carried out through contracts with Science Applications, Inc. (SAI). Investigators at SAI were involved in the probabilistic work performed for WASH-1400, so that it is not surprising that their work for EPRI should emphasize this aspect of nuclear reactor risk assessment. To date, the work funded by EPRI had led to the publication of six EPRI reports in a series, EPRI-217-2-

- -1 Summary of the AEC Reactor Safety Study (WASH-1400), April 1975
- -2 Generalized Fault Tree Analysis for Reactor Safety, June 1975
- -3 Critique of the AEC Reactor Safety Study (WASH-1400), June 1975
- -4 Probabilistic Safety Analysis, July 1975
- -5 User's Guide for the Wam-Bam Computer Code, January 1976
- -6 Sensitivity Assessment in Reactor Safety Analysis, February 1976

We discuss certain aspects of this work here, rather than under reviews of WASH-1400 (section 3.2), because it represents a significant extension of the methodology employed in WASH-1400, rather than a mere comment on that report. In particular, a major emphasis of the EPRI-SAI work has been the development techniques for reliability assessment (i.e., probabilistic methodologies), directed explicitly toward an understanding of improvements which may be made in the design or operation of nuclear power plants.

It is, however, true that two EPRI reports deal explicitly with the WASH-1400 draft. The main points of their critique of that report are of interest, both because these criticisms reflect on the utility of WASH-1400 and because they give a strong indication of the intended direction of the work at SAI, an indication that is borne out by the work they have performed.

As a general comment, SAI points out that WASH-1400 developed little information on the conservatism of their approach in instances where completely realistic calculations were not possible. In particular, little sensitivity work was done to show how variation of significant parameters affects the results. Moreover, the presentation in WASH-1400 did not break down contributors to risk in any detail, so that it is difficult to identify where improvements could be made to reduce risk.

More specific comments were also made on WASH-1400: in the important area of common mode failures, no systematic procedure for their identification was developed; the assumption that all core melting leads to complete core melting and to eventual breach of containment may not be correct; the use in draft WASH-1400 of large-LOCA sequences as the basis for release categorization, even for small breaks, was not justified in detail; with respect to consequences, use of site-averaged meteorological conditions and populations distributions may affect conservatism; moreover, the study did not consider the special aspects of multi-unit sites or the effects of growth of population around power plant sites.

These are by no means all of the comments in SAI's review, but they do reflect the major directions for the work for EPRI: the development of more general probabilistic analysis techniques, which would systematically include, for example, common mode failures and which would use sensitivity techniques to identify the major contributors risk. Note, however, that SAI's comments on WASH-1400 refer specifically to the 1974 draft version.

2.2.2 Methodological Development

2.2.2.1 Generalized Fault Tree Analysis

The fault trees developed in the Reactor Safety Study depended, basically, on two types of "gates" or logical connections, AND and OR. Failure at some level could be described as depending either on some combination of more basic failures, each of which must be present to cause the dependent failure (in which case the AND gate would be used), or on the failure of one of a selection of failures (in which case the OR gate would be employed). Such fault trees can only represent independent sequences, although to some extent the additional use in WASH-1400 of an INHIBIT gate to switch on specific logic when a conditional input is satisfied does extend the modeling capability. Even so, it is not fully capable of treating arbitrary dependencies or common modes.

SAI has used a more general set of logical gates, based on addition of a NOT function to the basic set. Combination of NOT with the AND and OR yields a total of 16 logical operations on two inputs. These combinations make it possible to study diagramatically and analytically a broader range of interactions, including dependencies, common modes, and mutually exclusive events.

The details of the resulting analytical capability are not of concern here. The fault trees which can be developed for reactor systems can be quite complex, and SAI has developed a computer code BAM (for Boolean Arithmetic Model) for reducing and evaluating such trees. This code accepts trees with the full set of 16 operations on two logical variables and uses a truth table technique (i.e., describes in tabular form the output of a particular gate, yes or no, as a function of the inputs to the gate) for tree reduction and evaluation of failure values. (To simplify use of this code, a preprocessor, WAM, was developed to accept a logical tree with input components and gates and to generate a numeric input for BAM. Hence, Wam-Bam.) The result is the ability to calculate point* unavailability of a complex system, including a variety of interdependencies presumably not available in the WASH-1400 approach except as rather arbitrary appendages to WASH-1400 fault trees.

2.2.2.2 Sensitivity Analysis

A substantial lack in the WASH-1400 report was that no sufficient indication was given of the major contributors to risks. There is great value in identifying how risk depends on component failure rate, external events, human interactions, or test and maintenance. SAI has developed a systematic approach to this question based, to some extent, on perturbation analysis. This approach defines "sensitivity indicators", which indicate the dependence of one probability (such as system unavailability) on changes in another probability (such as component failure probability). Such indicators may be developed at any level in risk analysis. They may be used to relate probabilities for overall conse-

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*"Point" unavailability is the failure probability, for a specific demand, as calculated presuming specific failure probabilities for the components making up the system. Point unavailability therefore does not give any indication of how uncertainties in the basic failure rates induce uncertainties in the calculated system unavailability.

quences, for release categories, or for system failures, to probabilities for more basic events, down to component failure, etc.

It is clear that such information is of considerable value, and we give some examples, not because the details of the methodology are of great interest here, but because the examples serve to elucidate certain points which were obscure in WASH-1400. In considering this information, it is important to realize that the SAI work described in the above reports was based on the Draft WASH-1400, rather than than on the final report.

SAI bases its release indicators nominally on certain classes of phenomena, labelled conventionally as:

C_i - the severity of consequence of type i, such as fatalities or acute illness,

R_j - the radioactivity release category j, for one of the six BWR release categories or nine PWR release categories,*

W_j - an accident consequence option j, such as whether or not evacuation is allowed,

F_k - a plant function k, made up of one or more systems, which is designed to mitigate particular accident results; for example, Emergency Core Coolant Injection,

S_k - a system k which performs all or part of a plant function, for example, Auxiliary Feedwater or High Pressure Coolant Injection,

E_m - an event m, such as a pump or valve failure.

In general, one may regard one of these types of phenomenon as depending on another and define an indicator I. Those specified by SAI are:

consequence indicators:

$I_{C_iR_j}$ - the indicator for the change in probability per year for a given severity of consequence C_i with a change in the probability for radioactivity release R_j.

$I_{C_iW_j}$ - the indicator for the change in probability per year for a given severity of consequence C_i with a change in the accident consequence option W_j.

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*Draft WASH-1400 release categories BWR 3 and 4 were combined to form BWR 3 in the final report. BWR 5 and 6 became respectively, BWR 4 and 5.

release indicators

$I_{R_jF_k}$ - the indicator for the change in the release category probability per year for category R_j with a change in the probability of failure for plant function F_k,

$I_{R_jE_m}$ - The indicator for the change in the release category probability per year for category R_j with a change in the probability of failure for event E_m,

system indicators

$I_{S_kE_m}$ - the indicator for the change in the probability per year for failure of system S_k with a change in the probability of failure for event E_m.

For those instances where well-defined probabilities exist, for both the dependent and the independent phenomena, the indicator may be defined in a rather general manner. SAI chooses to define I in such cases as the ratio of the change in the dependent probability to the change in the independent probability divided by the ratio of the original dependent and independent probabilities*.

The consequence indicators cannot be so defined, at least as presented in WASH-1400, because the results are displayed as a cumulative probability versus a continuous spectrum of consequences (see, for example, figures 2-4 to 2-10). As a result, SAI's consequence versus release indicators ($I_{C_iR_j}$) give directly the percentage that a given release category contributes to the risk, rather than giving the ratio of changes. Since this breakup of the overall consequences into the consequences from different release categories is of particular interest, we give some of SAI's results, emphasizing that they are based on the information in draft WASH-1400. To extract this information, SAI

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*For example, a system indicator for event failures may be expressed as:

$$I_{S_kE_m} = \frac{P^*(S_k) - P(S_k)}{P^*(E_m) - P(E_m)} \Bigg/ \frac{P(S_k)}{P(E_m)} = \left(1 - \frac{P^*(S_k)}{P(S_k)}\right) \Bigg/ \left(1 - \frac{P^*(E_m)}{P(E_m)}\right),$$

where the asterisk indicates the altered probabilities.

had to recalculate the consequences for the individual release categories. A selection of those results is given in figures 2-14 to 2-17, where early and long-term fatalities due to whole body doses are given for a PWR and a BWR as a function of release category. This information may then be used to obtain the consequence indicators just described. These are given in tables 2-12 and 2-13. Also given there are the summed consequences (T^*) for each type of consequence. (Note that these types differ from those presented in the final WASH-1400, particularly since both doses and their effects are given.) These indicators are the percentage that each release category contributes to each total consequence. This information was not available in the WASH-1400 reports, but is necessary to identify which release categories contribute the major risk. Note that category 2 dominates the PWR consequences, as does category 4 the BWR. (These refer to draft WASH-1400 release categories.) This is of some interest, particularly since PWR 2 is a low probability, but large release, category (see table 2-6). BWR 4, on the other hand, is the high probability category. (These consequences, for either type of reactor, cannot be taken as definitive, particularly since the calculations on which they are based presumably contain many of the same errors for which the draft WASH-1400 was critized.)

We also quote results for the less general indicators, which do correspond to a ratio of changes in probabilities as indicated above. In each case, the larger the magnitude of the indicator, the more sensitive is the dependent probability to changes in the more basic probability. For the PWR, we give the release indicators for event failures in table 2-14, where the indicator is given, in each case, for a range of alterations in the event failure probability. Note, for example, that release category 2 (the most consequential for the PWR) is the most sensitive to failures due to test and maintenance. On the other hand, category 7, which is the core meltdown category that occurs most often, is most sensitive to human error. We give similar information for the BWR in table 2-15 where the dependence of release category on selected plant function failures (QUV = water inventory make-up, W = decay heat removal, C = reactor shutdown system) is also shown. Category 4, the predominant risk contributor, is the most dependent on hardware failures or, alternatively, on failure of decay heat removal. Indicators for the specific plant functions just mentioned are given in table 2-16. Note that the hardware failures listed in the table do not exhaust the list of such failures. This accounts for the

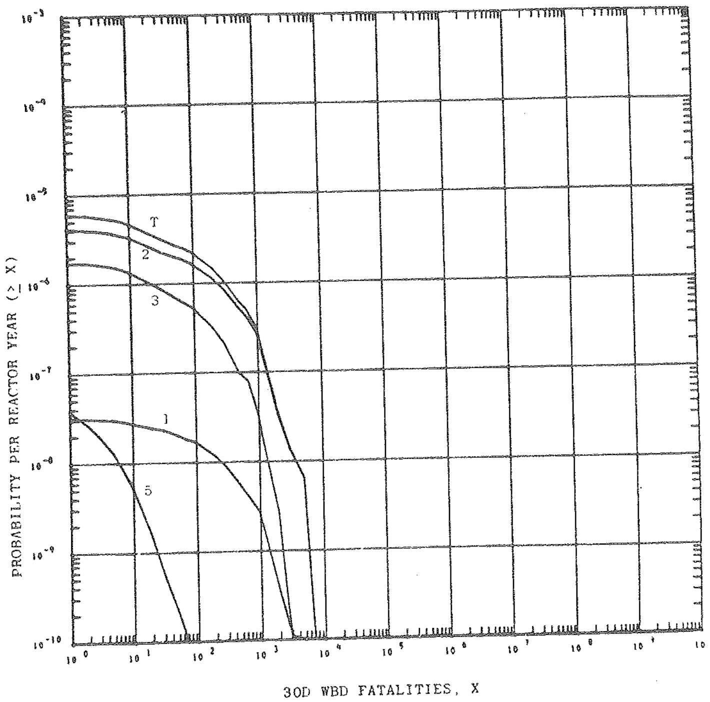

Figure 18F. Probability Distribution of Fatalities from 30 Day Whole Body Dose from PWR Accidents.

FIGURE 2-14 (Figure 18F of EPRI 217-2-6)

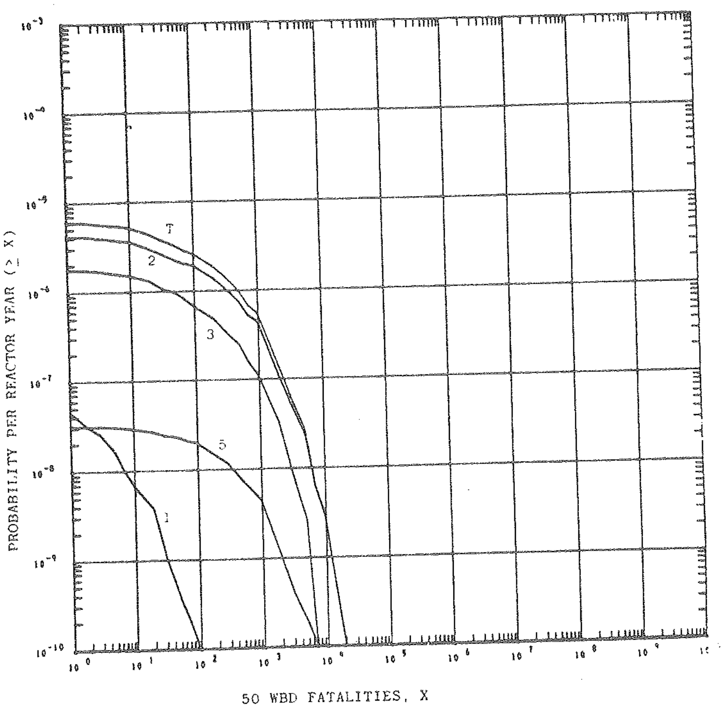

Figure 18D. Probability Distribution of Fatalities from 50 Year Whole Body Dose from PWR Accidents.

FIGURE 2-15 (Figure 18D of EPRI 217-2-6)

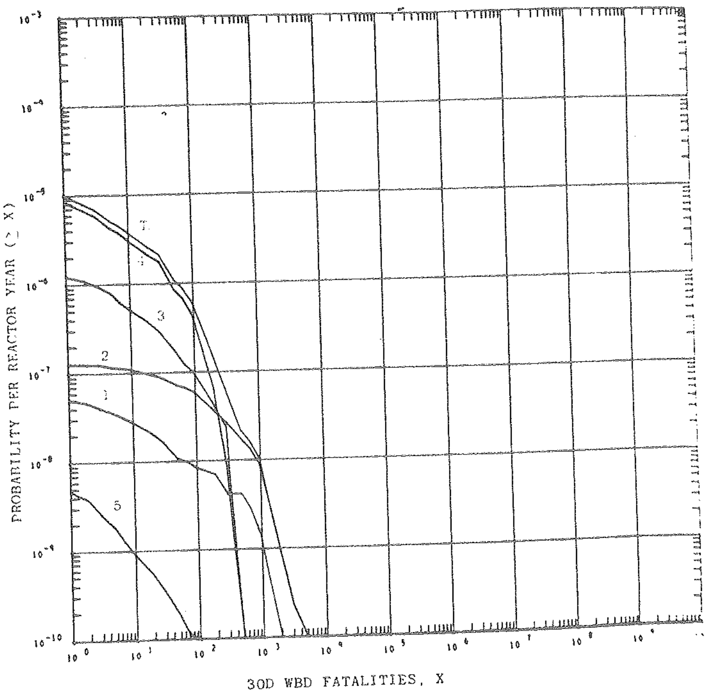

Figure 10F. Probability Distribution of Fatalities from 30 Day Whole Body Dose from BWR Accidents.

FIGURE 2-16 (Figure 10F of EPRI 217-2-6)

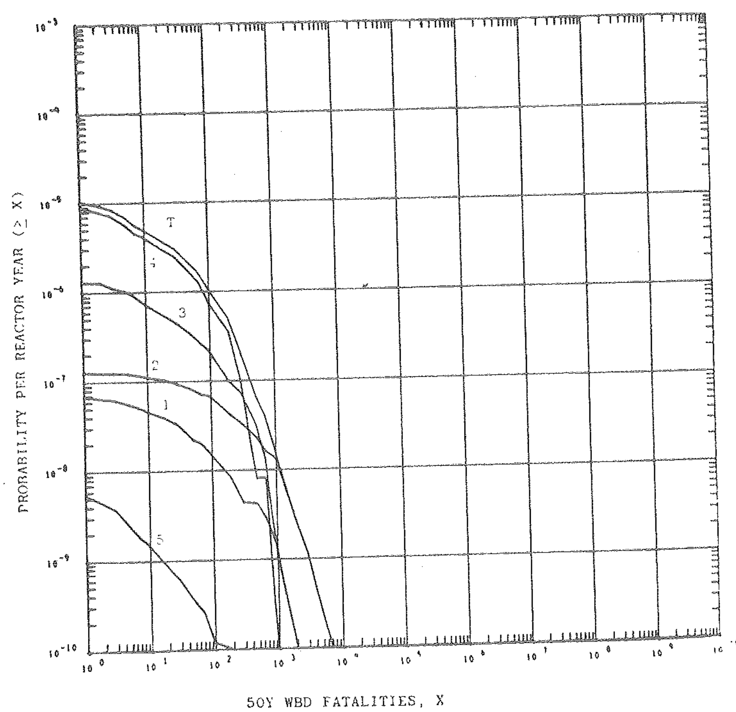

Figure 10D. Probability Distribution of Fatalities from 50 Year Whole Body Dose from BWR Accidents.

FIGURE 2-17 (Figure 10D of EPRI 217-2-6)

TABLE 2-12 (Table 10 of EPRI 217-2-6)

TABLE 10. Consequence Indicators for the PWR of WASH-1400, After Smoothing.

RELEASE CATEGORY

Consequence, C_i	T*	1	2	3	4	5	6	7	8	9
50Y WBD Man-Rem	2.31E+1	.008	.773	.215	0.	.001	.003	0.	0.	0.
Lung Man-Rem	1.64E+2	.010	.762	.225	0.	0.	.002	0.	0.	0.
30D WBD Man-Rem	1.29E+1	.009	.787	.198	0.	.001	.004	.001	0.	0.
Thyroid Man-Rem	7.92E+2	.007	.829	.145	0.	.002	.015	.001	0.	0.
50Y WBD Fatalities	1.83E-3	.007	.761	.231	0.	0.	0.	0.	0.	0.
Lung Fatalities	4.65E-4	.003	.665	.332	0.	0.	0.	0.	0.	0.
30D WBD Fatalitites	1.15E-3	.008	.801	.191	0.	0.	0.	0.	0.	0.
Thyroid Illnesses	3.71E-1	.008	.860	.119	0.	.002	.011	.001	0.	0.
Land Cost in Dollars	1.90E+4	.008	.860	.131	0.	.001	.001	0.	0.	0.
Evacuation Cost in Dollars	8.56E+3	.007	.848	.144	0.	0.	0.	0.	0.	0.
Total Cost in Dollars	2.49E+4	.007	.841	.150	0.	.001	0.	0.	0.	0.

$$*T = \sum_{j=1}^{9} \left\{ P(R_j) \int_{C_i(min)}^{C_i(max)} P(X|R_j)dx \right\}$$

TABLE 2-13 (Table 4 of EPRI 217-2-6)

TABLE 4. Consequence Indications for the BWR of WASH-1400.

RELEASE CATEGORY

Consequence, C_i	T*	1	2	3	4	5	6
50Y WBD Man-Rem	9.25E+0	.038	.033	.240	.689	0.	0.
Lung Man-Rem	6.31E+2	.053	.033	.276	.639	0.	0.
30D WBD Man-Rem	5.40E+0	.046	.037	.201	.714	0.	0.
Thyroid Man-Rem	4.15E+2	.034	.040	.102	.824	0.	0.
50Y WBD Fatalities	4.15E-4	.017	.098	.189	.696	0.	0.
Lung Fatalities	1.03E-4	0.	.095	.264	.640	0.	0.
30D WBD Fatalities	2.34E-4	.023	.128	.153	.695	0.	0.
Thyroid Illnesses	1.56E-2	.037	.050	.094	.818	0.	0.
Land Cost in Dollars	8.97E+3	.030	.037	.293	.634	0.	0.
Evacuation Cost in Dollars	3.69E+3	.025	.038	.241	.696	0.	0.
Total Cost in Dollars	1.28E+4	.025	.034	.262	.678	0.	0.

$$*T = \sum_{j=1}^{6} \left\{ P(R_j) \int_{C_i(min)}^{C_i(max)} P(X|R_j)dx \right\}$$

TABLE 2-14 (Table 11 of EPRI 217-2-6)

TABLE 11. Release Probabilities for Event Failures Contributing to PWR Sequence Probabilities.

	$P^*(E_m)/P(E_m)$	Release Category, R_i: 1	2	3	4	5	6	7
Base Case	1.00	5.28E-8	4.80E-6	2.00E-6	2.84E-11	1.44E-7	4.02E-6	2.65E-5
Event, E_m								
Human Error	0.05	3.82E-8	4.58E-6	5.80E-7	8.51E-12	5.89E-8	2.88E-6	1.51E-5
	0.10	3.89E-8	4.59E-6	6.54E-7	9.27E-12	6.34E-8	2.94E-6	1.57E-5
	0.50	4.50E-8	4.68E-6	1.25E-6	1.65E-11	9.89E-8	3.41E-6	2.05E-5
	2.00	6.93E-8	5.05E-6	3.51E-6	6.18E-11	2.33E-7	5.28E-6	3.87E-5
Test & Maintenance	0.05	2.24E-8	4.33E-6	1.76E-6	2.48E-11	1.37E-7	1.63E-6	2.10E-5
	0.10	2.40E-8	4.35E-6	1.77E-6	2.50E-11	1.37E-7	1.76E-6	2.13E-5
	0.50	3.68E-8	4.55E-6	1.87E-6	2.64E-11	1.40E-7	2.76E-6	2.36E-5
	2.00	8.48E-8	5.31E-6	2.28E-6	3.30E-11	1.51E-7	6.53E-6	3.24E-5
Pumps	0.05	4.88E-8	4.76E-6	1.80E-6	2.66E-11	1.43E-7	3.83E-6	2.62E-5
	0.10	4.89E-8	4.77E-6	1.81E-6	2.66E-11	1.43E-7	3.84E-6	2.62E-5
	0.50	5.04E-8	4.78E-6	1.86E-6	2.71E-11	1.43E-7	3.92E-6	2.64E-5
	2.00	6.05E-8	4.85E-6	2.54E-6	3.33E-11	1.44E-7	4.22E-6	2.68E-5

TABLE 2-15 (Table 6 of EPRI 217-2-6)

TABLE 6. Release Indicators for the BWR of WASH-1400

Plant Function Failure, F_k		Release Category, R_i: 1	2	3	4
QUV		0.0228	0.0956	0.0223	0.0264
W		0.7853	0	0.6127	0.9736
C		0.1919	0.9044	0.3650	0
Event Failure E_m	**$P^*(E_m)/P(E_m)$**				
HUMAN ERROR*	0.05	0.3476	0.9918	0.4972	0.1829
	0.1	0.3550	1.0266	0.5112	0.1829
	0.5	0.4140	1.3048	0.6235	0.1829
	2.0	0.6353	2.3479	1.0443	0.1829
TEST AND MAINTENANCE	0.05	0.1646	0.1879	0.1554	0.1772
	0.1	0.1650	0.1897	0.1558	0.1777
	0.5	0.1685	0.2041	0.1592	0.1817
	2.0	0.1814	0.2585	0.1718	0.1967
ALL HARDWARE	0.05	0.8366	0.2121	0.6845	1.0053
	0.1	0.8409	0.2145	0.6880	1.0105
	0.5	0.8754	0.2346	0.7158	1.0529
	2.0	1.0091	0.3282	0.8245	1.2169

* On the C tree, the two errors were of order 10^{-6} and 10^{-2} while the ones for W and QUV were of order 10^{-5} and 10^{-3}, respectively.

TABLE 2-16 (Table 8 of EPRI 217-2-6)

TABLE 8. System Indicators for the BWR of WASH-1400

E_m	$P^*(\tilde{E}_m)/P(E_m)$	System, S_k: QUV	W	C
Active Failures, Motor Operated Valves	0.05	0.4307	0.0076	No Change
	0.1	0.4331	0.0076	
	0.5	0.4528	0.0076	
	2.0	0.5302	0.0076	
Passive Failures, Motor Operated Valves	0.05	0.0780	0.0076	
	0.1	0.0781	0.0076	
	0.5	0.0782	0.0076	
	2.0	0.0814	0.0076	
Pump Failures	0.05	0.2954	0	
	0.1	0.2962	0	
	0.5	0.3028	0	
	2.0	0.3282	0	No Change
Test and Maintenance	0.05	0.9736	0.1557	0.1052
	0.1	0.9926	0.1557	0.1052
	0.5	1.1448	0.1557	0.1052
	2.0	1.7154	0.1557	0.1052
Human Error	0.05	0.5485	0.1730	1.038
	0.1	0.5485	0.1730	1.077
	.5	0.5485	0.1730	1.384
	2.0	0.5485	0.1730	2.537

fact that failure of decay heat removal is not strongly dependent on those listed, in spite of the fact that table 2-15 strongly suggests that this function is most sensitive to hardware failures. (That is, Table 2-15 shows that release category 4 depends strongly on decay heat removal and, more fundamentally, on hardware failures.)

SAI concludes from its work on sensitivity analyses that, as expected, the results are similar for BWR and PWR, and show the importance of human error and test and maintenance, but that there are situations where hardware failure has the dominant role and that such failure has more impact on overall risk than is indicated by a reading of (draft) WASH-1400.

2.2.3 Summary

We have briefly indicated two major areas in which EPRI-SAI is performing work relevant to the assessment of the risk from nuclear power plants. In large part, this work has been a response to and further development of the general approach of WASH-1400. It is important to note, though, that the reports issuing from this work are critical of certain aspects of the WASH-1400 work. Some of these criticisms were noted in section 2.2.1, but from the methodological development above, it is clear that a major shortcoming in WASH-1400, as viewed by SAI, is the presentation of the results. The probabilities and consequences, as given in WASH-1400, are not in a form that is useful for assisting in the identification of possible improvements in design or operation of reactors, leading to an increase in reliability or a decrease in risk. That WASH-1400 results should not be directly applicable for such purposes is not surprising, considering the fact that the purpose of the study was an assessment of the risk. Methods and results appropriate to such an assessment may not be in a form most useful for risk abatement. In fact, the authors of WASH-1400 specifically warn that their approach, designed as it was for the task of assessment, may not be more generally applicable.

From this point of view, the major work performed by SAI, as contractors to EPRI, may be seen as an adaption of the methodology of WASH-1400 for more general purposes. In particular, development of the more general fault-tree analysis capability described above is intended to remove certain *ad hoc* approaches taken in the course of the WASH-1400 work, particularly where common mode failures are concerned. And further, the sensitivity analysis described above is the first step in the improvement of design and operation. However, it is also clear that the more thorough presentation implied in such sensitivity analyses also aids in an understanding of the various ingredients implicit in the Reactor Safety Study's assessment, and therefore could usefully have been included in WASH-1400.

2.3. The Report to the American Physical Society by the Study Group on Light-Water Safety

2.3.1 Background and Objective of the APS Study

During 1973, the American Physical Society (APS) explored possible technical contributions which it could make as a society to the understanding of various aspects of the "energy crisis". This exploration resulted in APS sponsorship of three studies beginning in the summer of 1974, one of which was devoted to the subject of reactor safety. Support for this study was provided by the National Science Foundation and the Atomic Energy Commission - Nuclear Regulatory Commission. The twelve participants in the study began meeting in April 1974, spent the month of August in Los Alamos, and continued their work until completion of their report[5] during the spring of 1975.

This study differed in two major respects from the studies discussed in the previous sections. First, the study was not performed by an organization previously involved in the development of nuclear power (except perhaps, in that the basic understanding of the nucleus derives from an understanding of the physical world). Indeed, the participants in the study had widely varying degrees of experience with nuclear power, the primary point of their selection having been to bring together a group of individuals with high technical competence who could independently comment on important aspects of reactor safety, based primarily on an intensive examination of the technical and programmatic aspects of the subject.

Secondly, the study was not intended to "assess the risk" from present commercial nuclear power plants, but more generally to examine "reactor safety", a scope which had to be narrowed significantly, i.e., to light-water reactors, but which still left the study group with a range that was considerably broader than that of the Reactor Safety Study[1] or of the studies supported by EPRI.[4] In accordance with this broader mandate, the study examined both institutional and technical aspects of reactor safety and reactor safety research.

The narrowly technical material of the APS report is concentrated into three areas: 1) a discussion of events which may initiate accidents, 2) examination of the course of an accident, with special attention to LOCA phenomena (and the associated ECC systems), to containment behavior, and to accident consequences, and 3) an analysis of the light-water reactor safety research program. The report includes additional introductory and supporting material. Throughout the report runs an awareness of the importance of institutional questions, related to quality assurance, regulation, and safety research, that bear directly on the safety of nuclear power plants.

As viewed by the APS group, the basic purpose of reactor safety research and of the detailed nuclear regulatory apparatus was to design, build, and operate nuclear power plants in a manner that confines the large amounts of radioactivity in a reactor core sufficiently that the "safety" of the operators and public is assured. A primary aspect of safety design in reactors is the incorporation of redundant and diverse safety features, so that several relatively independent failures are necessary to generate a serious accident sequence. This design philosophy has less and less benefit at some level of complexity due to the increasing importance of "common mode" failures. Below this level of complexity, such safety features are expected to result in an accident spectrum in which the probability of occurrence decreases rapidly with accident severity. In such a situation, the design philosophy will have been justified. However, it does depend on the avoidance of substantial probabilities of common mode failure and, more straightforwardly, on the application of well-defined and conservative standards and criteria and on the design adequacy of the specific safety features of the reactor system.

Such adequacy has been difficult to assure in any general way. For the sake of simplicity, AEC (now NRC) licensing procedure have depended on a spectrum of "design-basis" accidents, formally defined hypothetical accidents which the safety features must be capable of handling. Due to the lack of understanding of hypothetical accident conditions, even the calculational methods to be used in analyzing accident situations are specified, the intention being that they make conservative assumptions at points of uncertainty. This formal approach to licensability does not rely on realistic assessments of behavior under accident conditions, the approach which the APS group recommended should be emphasized. In particular, this alternative approach would deemphasize the importance of the design basis accident, so that more realistic assessments of the safety of reactors, in terms of the full accident spectrum, could be made.

Such improvements in the licensing process would depend on the results of a reactor safety research program that was designed to this end.

Aside from the question of improving the safety of nuclear power plants through changes in design and licensing, based on the results of safety research, the APS group pointed out the importance of the complex interaction between the utility, the architect-engineer, the reactor vendor, and the AEC (now Nuclear Regulatory Commission) in the current implementation of nuclear power. Although it is the utility, the licensee, which bears direct responsibility for assuring the health and safety of the public, the accomplishment of this depends in practice on each participant in design, licensing, construction, and operation of the power plant. These phases, in turn, require proper design analysis and, in the end, inspection procedures to assure safe operation.

2.3.2 Initiating Events

Based on the fact that a light-water reactor core cannot support a nuclear explosion, the primary means for public harm would be melting of the core followed by a large release of radioactivity and substantial exposure of the public. Looking first at the events which may initiate accidents leading to melting, the study examined the following factors:

primary system integrity - a loss-of-coolant accident (LOCA) could be initiated by breach of the primary coolant system, the vessels and associated piping through which the water that actually cools the core flows. Failures in this system, including particularly the pressure vessel, were studied in detail. It was concluded that continuous and meticulous attention to inspection, maintenance, and operation can best guarantee the integrity of the primary system, including the pressure vessel.

transients - occasional departures from normal operating conditions are expected (anticipated transients), as well as events that are not expected (unanticipated transients). These include, for example, reactivity changes, equipment failure, and so on. Although most transients are handled by the reactor control system, some portion - about 10 per year, on the average - require shutdown of the reactor. The operations of reactor shutdown and subsequent decay heat removal afford opportunities for failure leading to core melting. The study group concluded that it is difficult to estimate the probability of such failures, particularly because of the difficulty of identifying all possible transients.

quality assurance - safe operation depends heavily on the high quality of components and the high reliability of systems. Quality assurance (QA) includes the procedures which assure that design and operating specifications are met in practice. The study group felt that QA problems may become an important limitation on nuclear reactor safety. For this reason, two questions were asked: 1) how adequate is the present level of QA and 2) how can a persistently high level be assured in the future expansion of nuclear power? Only the first question was directly addressed. The group asserted that no objective and quantitative measurement of the present QA system's effectiveness existed and recommended adoption of an objective measurement program. The report discusses possible forms for that program.

operator error - a significant number of licensee-reported abnormal occurrences, some with safety significance, are initiated or aggravated by operator error. This occurs in spite of the seemingly excellent qualifications of operator staffs, for two reasons: poor human engineering of the control room and control consoles, and limitations in what can be expected of operators in emergency situations. The report recommends improvements in the human engineering of reactor consoles, implementation of further automation of control sequences, and greater dependence on simulators for operator training.

sabotage - it is conceivable that saboteurs could act to release significant amounts of radioactivity from a nuclear power plant. The report concludes that it is difficult to conceive how they could initiate accidents any more severe than those which could in principle occur from equipment failure. However, because of the proximity of reactors to large population centers and based on considerations of possible ways to intentionally cause core meltdown and containment rupture, the study recommended more careful determination of the possible consequences of sabotage, of the cost-effectiveness of preventive measures, and of the effectiveness of mitigating measures, should sabotage occur.

2.3.3 Prevention and Mitigation of Radioactive Releases: Computer Modeling

Should an abnormality occur, various "engineered safety features" may be called into operation to control the event, preventing damage to the core or - should that be unsuccessful - mitigating any release of radioactivity. The APS study gave particular attention to the operation of the emergency core cooling systems (ECCS) and to the response of the containment to possible accidents. The emphasis on the ECCS arose from the basic understanding that substantial radioactive releases only occur as a result of insufficient cooling of the fuel, which - even when the reactor is shut down - must be cooled to prevent melting from either the stored energy (from the chain reaction) or from continuing generation of energy from decay of radioactive species produced in the course of the chain reaction. Assuming successful shutdown, the mere presence of water in the core would provide substantial cooling, due to natural convection. It is in the circumstance where the primary system is ruptured, with severe loss of coolant, that emergency cooling systems must be called into play. On the other hand, should these systems fail, the core would melt, throwing the burden of accident mitigation onto features of the containment. Minimum standards for ECCS performance are prescribed in the "Acceptance Criteria" that are part of the AEC (NRC) licensing procedure. The criteria go so far as to specify required and acceptable features of evaluation models used in predicting ECCS performance.

The report to the APS presented a moderately detailed discussion of the course of loss-of-coolant accident, first presuming that the ECCS performs as required. This discussion divides the accident conventionally into four time periods: blowdown, in which loss of the coolant inventory occurs, refill, the subsequent period during which loss of fluid through the break is negligible and the emergency coolant refills the pressure vessel to the bottom of the core, reflood, during which the coolant reaches a level that again covers the core, and long-term cooling, once cooling has been recovered. The temperature and structural behavior of the fuel and the dynamical behavior of the coolant are discussed for the various periods.

The acceptance criteria and the related evaluation models were examined qualitatively for adequacy. The nominal capacity of the ECCS systems was deemed sufficient for their purpose, but the critical questions had to do with actual system behavior under accident conditions. In particular, adequacy of the criteria specifications with respect to several model elements was examined: sources of energy in the fuel, the structural behavior of the fuel, dynamics and core thermal behavior, and the possibility of "steam binding" (prevention of reflood due to back pressure from the steam generators or pumps). The results of this examination are indicated in tabular form (table 2-17).

The specified evaluation models are regarded as key elements in evaluation of ECCS performance. These models are intended to be "adequately conservative" in areas where the physical processes are not well enough understood for realistic modeling. This leads to the questions whether they are indeed "adequately" conservative and whether the degree of conservatism can be determined quantitatively. On the last, the study noted that a realistic model would have to be available to test the degree of conservatism of the "conservative" models; however, the study felt that an adequate theoretical basis did not exist for either type of model, and the experimental work necessary to test their adequacy had not been performed.

The group identified the basic difficulty as the complexity of LOCA phenomena, which make it necessary that any computational code make great simplifications in describing the physical situation to make a "best estimate"! Rather than describing the reactor system structure as it actually is and coolant flows as they exist (including their complex time dependence), the system is divided into nodes, representing various volumes or parts of the system, and the manner in which they are connected is prescribed (based on the presumption that all changes in flow are gradual), thus yielding a very simplified representation of flow within the system. Due to the uncertainties inherent in such an approach, conservatism has been sought by prescribing conservative discrete elements of the evaluation model, on the presumption that the resulting overall model will yield conservative results. The correctness of this assumption has not been demonstrated, nor does it appear that present theoretical or experimental programs will lead to quantitative evaluation of the margin of conservatism.

According to the APS group, one fruitful approach to identification of parameters on which conservatism depends most strongly might be an extensive numerical parametric analysis of overall system results, carried out to evaluate predicted system response to systematic variations of parameters through the range of uncertainties for the individual model elements. Alternatively, efforts to simplify the problem of analyzing ECCS performance might be directed at development of alternative ECC systems to overwhelm the problem; one example would be to specify reflood rates much more substantial than those presently

TABLE 2-17 (Table IX of the APS report)

TABLE IX. Critical LOCA phenomenological behavior and conservatism of related ECCS criteria treatment. Criteria adequacy in relevant LOCA period.

LOCA parameters	Blowdown		Refill		Reflood	
A. Energy sources						
1. Initial stored energy		C[a]		C		...
2. Fission heat		C		NA		NA
3. Actinide decay		C		C		C
4. Fission product decay		C		C		C
5. PWR secondary-to-primary system heat transfer		OK/W		OK/W		OK/W
6. Metal–water reactions		S		S		S
B. Structural behavior						
1. Core loads		NT(NC)	NA (negligible fluid flows)		1. Fuel rod thermal shocks (clad brittle failure)	W
2. Steam generator tube loads/failure		NT(NC)				
C. Coolant fluid dynamics	1. Break flow/location	S	1. ECCS downcomer refill rate	OK	1. Reflood rate	W
	2. Frictional Δp	OK	2. Injected ECC fluid–steam interaction	S/OK	2. Fluid "carryover" from core	C
	3. Pump Δp	OK(WDB)			3. PWR steam binding	
					a. Pump Δp	S(WDB)
					b. Steam generator superheat	OK
					c. Steam generator tube leaks	NT(NC)
	4. Coolant "by-pass" of core	OK(WDB)			4. Blockage and radial flow	W(WDB)
	5. Blockage and radial flow	W				
	6. Computational models	W				
	7. Steam generator tube leaks	NT(NC)				
D. Core thermal behavior	1. Critical heat flux (CHF)/departure from nucleate boiling	C	1. Post-CHF heat transfer	C	1. PWR reflood heat transfer	OK
	2. Post-CHF heat transfer	C(WDB)			2. BWR core spray	OK
	3. Swelling and rupture of fuel cladding	S(WDB)			3. Core migration and pressure vessel melt through	S
	4. Fuel/clad thermal property models	OK(WDB)				
E. Containment structure influence						
1. Pressure buildup during LOCA		NT(C?)[b]		NT(C?)		NT(C)[b]
2. ECC related equipment survival (pumps, spray washdown, etc.)		OK		OK		OK

[a] Definition of symbols: S, strongly conservative; C, conservative; OK, adequate or "realistic;" W, weak; NT(NC) not treated in criteria—nonconservative omission; NA, not applicable; WDB, weak data base.
[b] Definition of symbols: NT(C?), not treated in criteria—conservatism of omission uncertain; NT(C), not treated in criteria—omission is conservative.

required, an alteration that would involve convincingly overcoming the problem of steam binding by some means.

The basic view of the APS group was that the experimental data available were not sufficient to demonstrate the overall system conservatism of ECCS criteria, nor was the current generation of codes adequate to evaluate system effectiveness. In view of the fact that no large-scale tests were planned, greater confidence in calculational methods must be gained through a much stronger code development program, supported by a much improved experimental data base, and coupled with a strong program in code assessment and evaluation.

Should a LOCA occur, systems within the containment building are available to remove heat and radioactivity released into the containment. These would also operate should effective emergency core cooling fail, but in this case the amounts of heat and radioactivity released would be much larger than otherwise, and it is presumed that the molten core would eventually breach containment (within about a day). The APS group questioned the soil filtration attenuation factor which was assumed by draft WASH-1400 for radioactivity reaching the atmosphere in instances where the core melts through the bottom of the containment. The group also urged consideration of: design of containment for controlled failure at some pressure through a filtration system, underground siting, core catchers (to prevent containment melt-through), and improvement of the reliability of various containment systems.

2.3.4 Radioactive Releases and Their Consequences

The study examined the various classes of radionuclides which would be released from a molten core, and then considered - in a simple "wedge" model - the transport of released radioactivity and the resulting human exposures and consequent health effects. The wedge model simply assumes that any radioactivity released proceeds uniformly in velocity and directional dependence away from the source within a specified opening angle and within the "mixing layer" of the atmosphere. This specifies a wedge within which concentrations of radioactivity will be confined and wherein the time-integrated radioactivity will depend only on distance from the reactor. Although such a model would not be applicable to determination of early deaths and illness, effects which depend critically on the details of local meteorology and population densities, it might be expected to yield useful results for latent effects, which are presumed to depend only on the integrated dose in man-rem.

The study specifically applied this simple model to a release and release conditions chosen to simulate one accident whose consequence calculations were described in detail in draft WASH-1400. This draft was released during the August 1974 meeting of the APS group. For this large release accident (PWR release category 2; see section 2.1), the APS study found that the WASH-1400 draft has seriously underestimated the total whole-body population dose, with a particularly serious omission being the neglect of most of the dose from ^{137}Cs deposited on the ground. This correction alone led to an increase in the number of latent cancer deaths by a factor of 25, an increase which caused such deaths to dominate the total number of fatalities from the hypothetical release. (We may note here that the final WASH-1400 accounted for this source of exposure, thereby yielding the result that it dominated the total number of fatalities, but that the final WASH-1400 numbers were not as high as suggested by the APS study.) Using the wedge model, the APS group also estimated other long-term consequences, most yielding larger numbers than those given in the draft WASH-1400. (The required alterations are shown in table 2-18). With the wedge model, the connection between the various parameters affecting release consequences is particularly transparent. For example, the influence of evacuation assumptions on total population dose is very easy to assess, presuming that the assumed population distribution is simple (and uniform). In many cases, corresponding changes were made in the WASH-1400 final report.

In addition to the use of evacuation, sequestering of contaminated food, and decontamination for reducing the consequences of a release, the study urged investigation of: use of iodine blocking (based on distribution of pills) to reduce uptake of radioiodine, installation of air filtration systems on large buildings near to nuclear power plants, and distant siting to reduce population at risk.

2.3.5 The Light-water Reactor Safety Research Program

A major part of the APS study was an examination of the then current (late 1974) reactor safety program, largely to seek an understanding of the extent to which the program could hope to elucidate reactor behavior during a loss-of-coolant accident. The main burden of research on the safety of light-water reactors was carried by the AEC Division of Reactor Safety Research (RSR - now part of the Nuclear Regulatory Commission), whose efforts are supplemented

TABLE 2-18 (Table XIV of the APS report)

TABLE XIV. Effect of changed assumptions on Draft WASH-1400 reference accident average man-rem consequences.

Assumption change [a]	Relative effect	Total effect [b]
A. Cancer Deaths		
1. Truncation of ground dose at one day... to... no truncation	Factor of about 25	10 000 cancer deaths instead of 310 cancer deaths
2. 100 cancer deaths/(10^6 whole body rem)... to... (about 130)/(10^6 whole body rem)	Factor of 1.3	
3. Neglect of cancer induction from lung dose... to... (20-50)/(10^6 lung rem) over 40 years following exposure	Not calculated in Draft WASH-1400	Add 600-1600 lung cancer deaths during following 40 years
4. Neglect of deaths from beta-ray induced thyroid cancer... to... (12-75 thyroid cancers)/(10^6 thyroid rem) over 30 years following exposure, use range of AEC-EPA conversion factors, assume 4% mortality for persons exposed as children, 15% for persons exposed as adults.	Not calculated in Draft WASH-1400	Add 500-4 000 thyroid cancer deaths in 30 years following exposure
B. Morbidity		
1. Show full range of uncertainty of thyroid exposure-dose and dose consequence coefficients	Factor of 0.9-12.0	22 500-300 000 thyroid nodule cases instead of 25 000 (our estimate—not explicitly stated in Draft WASH-1400)
C. Genetic Defects		
1. Truncation of ground dose at one day... to... no truncation	Factor of about 25	3 000-20 000 genetic defects instead of 310 genetic defects
2. (100 genetic defects)/(10^6 whole-body rem)... to... (25-250) identifiable dominant genetic defects and 12.5 noninheritable genetic defects/(10^6 whole-body rem). Not included are 0-500 additional constitutional or degenerative diseases/(10^6 whole-body rem)	Factor of 0.4-2.5 in genetic defects	

[a] ... indicate assumption change from... to... .

[b] It is essential to note that these incremental cancer deaths and morbidity would occur over natural lifetimes of a very large exposed population. These calculations are based on a population density of 300/mi^2 which results in an exposure of a population of about 10 million. A population density of 165/mi^2 would multiply all total consequence estimates by 0.55 (See Footnote, page xxxx). Independent of assumed population density, there would be an additional risk of cancer to the average exposed individual of one chance in a thousand, with the risk distributed over a substantial fraction of his natural lifetime.

in a major way by the reactor vendors and by the Electric Power Research Institute. The RSR program on LWRs was based primarily on: separate effects tests of various components and subsystems, computer code development which uses information developed in the separate effects tests, integral system experiments on a scale smaller than an LWR but hopefully large enough to test the predictive ability of the codes, and primary system integrity tests. The long-term goal of the research program is to be able to understand and quantitatively predict the important safety aspects of reactor behavior.

The report delineates the 1975 breakdown of funding for the individual research areas, then discusses the work being performed in each area. The program on primary system integrity emphasized the physical bases of integrity and also the attention given to quality assurance and inspection, areas which the APS group would indeed emphasize. The separate effects tests can be associated with the various elements of LOCA phenomena, as detailed in the acceptance criteria (see, for example, the phenomena mentioned in table 2-17), and the study examined the extent to which our understanding of these phenomena is being strengthened. Although the number of separate effects tests being performed has increased in recent years, the group recommended a further increase in such experiments, particularly on ECC bypass, heat transfer with cross flow, non-equilibrium two-phase flow, and core blockage. Such experiments relate strongly to code development and system testing. Of particular importance in the research program was that tests of fuel and cladding problems at the Power Burst Facility proceed on a timely basis.

The major integral system testing program has been planned at the Loss of Fluid Test (LOFT) facility. The LOFT system is a specially designed and instrumented PWR of intermediate scale (55 MWt) relative to a large commercial PWR (3300 MWt). Objectives of the LOFT program are: 1) to provide data for testing the adequacy of analytical models which predict transient response of the core, primary system, and coolant and which predict the capability and the margin of safety of current ECCS designs, 2) to verify the adequacy of the design criteria used to establish ECCS capabilities, and 3) to reveal thresholds or unexpected phenomena affecting either the validity of analytic models predicting transient response or the selection of ECCS design parameters.

A crucial issue is the relationship between LOFT system parameters and those of a PWR, i.e., the matter of scaling. Since complex transient two-phase flow is not well enough understood to develop useful scaling criteria through the definition of characteristic dimensionless numbers, the basic criterion chosen is volumetric and power density scaling, basically to assure that the same relative amounts of fluid are available for energy exchange in LOFT as in a PWR. On the other hand, an attempt is made to scale break areas and fuel cross sections. It is clear that volumes and areas cannot be scaled simultaneously. Thus, although LOFT results may be representative of a PWR in some respects, in others it will not. As a result, LOFT cannot be regarded as a test of the PWR ECCS, but as an integral test of the many separate effects important in LOCA phenomena. The group urged that the test program be kept flexible enough to take into account new results and also to test alternative ECC concepts. In a similar vein, it was urged that care be taken that the test program not be restricted to initial conditions associated with normal PWR operation, but also allow for the possibility of abnormal initial conditions existing prior to the onset of loss-of-coolant.

Scaling compromises have made the use of LOFT data for code verification a difficult issue, particularly because of the nodalized character of present calculation, which often requires adjustment of parameters to obtain agreement. Significant comparisons with data from LOFT and, certainly, dependable scaling to a PWR will require codes with more realistic physical treatment and fewer arbitrary parameters. The APS group questioned the ability of the then current experimental and code development programs to achieve a satisfactory quantitative understanding of phenomena at full PWR size. Moreover, no test comparable to LOFT exists for the BWR.

Design of the experimental program aside, the ability of the LOFT system actually to make the intended measurements, was examined in terms of the instrumentation on the system. Core measurements include many temperature sensors, several absolute and differential pressure sensors, and mass flow and velocity sensors. Neutron flux is monitored by power measurements. In each instance, the group identified some deficiency either in the intended measurement, the accuracy of the instrumentation, or its calibration. It was particularly critical of the absence of direct measurement of the core water level. The primary system measurements relied on temperature sensors, pressure sensors, a liquid level monitor, a density monitor, and mass flow indicators. For these systems, too, the group questioned their ability to make the required measurements. Overall, they recommended a major effort to upgrade the quality, range, quantity, and redundancy of LOFT instrumentation, so that measurements appropriate to testing code calculations would be possible.

An earlier companion of LOFT is "Semiscale", a small, essentially one-dimensional system, with a 5 ft. long 1 MWt electrically-heated core, intended to provide data for basic LOCA model development and for LOFT design assessment and instrumentation evaluation. This is the device which first gave indications of the possible importance of ECC bypass, the phenomena where injected coolant bypasses the core and goes out the LOCA-inducing break. Although Semiscale is very small and one-dimensional, parametric measurements should be important when compared with LOFT results.

(The report goes on to discuss current and suggested research in containment system response and radiological consequences. Although containment systems have worked well to control routine releases, there are major uncertainties in containment conditions during severe accidents - involving, for example, core meltdown - and in corresponding effectiveness of specific systems. The research program should be upgraded to examine and improve the reliability and effectiveness of specific containment systems and to examine some of the possible changes mentioned in the previous section. In a similar way, the consequences of meltdown and containment failure could be much better understood as a result of increased efforts in the areas of radionuclide release from molten fuel, dispersion through soil and water, meteorological analysis, mitigation of biological effects, and decontamination effectiveness.)

The view of the APS group is that a major consideration in the LOCA analysis portion of the light-water reactor research program is not the specific work, either experimental or theoretical, that is carried out, but the manner in which it is carried out. There is an especially delicate connection between code development and experiment: the new generation of more realistic and complex codes depends on the results from separate effects tests, and their predictive ability must be evaluated at LOFT, a sub-PWR-scale test. Based on the extent to which their predictions are verified, some judgment can be made as to the strength of an extrapolation, using these codes, to PWR scale. Such a verification at LOFT scale is critical, rather than the alternative of falling back to a position where LOFT simply tests whether the conservative evaluation model codes are conservative at the LOFT scale, a test that would say little about their conservatism at the PWR scale and that would seriously degrade the contribution of LOFT to a physical understanding of systems at PWR scale.

As a result the more realistic codes must have made predictions prior to the LOFT series.

The APS study also emphasized the limitations of the tests and codes being planned. For example, LOFT can not be thought of as a test of a PWR system. Furthermore, however complex a code, it will be applicable within limits and will be unable to handle unanticipated phenomena. It is, moreover, important that the assumptions on which they are based be openly available for review by the scientific community. This is especially true of the conservative evaluation model codes, for which the APS group thought the basic premise to be extremely weak, i.e., that making conservative assumptions at each point may lead to a net conservative result, even beyond the range where the code has been tested.

Even with the results from the separate effects tests, the LOFT program, and advanced code development as now planned, the group was skeptical of the ability to scale our understanding of transient conditions to PWR (and also BWR)scale. They identified several possible, and perhaps complementary options for the reactor safety research program: 1) Limit scope to the present program, somewhat improved as suggested above, 2) Augment the research program by pushing the investigation and development of alternative concepts to cope with LOCAs and with other transients of concern, thus obtaining analyzable concepts, 3) Augment the present program by larger integral system tests combined with a move toward standardization of LWR designs, thereby making the scaling question moot, 4) Augment the research program by placing additional emphasis on better containment, consequence mitigation, and accident recovery, research emphases that should be adopted independently of the above choices, 5) Complement the present program through emphasis on remote and other conservative types of siting, sharply reducing risk.

The specific recommendation of the APS group was that the research program be extended to implement fully options 2 and 4 and that plans for 3 be drawn up as an alternate program should options 1 and 2 not be leading to success on ECCS understanding. It was thought to be particularly important that the experiments and advanced code work proceed with the help of the best qualified scientific personnel in the country.

2.3.6 The Major APS Recommendations

In the words of the APS report:

"... we have not uncovered reasons for substantial short range concern regarding risk of accidents in light-water reactors ...we are confident that a much better quantitative evaluation and consequent improvements of the safety situation can be achieved over the next decayed if certain aspects of the safety research program are substantially improved and the results of the research are implemented" (our emphasis added).

Summary of conclusions and major recommendations S7

D. Major recommendations

Many recommendations are made in the body of this Report. A few of the major ones are summarized here, but in each case the reader is referred to the main text for detailed discussions of the background and rationale. Our major recommendations, which have not been ranked according to their importance, include the following:

(1) Human engineering of reactor controls, which might significantly reduce the chance of operator errors, should be improved. We also encourage the automation of more control functions and increased operator training with simulators, especially in accident-simulation mode.

(2) Measures should be taken to quantify the effectiveness of the present quality assurance program, using both the analysis of experience already reported and new measurements on the quality assurance system.

(3) The techniques used in Draft WASH-1400 for the calculation of accident sequences and their probabilities should be:

- employed to estimate quantitatively whether assumed subsystem failure data are compatible with the observed individual small accidents;
- used to provide parametric studies of the effects of phenomena which are ill-understood in the identified sequences;
- refined so that they can be used for continuing risk assessment on a routine basis with a growing data base of failure data.

(4) The Draft WASH-1400 analysis of accident consequences should be redone taking into account the modifications discussed in our report, in order to obtain corrected consequence estimates. The results will help to determine the magnitude of the benefits which might be obtained from the introductions of design changes and means of mitigation of accident consequences.

(5) The problem of sabotage and its effect on increasing the risk of radioactivity release should be studied carefully. We have no way of estimating the present likelihood of sabotage; however, we believe that reactor security can be improved and have specific recommendations for studies that go beyond those already underway.

(6) The ECCS safety margin should be quantified, and if necessary, improved through one or more of the following approaches:

- the substitution of more easily analyzable or more effective ECCS concepts;
- a much stronger theoretical and calculational development effort combined with a much improved experimental program, the results of which must be published openly for evaluation by the technical community;
- a series of large-scale experiments along with some standardization of reactors. Detailed planning and analysis for this approach should begin immediately in case it should be decided in the future that it is needed.

There should be increased emphasis on realistic calculations and experiments as opposed to those which merely attempt to set upper limits on the behavior of a reactor in an accident. In view of the number of reactors now operating and being planned, we believe it is important that the reactor safety research program quickly take major steps to bring about a convincing resolution of the uncertainties in EECS performance.

(7) In the area of safety research, more emphasis should be placed on seeking improvements in containment methods and technology. In particular, controlled venting of the containment building in case of overpressure should be studied. A careful assessment should also be made of the benefits and costs of alternative siting policies, such as remote, underground, and nuclear-park siting.

(8) There should be more effort to resolve major uncertainties in estimating consequences, including improvement of the biological-effects data base. Techniques for mitigation of consequences should be developed, especially in connection with the problems of decontamination after a large accident.

(9) While we strongly endorse the substantial improvements that have been made in the safety research programs and in the openness to scrutiny by the technical public in the last two years, additional measures should be taken to continue to improve the research program and techniques and to assure that the results of both experimental and computer code development work related to safety are openly published.

References for Section 2

1. "Reactor Safety Study: An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants", U. S. Nuclear Regulatory Commission Report WASH-1400 (NUREG-75/014), in 9 volumes, October 1975 (available from NTIS).

2. "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants", USAEC Report WASH-740, 1957 (available from NTIS).

3. National Academy of Sciences - National Research Council, Report of the Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR), "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation", November 1972.

4. "Summary of the AEC Reactor Safety Study (WASH-1400)", EPRI 217-2-1, April 1975; "Generalized Fault Tree Analysis for Reactor Safety", EPRI 217-2-2, June 1975; "Critique of the AEC Reactor Safety Study (WASH-1400)", EPRI 217-2-3, June 1975; "Probabilistic Safety Analysis", EPRI 217-2-4, July 1975; "User's Guide for the Wam-Bam Computer Code", EPRI 217-2-5, January 1976. All of these are Electric Power Research Institute Reports, prepared by Science Applications, Inc., and available from NTIS.

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3. OTHER STUDIES

3.1 Studies Completed Prior to WASH-1400: Mostly WASH-740

Few assessments of the safety of nuclear power were completed prior to the work described in the previous sections. The most important example, yielding a report entitled "Theoretical Possibilities and Consequences of Major accidents in Large Nuclear Power Plants"(AEC report WASH-740[1]), was performed in 1957 at Brookhaven National Laboratory for the Atomic Energy Commission. Its purpose was to assess the "possibilities" and consequences of accidents at the relatively "large" commercial nuclear power plants that were being planned at the time of the report. An important distinction is that an identification of "possible", that is "conceivable," accidents does not carry the connotation of assigning probabilities to those accidents. The report does indicate ranges of probabilities for three classes of accidents, but these are based on estimates of experts who would venture an opinion, not on any probabilistic modeling. A second clarification of the stated purpose of the study is that the "large" plants considered were 500 megawatts thermal, about one sixth the size of current large nuclear power plants (3000 MWt or about 1000 MW electrical). The growth in plant size has led to possible invalidation of one of the main assumptions of WASH-740: that a molten core would, under most circumstances, be indefinitely contained by the pressure vessel, and certainly by the containment structures. As noted in section 2, precisely the opposite assumption is now made for the large plants, the reason being that the larger volume of molten material would have insufficient surface area for heat transfer.

The study explicitly treated three classes of accidents:

Class 1, the contained case, assumes that all of the fission products are vaporized and dispersed within the containment shell, but that there is no release to the atmosphere. (Estimated probability of occurrence: 10^{-2} to 10^{-4} per reactor-year)

Class 2, the volatile release case, assumes that all volatile fission products are released to the atmosphere at the time of the accident. (Estimated probability of occurrence: 10^{-3} to 10^{-4} per reactor-year)

Class 3, the "50 percent" release case, assumes that 50 percent of all fission products are released to the atmosphere. (Estimated probability of occurrence: 10^{-5} to 10^{-9} per reactor-year)

In each case, specific accident mechanisms were not treated, but the stated releases were presumed. The study then calculated early fatalities (within 15 miles) as well as costs of evacuation and of contamination of the foodstuffs and land, and illness for each accident class. The assumed population distribution was designed to be characteristic of a reactor sited at a distance of 30 miles from a city.

In cases involving releases, a range of assumptions affecting dispersion and exposure was made, yielding widely varying consequences. The parameters varied included weather conditions (inversion or not, rain or not), size of the radioactive particulates dispersed, evacuation speed, and amount of heat released with the radioactivity.

The class 1 accident exposes surrounding populations to gamma radiation that penetrates the containment. On the basis of the typical population distribution beyond an assumed 2000 foot site boundary, it was found that no fatalities would occur, and several illnesses could occur if evacuation was slow.

Class 2 accidents are major releases, with deaths ranging from 2 to 900 and illnesses from 10 to 13,000, depending on the specific parameters used. Property damage ranged up to approximately $0.5 billion (1957 dollars).

Class 3 accidents, even larger releases than those above, were estimated to kill 0 to 3400, produce illness in 0 to 43,000, and cause damages up to $7 billion. (The heat released in this accident type presumably accounts for a lower limit on deaths that is below that of Class 2.)

All the harm to individuals specified above is in the form of early death or illness. <u>The study did not consider latent effects</u>.

In a sense, the accident leading to 3400 early fatalities was regarded as a maximum conceivable accident. It was not literally so, since the group could imagine worse combinations of the various parameters. However, even given a class 3 release, the combination of conditions resulting in 3400 deaths was itself very unlikely. As indicated above, the study did not make an attempt to calculate probabilities associated with a spectrum of accidents in order to assess the overall risk. The purpose of the study was much less ambitious, to indicate the range of possibilities and magnitudes for releases from large nuclear plants.

Almost a decade later, workers at Brookhaven again examined the question of the consequences of major releases from nuclear power plants, this time from the substantially larger plants that were beginning to dominate the scene. The results of their work to revise WASH-740 did not reach the form of an AEC report. However, in view of the fact that they were not attempting to apply any substantially different methodology than was used for WASH-740, it is to be expected that their results would be similar, and they were. The major alteration was that the overall magnitude of the consequences from a 1000 MWe power plant would be about an order of magnitude greater than that for smaller plants, largely because the inventory of radioactivity in the core is proportional to the rate of heat generated.

The only other study of note that bears similarity in approach to WASH-740 was an investigation[2] of the consequences of a major release from a 60 MWe demonstration liquid metal fast breeder reactor power plant known as Fermi 1 and located on the outskirts of Detroit. Such reactors are beyond the scope of our discussion. However, once the release of fission products has occurred, an accident at such a plant would be largely indistinguishable from that at an LWR plant. Because of the proximity of the plant to Detroit, the potential consequences of a major release were found to be very severe, the maximum number of early fatalities exceeding 100,000.

Except for the reports just mentioned, little public information has been developed, until recently, relevant to an overall assessment of the safety of nuclear power plants. Perhaps the most voluminous such information resulted from the rule making hearings[3] on the ECCS acceptance criteria. (The APS report[4] discusses these criteria.) The subject of the hearings was the interim acceptance criteria, with amendments, for ECC systems for LWR power plants. The AEC promulgated revised acceptance criteria in late 1973.[3] Such detailed ECCS specifications arose out of many years' consideration of how such systems should be implemented. One of the basic documents from this period is a 1967 report to the AEC by Ergen _et al._[5] on emergency core cooling systems.

During the same period that specifications for ECC systems were being developed and reviewed, reports appeared which gave an inkling of the future direction of risk assessment. One example is a paper by Otway and Erdmann,[6] discussing reactor siting and design from a risk viewpoint, wherein consequences versus probabilities were displayed and the idea of calculating the probability of failure of the reactor system from the probability of failure of component systems was outlined. During this same period, the AEC was, of course, beginning the study that resulted in WASH-1400.[7] Moreover, the AEC was also assembling information on the safety of nuclear power reactors, but one resulting report (WASH-1250[8]) is best described as a discussion of a light-water reactor nuclear power system and of various health and safety criteria pertinent to such a system, rather than as an analysis of the actual risk from LWRs. However, it may be said that it was the AEC awareness of the public concern for the safety of nuclear power (an awareness indicated by the publication of WASH-1250 as well as by many other events, such as the criteria hearings), coupled with the increasing interest in probabilistic methodologies, that led to the work reported in WASH-1400.

3.2 Reviews of WASH-1400

Because of the seeming primacy of WASH-1400 among various studies of light-water reactor safety, it is appropriate that we indicate briefly, but explicitly, many of the public comments on that work. A summary of the comments on the WASH-1400 draft was given in section 2.1.4, and, in the course of our discussion of the EPRI work and the APS study, we indicated their major comments on WASH-1400. Of the numerous other comments, we will treat those of the U.S. Environmental Protection Agency in section 3.2.1 and summarize some of the remainder in section 3.2.2.

3.2.1 Reviews by the U.S. Environmental Protection Agency

The Office of Radiation Programs of the Environmental Protection Agency (EPA) conducted reviews of the draft and final WASH-1400 reports. Results of these reviews were published in 1975[9] and 1976[10]. In each instance, the EPA tended to concentrate its own internal review on the calculation of consequences, particularly health effects, as presented in WASH-1400. The EPA relied on contractual work with Intermountain Technologies, Inc. (ITI) to review the WASH-1400 accident analysis.

The EPA reviews identified "several significant areas in which we have found the WASH-1400 report either deficient or containing unjustified assumptions. These are

1) failure to address fully the health effects expected after an accident and to consider adequately a technical basis, which

includes a broad range of perspectives, for estimating the incidence of the associated bioeffects,

2) the assumptions made in regard to evacuation as a remedial measure,
3) improperly or incompletely evaluated parameters used in determining the accident event-sequences and probabilities, and
4) inadequate description of the analysis of the consequences of the release of radioactive materials to the environment."[10]

Area 1 was a major source of dissatisfaction in both reviews. In particular, the EPA contended that the final WASH-1400 lower bound, central, and upper bound estimates for the relationship between exposure and latent health effects should all be altered in ways yielding larger consequences, thereby increasing the middle estimates by a factor of 2 to 10.

The second deficiency arose a) from WASH-1400 use of a constant radius (25 miles) within which evacuation takes place, a simplistic assumption that is inconsistent with present and planned practice (which would relate evacuation procedures more directly to the details of a particular release), and b) from the assumed timing and speed of evacuation, the details of which are not clear from the discussion in WASH-1400.

Third, in both reviews, ITI identified the analysis of the BWR transient-without-scram accidents as the most significant accident analysis problem in WASH-1400 and stated that re-evaluation may increase the risk of BWR accidents. Moreover, the reviews pointed out numerous instances of information insufficient to assess risk impact, both in the details of quantification of accident probability based on human error, common modes, and so on, and in the description of core meltdown and containment response (analysis of ECCS functionability appears incomplete, and the PWR containment failure pressure appears too high).

Lastly, the EPA regarded as deficient the description of how the analytical framework for consequences calculation was applied. There was also too little information on results obtained at intermediate steps in the calculation, causing difficulties to others who would use these techniques or trace their application in WASH-1400.

The EPA supported the concept of the Reactor Safety Study. However, it found that what it considers "more reasonable assumptions in health effects, emergency actions, and estimates of probabilities of releases" would significantly alter the results of the study. Finally, the EPA was concerned about the implied acceptability of the estimated risks to society that is contained in the comparisons made between risks from nuclear reactors and from other sources.

3.2.2 Other Reviews

The comments on WASH-1400 to which we have so far referred have dealt gently with the basic methodology of that study. Although details of the probabilistic accident analysis have been questioned, particularly the treatment of common mode failures and the often-cited lack of clarity on how specific factors contribute to risk, neither the EPRI work, the APS study, nor the EPA review found basic inadequacies in this approach to risk assessment. The most serious criticism offered was by the APS group, who said "based on our experience with problems of this nature involving very low probabilities, we do not now have confidence in the presently calculated absolute values of the probabilities of the various branches."[4]

Others have sharpened this point of view to a fundamental attack on the WASH-1400 methodology. One of the earliest public reviews of the draft report was the "Preliminary Review of the AEC Reactor Safety Study,"[11] published by the Sierra Club and the Union of Concerned Scientists (the UCS being one of the protagonists in the acceptance criteria hearings). This review strongly criticized WASH-1400 both in its application of the event-tree-fault-tree methodology and in its calculation of consequences of the identified release categories.

The Sierra-UCS report presented the basic view that experience shows that the use of fault trees has not been dependable for prediction of absolute failure rates, a fundamental requirement of the WASH-1400 risk analysis. This distrust of the methodology was based on two general criticisms: first, the review committee regarded it as unlikely, and even impossible, that all important accident sequences were identified or that all common mode failures were identified; second, the review doubted the dependability of the failure rate data, a doubt based on questions regarding design adequacy, human failures, rarity of many of the failures (particularly of structures), the importance of secondary (stress) failure, and the pressure vessel failure data.

The review also concluded that the human consequences of the major accidents treated in WASH-1400 were understated by at least an order of magnitude. The committee identified as causes for this discrepancy: an understatement of the amount of radioactivity released in given core melt accidents, underestimation of the damages to human health from radioactive exposure, an overstatement of the effectiveness of evacuation and shielding in reducing exposures, and a neglect of expected population growth in the vicinity of the identified sites.

Strong contrasts were drawn between the conclusions of WASH-1400 and the previously accepted assessments of the risk from large loss-of-coolant accidents and/or core melting. The report noted the substantially increased probability of core melt accidents as calculated by WASH-1400 and the decreased typical human consequences of such accidents, as compared with WASH-740 and its update (see section 3.1). Furthermore, the primary contributor to meltdown accidents was found by WASH-1400 not to be large LOCAs, thereby suggesting a misplaced emphasis in the AEC (now NRC) approach to regulation of nuclear power, an approach which strongly emphasizes the large-LOCA design basis accidents. Finally, the Sierra-UCS review committee regarded the AEC public use of draft WASH-1400, with its uncertainties and errors, to be improper, particularly because of the lack of opportunity for prior review.

We should emphasize that this review dealt with the 1974 draft. However, similar criticisms,[12] including even that of improper use of the results of the study, have been leveled against WASH-1400 since publication of the final version in late 1975. A useful selection of comments on the Reactor Safety Study were made at Congressional hearings[13] during June 1976 before the Subcommittee on Energy and the Environment of the Committee on Interior and Insular Affairs. Testimony was heard from the Nuclear Regulatory Commission and the Environmental Protection Agency, as well as from a number of individuals representing a wide range of opinion of the accuracy or utility of WASH-1400. Those giving testimony included* Norman C. Rasmussen, Saul Levine,

*For clarity in the context of this report, these individuals are listed in the order in which the studies or reviews with which they were associated were discussed in this report.

Niel Wald, M.D., and Marvin Goldman, participants in the Reactor Safety Study; R. C. Erdmann, participant in the EPRI-SAI work; Frank von Hippel and W. K. H. Panofsky, participants in the APS study; Dr. William Rowe, involved in the EPA reviews; and Henry W. Kendall, involved in the Sierra-UCS review.

It is not possible to state briefly the various comments and responses that were made during these hearings (or, indeed, that are being made in the continuing discussion of the Reactor Safety Study). Many points raised have been discussed previously in this report. Two items from the hearings may usefully be mentioned. The first is the particular attention given by participants in the Reactor Safety Study to the criticisms voiced by Rowe in behalf of the EPA; they replied during the hearings to each of the main points listed in the last section. They especially emphasized the reasons for their choice of dose-response relationships for latent health effects.

The second item is the somewhat delicate question often raised of the manner in which the results of the Reactor Safety Study were stated in WASH-1400 and the manner in which that report is being used and may be used. The Executive Summary, the several pages at the beginning of WASH-1400, is intended to summarize the results of the study, presumably for a general audience. In doing so, the three page "Introduction and Results", and the questions and answers which follow, appear to many observers to intentionally state the results so as to minimize the consequences of nuclear accidents, particularly where latent fatalities are concerned. Since it is difficult to compare such latent effects with similar effects from other sources, the summary emphasizes its comparison of estimated early fatalities (although the summary's graphical displays omit the word "early") with those from other sources. About latent effects, it says "The number of cases of genetic effects and long-term cancer fatalities is predicted to be smaller than the normal incidence rate of these diseases", neglecting to note that estimated long-term fatalities constitute 99.9% of total estimated fatalities from reactor accidents.

A second aspect of the manner of use of WASH-1400 is the limitations of the study. One easy observation is that the study did not deal with reactor types other than LWRs. Application of the methodology to other types would be useful as an assessment tool. A second observation is that the study eliminated any site-specific results. Construction of its "typical" sites, with corresponding meteorological conditions and population distributions eliminated the possibility of assessing the range of risk, as it varies from one site to another. (Possible consequences from accidents, for example, in the vicinity of large population concentrations, such as New York City, are of great interest.) WASH-1400 acknowledges these observations, restricting itself to an assessment of the overall risk from the nuclear power system. The application of the methodology, presuming its reliability, to other reactor types and to specific sites would increase the amount of information available for risk assessment. As discussed below, some efforts in this direction are being made by the Nuclear Regulatory Commission.

In section 4, we discuss the question of how the results of the Reactor Safety Study, and others, might be used. Highly relevant to that is the work that is presently being performed to refine and extend WASH-1400. This is a subject of the next section.

3.3 Studies being Performed or Planned

Safety assessments

Considerable resources are required to conduct a substantial, independent assessment of reactor safety. Not surprisingly, most recent efforts in this area have been largely devoted to analysis, criticism, or extension of the work of the NRC's Reactor Safety Study. In fact, many of the most important studies related to WASH-1400 have been the work of substantial groups; these include the EPA, EPRI, and the APS (although WASH-1400 was not the main subject of the APS study). As indicated above, EPRI is continuing such work and, no doubt, the EPA has a continuing interest in scrutiny of the WASH-1400 work. Moreover, individual members of the APS group are remaining active in independent analysis of topics relating to reactor safety and the assessment thereof. In addition to these groups and individuals, numerous others are active, at one level or another, in similar areas. However, the group that is extending the WASH-1400 work most substantially is the group at the NRC which is continuing applications of the methodology, the Probabilistic Analysis Branch of the Office of Nuclear Regulatory Research.

The NRC is, first of all, attempting to adapt the WASH-1400 methodology to use in licensing. The study itself only analyzed in detail two specific reactors, which are older than those now being built, and somewhat different in design. The NRC is examining specific features which may distinguish other reactors from those on which the study was based. A principal feature to be considered at the present time is the containment. The NRC staff is interested in examining differences between various containment concepts, sometimes even as offered by a single manufacturer. A specific containment feature which may distinguish between results for different PWR manufacturers is presence of an ice condenser (versus an alternative) used in the containment designs. On the other hand, the BWR examined in WASH-1400 used a Mark I containment, whereas reactors now being designed by the same manufacturer use Mark III.

In addition to analysis of reactor-specific features, for use in licensing of power plants, the NRC is considering application of this methodology to other elements of the nuclear fuel cycle, such as reprocessing plants and waste management facilities.

Further work is being done to improve the methodology itself. In the area of probabilistic accident analysis, the NRC hopes to identify which parameters are the driving forces for uncertainties in calculated results, thereby leading to work in those areas which would be most profitable in reducing the uncertainties. The group is examining the details of the models for core meltdown and fission product transport (within the containment) to see how they affect the radioactive release fractions. The consequences model is also being scrutinized, primarily to remove conservatisms that were accepted in the interest of timely results. One example is improvement of the precipitation model.

Thus the main areas of work at the NRC are to improve the details of the methodology and to extend its range of application to facilities other than the specific light-water reactors examined in WASH-1400.

Studies of safety design

This report has emphasized the form and adequacy of analytical techniques for predicting the probability and consequences of reactor accidents. Of the studies discussed in section 2, only the American Physical Society study group on light-water reactor safety devoted a significant portion of its effort to the basic question of reactor design and related analysis. The APS report reviews a number of important areas for reactor safety, including pressure vessel integrity, emergency core cooling system design, containment response, quality assurance, and computer modeling of LOCA phenomena. It also provides a view of the reactor safety research program which has been pursued in recent years. Although this view was based on information available in late 1974, the situation has not changed drastically, except that the responsibility for licensing (and related research on) light-water reactor power plants now resides with the Nuclear Regulatory Commission, rather than the Atomic Energy Commission.

A more recent perspective on the present status of reactor safety can be had indirectly through the eyes of the Advisory Committee on Reactor Safeguards (ACRS), the committee which advises the Nuclear Regulatory Commission in regulatory matters. Although the ACRS regards the current safety design of light-water reactors adequate to warrant their licensing for operation, it has established the practice in recent years of maintaining a list of "generic items" relating to light-water reactors. These are items which indicate specific areas of uncertainty related to light-water reactors. They do not necessarily imply that LWR design is deficient in these areas, but rather that an area has been identified as being unsatisfactory in some respect. Often it is the data base or analytical technique that is unsatisfactory, so that there is not sufficient information on which to base a judgment. Resolution of an item usually involves an improvement in the data base of the available analytical tools (or in the manner in which standards are formulated) and may or may not involve an alteration in reactor design of operation.

The ACRS began reporting such a list of generic items in 1972 and has updated the list on a roughly yearly basis. Of the approximately 70 items which had been placed on the list by the time of the fourth report (April 16, 1976), about half had been resolved by that time. Because these items indicate areas of uncertainty in LWR safety, we list them in Table 3-1. We have categorized them by broad safety-related areas. As might be expected, these broad areas themselves constitute a list of the important areas of concern in reactor safety, from the point of view of both the partisans and critics of nuclear power.

For each of the areas displayed in Table 3-1, the resolved items are listed first and followed by items which in April 1976 are outstanding. The fact that such items are brought up for consideration and gradually resolved is not surprising, considering how complex, important, and highly-regulated the safety aspects of nuclear power plants are. Consider, for example, the items listed under "ECCS and LOCA related items, including containment response." Both the resolved and outstanding items include specific areas of emergency core cooling design, containment design, and component behavior in a post-accident environment, all of which are fundamental areas of safety design. Considered as a whole, the items listed in the table may be regarded

Table 3-1. Rough Categorization of ARCS Generic Items Relating to Light Water Reactors

ECCS AND LOCA RELATED ITEMS, INCLUDING CONTAINMENT RESPONSE

I-1 Net Positive Suction Head for ECCS Pumps
I-3 Hydrogen Control After a Loss-of-Coolant Accident (LOCA)
I-20 Capability of Biological Shield Withstanding Double-Ended Pipe Break at Safe Ends
IA-5 ECCS Capability of Current and Older Plants
IB-3 Performance of Critical Components (pumps, cables, etc.) in post-LOCA Environment
IB-4 Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containment

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*II-2 Effective Operation of Containment Sprays in a LOCA
II-8 BWR Recirculation Pump Overspeed During LOCA
II-10 Emergency Core Cooling System Capability for Future Plants
*IIA-1 Pressure in Containment Following LOCA
IIA-3 Ice Condenser Containments
IIA-5 PWR Pump Overspeed During a LOCA
*IIB-3 Behavior BWR Mark III Containments
IIC-1 Locking Out of ECCS Power Operated Valves
IIC-5 Vessel Support Structures
IIC-8 Behavior of BWR Mark I Containments

QUALITY ASSURANCE, INSPECTION, TEST, AND MONITORING

I-9 Vibration Monitoring of Reactor Internals and Primary System
I-11 Quality Assurance During Design, Construction and Operation
I-12 Inspection of BWR Steam Lines Beyond Isolation Valves
I-15 Pressure Vessel Surveillance of Fluence and Shift
I-18 Criteria for Preoperational Testing
I-23 Quality Group Classifications for Pressure Retaining Components
I-25 Instrumentation to Detect Stresses in Containment Walls
IA-2 Primary System Detection and Location of Leaks
IB-2 Fixed Incore Detectors on High Power PWRs

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*II-4 Instruments to Detect Fuel Failures
II-5 Monitoring for Excessive Vibration or Loose Parts Inside the Pressure Vessel
II-11 Instrumentation to Follow the Course of an Accident
IIA-8 ACRS/NRC Periodic 10-Year Review of all Power Reactors
IIC-7 Maintenance and Inspection of Plants

EMERGENCY CONTROL

I-2 Emergency Power
IA-4 Anticipated Transients Without Scram
IB-5 Emergency Power for Two or More Reactors at the Same Site
IB-7 Control Rod Ejection Accident

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* IIA-2 Control Rod Drop Accident (BWRs)

PROTECTION AGAINST SABOTAGE

I-8 Protection Against Industrial Sabotage

IIC-3 Design Features to Control Sabotage

GENERAL EQUIPMENT AND SYSTEM ADEQUACY AND PROTECTION

I-6 Fuel Storage Pool Design Bases
I-7 Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles
I-13 Independent Check of Primary System Stress Analysis
I-14 Operational Stability of Jet Pumps
I-19 Diesel Fuel Capacity
I-24 Ultimate Heat Sink
IA-1 Use of Furnace Sensitized Stainless Steel
IA-2 Primary System Detection and Location of Leaks
IC-1 Main Steam Isolation Valve Leakage of BWR's
IC-2 Fuel Densification

II-1 Turbine Missiles
*II-6 Common Mode Failures
II-7 Behavior of Reactor Fuel Under Abnormal Conditions
*IIA-4 Rupture of High Pressure Lines Outside Containment
*IIA-6 Isolation of Low Pressure From High Pressure Systems
IIA-7 Steam Generator Tube Leakage
IIB-2 Qualification of New Fuel Geometries
IIB-4 Stress Corrosion Cracking in BWR Piping
IIC-2 Fire Protection
IIC-6 Water Hammer

SEISMIC RESPONSE

I-5 Strong Motion Seismic Instrumentation
I-22 Seismic Design of Steam Lines
IC-4 Seismic Category I Requirements for Auxiliary Systems

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*II-9 The Advisability of Seismic Scam

REACTOR PRESSURE VESSEL

I-10 Inservice Inspection of Reactor Coolant Pressure Boundary
I-16 Nil Ductility Properties of Pressure Vessel Materials

II-3 Possible Failure of Pressure Vessel Post-LOCA By Thermal Shock

GENERAL REACTOR OPERATION: CONTROL AND INSTRUMENTATION

I-4 Instrument Lines Penetrating Containment
I-17 Operation of Reactor With Less Than All Loops in Service
I-21 Operating One Plant While Other(s) is/are Under Construction
IB-1 Positive Moderator Coefficient
IC-3 Rod Sequence Control Systems

IIB-1 Hybrid Reactor Protection System

EFFLUENTS AND DECONTAMINATION

IB-6 Effluents from Light-Water-Cooled-Nuclear Power Reactors

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IIC-4 Decomination and Decommissioning of Reactors

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[a]Class I items are "resolved"; class II are not. A, B, and C indicates, respectively, items that were added in the second, third, and fourth ACRS reports.

*Items considered resolved by the NRC staff but pending by the ACRS either as a guide to areas of concern in reactor safety or as a glimpse of the manner in which uncertainties in reactor design are identified and resolved. It is therefore not surprising that these categories include the primary concerns expressed by organizations such as the Sierra Club and the Union of Concerned Scientists and by various individuals in recent hearings before the Joint Committee on Atomic Energy[14] and before the Subcommittee on Energy and the Environment of the House Committee on Interior and Insular Affairs.[13] Work in all of the areas listed by the ACRS and indicated in Table 2-3 is continuing, largely as part of the NRC program on reactor safety research.

3.4 Foreign studies

Only a small effort was devoted to examination of foreign efforts related to light-water reactor safety. In many respects, such efforts in this area follow the lead of work in the United States, as is to be expected considering that this country led in the development of these reactors. However, some important considerations are highlighted in foreign work, and we briefly summarize the information which is publicly available on European work. However, it should be noted that the foreign work is not conducted in as open a manner as in this country, so that the public information can only be regarded as representative of foreign work.

Substantial efforts in probabilistic analysis have been taking place in recent years in Europe. The earliest such study often referred to is the Swedish Urban Siting Study,[15] which analyzed the potential impacts of siting dual purpose power plants in urban areas for power generation and district heating. However this study adopted a probabilistic approach only to the consequences modeling and not to the matter of accident probabilities. The authors were of the opinion that the most important initiator of core-meltdown accidents was catastrophic reactor vessel failure, with a probability of occurrence of 1 in 1 to 10 million reactor years. This core meltdown probability is 50 to 500 times smaller than that calculated in WASH-1400[1]. Because the authors adopted, rather than calculated, a meltdown probability, the results of this study are actually more comparable to the WASH-740 study (of consequences of postulated accidents) than to the WASH-1400 mechanistic risk assessment.

In some respects, European work has extended or supplanted the techniques of WASH-1400, though -- even in the European community -- WASH-1400 is regarded as the archetype and most complete example of such studies. Some of these extensions take the form of different, perhaps, more sophisticated, probabilistic analysis techniques, much as work in this country (such as that pursued by EPRI) constitutes such improvements. Another major area where European work takes place is to apply these techniques to site-specific risk analysis.* The possibility of extending WASH-1400 for use in examining specific sites was mentioned above and is discussed further in section 4. In any event, it is worth noting that the practitioners of probabilistic analysis in Europe do not appear to obtain results which differ greatly from those of WASH-1400, except in specific respects which result from differences in reactor design or in population distributions.

In LWR reactor safety design, European work depends heavily on that performed in the United States. However, with respect to the safety of PWRs, an extremely interesting report[17] was recently made to the United Kingdom Atomic Energy Authority by a study group on pressure vessel integrity chaired by W. Marshall. Concern over the probability of pressure vessel failure, voiced most prominently by Sir Alan Cottrell had been one of the reasons for the British decision in 1974 to emphasize other types of reactors and had resulted in initiation of the Marshall study. (This study is now one of the main inputs to a generic review of PWR safety now being performed by the British Nuclear Installations Inspectorate and due to be completed shortly.) The Marshall group was satisfied that PWR vessel integrity could be satisfactorily assured provided NRC regulations were fully implemented and supplemented by a number of other specifications; Cottrell himself appears satisfied with the Marshall Report, but points out the importance of three of these specifications, having to do with: 1) limiting operational transients, 2) injection of ECC water at high temperatures, and 3) rigorous inservice inspection. In this country, the NRC/ACRS appear satisfied with 1 and 3, but 2 is among the items in Table 3-1.

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*A representative sample of publicly available information on European efforts is given in the report of the latest general meeting of the American Nuclear Society.[16]

In general, the European community appears more sanguine about the risks from nuclear power than does the United States community. Often regulatory requirements relating to routine emissions or to reactor safety are not as severe as in the United States. Although there are European critics of nuclear power, the public as a whole more readily accepts the potential hazards associated with its use and often regards these hazards as smaller than the risks from other technologies. A sentence from the recent British report on "Nuclear Power and the Environment"[18] (the "Flowers" report) could easily have come from WASH-1400:

> "The risk of serious accident in any single reactor is extremely small; the hazards posed by reactor accidents are not unique in scale nor of such a kind as to suggest that nuclear power should be abandoned for this reason alone."

References for Section 3

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13. This prepared testimony, as well as remarks at the hearings and auxiliary material, has been prepared by the Government Printing Office as Serial No. 94-61. The affiliations of those testifying are: Norman C. Rasmussen, Massachusetts Institute of Technology; Saul Levine, Nuclear Regulatory Commission; Niel Wald, M.D., University of Pittsburg; Marvin Goldman, University of California, Davis; R.C. Erdmann, Science Applications, Inc.; Frank von Hippel, Princeton University; W.K.H. Panofsky, Stanford University; William Rowe, Environmental Protection Agency; Henry W. Kendall, Massachusetts Institute of Technology.

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4. IMPLICATIONS OF SUCH STUDIES FOR REACTOR SAFETY

We have indicated in our previous discussion the reasons for emphasizing WASH-1400, the EPRI work, and the APS study. The first (extended by the second) constitutes a major methodological improvement in risk assessment, presuming the methodology can be accurately applied. The APS work, on the other hand, examines in a useful way the physical basis of reactor safety and the framework in which safety research has been proceeding.

The situation to be preferred would include an iterative interaction between the two points of view. Risk assessment requires information on design and design adequacy and can identify those areas where improvements are most needed. Safety system analysis and research supplies the information required in risk assessment and, in turn, may benefit from the results of such assessment. To complete the interactive process, a consideration of costs versus benefits is needed so that decisions on possible changes in research, design, or licensing priorities may be made in the light of benefits which may accrue from such alterations. Drawing primarily on the material presented in the previous sections, we summarize the major factors pertinent to safety assurance, safety assessment, and interpretation and application of safety studies.

4.1 Safety assurance

The most fundamental aspect of safety assurance is the adequacy of the reactor and associated systems. Such adequacy depends on design adequacy and quality assurance. The APS study devoted most of its attention to these questions, particularly design adequacy, even to the point of discussing specific portions of the reactor safety research program in some detail. Since the APS examination of the research program in 1974-1975, two major developments relevant to their discussion have occurred. First, the LOFT series of tests has begun, although the testing has so far been restricted to use of a non-nuclear core. Nuclear tests are not scheduled until 1977. According to the NRC, the non-nuclear tests are proceeding successfully. It is not clear to what extent this means that the test results are successfully being used to verify calculations of "realistic", rather than "conservative" codes. Secondly, one of the most important "separate effects" tests, the Plenum Fill Experiment scheduled to be performed at Battelle's Pacific Northwest Laboratory, has been cancelled, largely due to rapidly mushrooming costs and to delays in the tests. This test was designed to check the manner in which emergency cooling water injected during "blowdown" succeeds in filling the vessel below the core, rather than bypassing it and escaping out the break from which the ordinary coolant escapes. The NRC was, until recently, negotiating construction of a larger "ECC bypass" facility to be operated by ERDA. Although close to PWR scale, this would have been strictly a separate effects, not a system, test. (No nuclear core would be included.) It was recently judged unnecessary on the basis of new data. Overall, it appears that the specific APS recommendations for increased efforts in safety research have not been implemented. A factor that may have lessened the APS study's impact was the essentially concurrent release of draft WASH-1400 and the prevalent impression that the APS study was primarily a review of WASH-1400, which it was not.

Adequacy of reactor systems also depends on quality assurance in the design, construction, and operation of each reactor. Failures in quality assurance can directly degrade the performance of individual systems and, perhaps even more seriously, lead to system dependencies which directly undermine the basic safety philosophy of redundant and independent engineered safety features. As a result, a major effort of the utility constructing a plant and of agencies responsible for its inspection must be assurance of specified levels of quality in components and procedures in design construction, and operation. As indicated in the APS study, efforts for checking the effectiveness of quality assurance procedures are an essential ingredient in understanding the extent to which these procedures are fulfilling their role in reactor safety.

The single incident that has aroused the greatest interest in quality assurance is a fire which occurred in 1975 at Tennessee Valley Authority's Brown's Ferry site. This fire, started by a workman in a cable spreading room, destroyed a significant portion of the control cables for an operating reactor. This caused failure of coolant pumps as well as emergency systems which could have replenished the primary coolant that was being lost, over a period of hours, due to boiling induced by decay heat. In the end, a control rod pump was used to keep the water level in the core sufficiently high to prevent any damage to the fuel. Serious questions about the quality assurance program were raised by the fact that multiple failures could arise from a single human error; apparent violation of many safety procedures and poor layout of the important control cables permitted such an incident. The subsequent investigation had one tangible result: a new standard on fire protection has been proposed that is much more stringent than the one that probably would have been developed otherwise. However, aside from the development of the standard, quality assurance assumes some level of adherence to good work practices and to the standards that are specified, and it is not clear that such adherence occurred at Brown's Ferry.

An issue which lies very much at the interface between design adequacy and quality assurance is whether any degradation in plant performance or safety should be expected as a result of plant aging. The design of a plant takes into account the aging of systems and components, and inspection procedures are designed to circumvent difficulties arising from this aging. Moreover, plant operators often express the opinion that a plant becomes more dependable as it is "run in". Such an increase in dependability is certainly to be expected soon after start-up, simply because the "bugs" are worked out. However, the question remains whether the dependability curve turns over as the plant reaches middle age, so that a plant experiences decreased availability during the last half of life. Decreased availability may imply, in turn, greater amounts of test and maintenance during reactor operation, and certainly means more frequent shutdown of the reactor. Both of these factors can lead of themselves to a higher probability of accidents. A balancing factor, overall, is that the increasing total experience with design and operation of LWRs would tend to increase their overall safety, particularly in newly designed plants, provided vigorous efforts are made to maintain design adequacy and quality assurance standards.

Safety assurance depends not only on features of the nuclear power plant itself, but on its relationship to its surroundings. Strictly mechanical features of this relationship are the potential for earthquakes, floods, and tornadoes. Such potential must be considered directly in the plant design. Less mechanistic considerations include the distribution of populations around a nuclear power plant and the meteorological conditions which will be involved in any dispersal of radioactive materials following an accident. We will consider these factors below in connection with the manner in which consequences of nuclear accidents are calculated. Finally, an intrinsically difficult factor to evaluate is the potential for sabotage, a factor that was not incorporated into the risk assessment of WASH-1400, but which was considered briefly there and in the APS report.

4.2 Safety assessment

The methodology of WASH-1400 provides a framework in which the safety of nuclear plants may be assessed. The portion of that framework that constituted a new development during the course of the study is the accident sequence identification and quantification portion, and it is this portion that depends most strongly for its accuracy on a consideration of safety assurance, based on design adequacy and quality assurance. The basic input to the probabilistic methodology, in addition to information on the logical structure of the plant design, is the data on human or component failure or on test and maintenance. Somehow failures in design or quality assurance must be included in this information. Such failures may lead, not only to failures of specific components and systems, but also to common mode failures.

WASH-1400 attempts to include an examination of design adequacy in its consideration of the possibilities for failure. Observers have questioned the adequacy of this examination. From the point of view of the safety research program, one of the most interesting results of the study was that risk was largely independent of "functionability"* of the ECCS during large LOCAs. More specifically, sensitivity studies showed that emergency cooling functionability failures during as much as 10% of large LOCAs would not affect the study results "significantly". A closer reading indicates the 10% lack of functionability assumption would increase the risk by 10 to 30%, depending on the magnitude of consequences considered. Since many doubt the adequacy of the ECCS to handle large LOCAs, this question is of some interest. It would also be interesting to know, in general, the relevance of the recently discovered error in BWR torus-design calculations to such questions.

The final version of WASH-1400 also dealt explicitly with the relevance of the Brown's Ferry fire to the dependability of their results. Many critics of nuclear power tend to refer to Brown's Ferry as a "near meltdown accident", despite the great pains taken by the Nuclear Regulatory Commission to eliminate this impression. Regardless of whether such an impression is correct, the question was asked of the Reactor Safety Study whether their methodology had included the possibility that a fire could remove so many systems from operability.

*For lack of an English word, "functionability" is used to signify the degree to which a particular system can perform the intended task, assuming it operates. Functionability is thus a measure of design adequacy.

WASH-1400 indicates that incidents of this type would only contribute 20% of the overall risk from nuclear power, leaving the impression that their prior treatment had not considered such an accident sequence.

We should emphasize that a 20% alteration of the study results is well within the error bounds quoted on either probabilities or consequences. Moreover, WASH-1400 is very emphatic in discounting any intention of identifying every accident sequence or every possible common mode failure. In risk assessment, it is adequate to anlyze carefully only those that contribute significantly to the overall risk. This leads naturally to the questions often asked by critics, i.e., what is significant and, regardless of the precise answer to that, have all the significant contributors indeed been identified? It is particularly difficult to give firm answers to this last question if significant contributors to risk turn out to be failures in quality assurance, leading to unanticipated common mode failures.

Presuming an adequate understanding of design and quality assurance contributions to failure, the basic input to the quantification of the fault trees is actual failure data. Due to the lack of historical data on many specifically nuclear components, a major portion of the data base is abstracted from experience in other industries. The actual input to the fault trees typically has large uncertainties, factors of 3 to 10. When propagated systematically through the trees, these still yield results with small enough uncertainties to be useful. (Such uncertainties are stated in captions of the various figures of section 2.1.) It is, nevertheless, important that the failure rate data be dependable. This, in turn, depends on an accurate understanding of possible failure due to inadequate design or quality assurance. Furthermore, the failure rate uncertainties were propogated through the fault trees assuming that the uncertainties were entirely random. This becomes an important assumption if multiple failures contribute substantially to the risk. If the central values of the failure rate data have any systematic error in them, this fact would then change the accuracy of the overall risk assessment, possibly even outside of the quoted uncertainties, since the influence of multiple failures depends on a product of the failure rates; if each of these has a systematic error, the product has a greater error. WASH-1400 suggests that most of the risk, however, is due to single component failures.

The fundamental requirements for accurate probabilistic analysis would, then, appear to be: a good analyst, an accurate representation of the plant design for the analyst to use, adequate information on design adequacy and on the effectiveness of quality assurance, and correct information on failure rates. It is not even necessary that the analysis be complete, only that the important contributors to accidents be included. One might surmise, from the range of opinion expressed on the probabilistic methodology and on the basis of reactor safety in general, that the largest uncertainty in these requirements is the quality assurance program, both because it may fail to assure the quality of components, and because it may fail to prevent fundamental design and construction errors.

Given an accurate representation of the probabilities of releases, accurate risk assessment next requires adequate treatment of the dispersal of radioactivity and its effect on humans and other components of the environment. The APS study group and others helped the Reactor Safety Study to improve its calculation of dispersal and exposure. The EPA continues in its recommendation of alteration of the dose-response relationship used in the consequence calculation. This insistence arises from the EPA's adoption of a linear, non-threshold dose-response, without correction for low dose or dose rate. The Reactor Safety Study, on the other hand, feels that a realistic, as opposed to conservative, assessment of the risk must make such corrections and emphasizes the concurrence of the National Council on Radiation Protection and Measurements with this point of view.

There is also a continuing controversy over the adequacy of the assumptions on evacuation made by WASH-1400, a controversy that is fueled by the fact that often the evacuation plans in the vicinity of presently operating nuclear plants do not appear to meet the specifications suggested by the NRC. However, it is not clear how much effect the availability of an evacuation plan has on the speed of evacuation, or, in turn, how much effect a rapid evacuation actually has on the consequences of a radioactive release. On the first, the data base is not adequate; on the second, it is difficult to extract from WASH-1400 the importance of evacuation in the reduction of consequences.

In summary, then, there are many questions on the detailed application of the WASH-1400 methodology and, more fundamentally, on its ability to identify the accidents which contribute substantially to the risk. Moreover, the manner in which the results are presented does not leave the study open to easy interpretation or application, as discussed below. These difficulties aside, the success of any such risk assessment immediately leads back to its starting point, i.e., to safety assurance. Presuming dependability of the assessment, the identification of important contributors to risk can be used in determining priorities in reactor safety research or in the design of nuclear power plants. This is a step beyond risk assessment.

4.3 Interpretation and application of risk assessments: WASH-1400 problems and possibilities

Interpretation of any results from a study intended to assess risk must necessarily take into account the considerations of the last section. The manner of application of the assessment methodology, including the assumptions used and the data base employed, obviously affects the validity of the results and the manner in which they should be interpreted or used.

A less fundamental consideration, but still an important one, is the manner in which the study and its results are presented. In the case of WASH-1400, for example, it is clear that a reading of the executive summary alone would never suggest that latent fatalities completely dominate the total number of predicted fatalities from large reactor accidents. In fact, there is no direct statement or indication in the summary that any accident would ever cause more than one cancer fatality. Regardless of the intent of this omission, policy decisions or other actions taken considering such information would be poorly based. Granted, the brief executive summary is not intended to be comprehensive. But it should also not mislead. A less severe omission, perhaps, occurred in the summary of the main report, in the use of the total of 2000 years of power reactor experience, without a serious accident, to show that the study's result of one core meltdown per 20,000 reactor years is not unreasonable. That is, this is not a smaller probability than the rough maximum of one per thousand reactor years that one can get on the basis of past experience (i.e., the 2000 reactor years). The study omits any emphasis of the differences between commercial and military power reactors, differences that might invalidate the use of military data. More important, perhaps, the study gives no indication of the precision of the data from the military program; is it percise enough to justify a statement that no accidents resulted in elevated fuel temperatures?

A much more substantive difficulty with the presentation of the results in WASH-1400 is that it is very difficult, if not impossible, without completely re-doing the calculations, to see how the results develop through the various steps of the calculation. This obscurity comes close to overwhelming the reader in the discussion of the consequences calculation. As a result, it is very difficult to effectively criticize the results, a process that is extremely important, as discussed below. Moreover, it makes application of the results in making various kinds of decisions equally difficult, as we shall see. It would be quite valuable for the calculations of WASH-1400 to be presented in enough detail, including intermediate results, that investigators outside the study group itself could effectively use and reproduce the results.

An example of some interest is the difficulty in following through the calculation to see how much of the overall risk is caused by each release category, and hence by specific accident sequences in these categories. The discussion in WASH-1400 leaves one with the impression that the accidents with small consequences are the important ones, although all that it states directly is that they are the more probable ones. The distinction is important, because risk does not depend alone on the probability of accidents. It depends on a product of probability and consequences, as is made clear in the study's discussion of the meaning of risk. As for the release categories that are specified in the study, since these serve in some sense as the source term for radioactivity, one would think that one quantity of some interest is not the probability of the individual categories, but some indicator of the total released radioactivity from these categories, i.e., some product of the probability for the category times the amount released. Granted, this is not easy to define precisely, since the release is composed of many types of radioactivity. However, even a crudely defined indicator would be helpful for judging the relative importance of possible accident sequences at nuclear power plants. Many of the important radionuclides maintain a roughly constant ratio from one release category to another, so that if we roughly multiply the category probability by the fraction of iodine released, the result should be useful. We display the results of this exercise in table 4-1, where it is easily seen that the accidents known as PWR 2, BWR 2, and BWR 3 would be expected to be the major contributors to risk, on the simple grounds that they contribute most of the source. The details would be modified by the fact that the radioactivity is given off differently in the various release categories; the number of early fatalities would be particularly sensitive to these details. Furthermore other inputs to the consequences calculation may affect the relative importance of the release categories. However, it is clear that the higher probability core meltdown release categories, such as PWR 7 and BWR 3 are not necessarily the dominant

TABLE 4-1 A RELEASE MAGNITUDE INDICATOR*

	Probability per reactor year	Fraction of Iodine Released	Indicator* (Probability x Fraction)	
PWR 1	9×10^{-7}	0.7	6×10^{-7}	
PWR 2	8×10^{-6}	0.7	6×10^{-6}	←
PWR 3	4×10^{-6}	0.2	8×10^{-7}	
PWR 4	5×10^{-7}	0.09	4×10^{-8}	
PWR 5	7×10^{-7}	0.03	2×10^{-8}	
PWR 6	6×10^{-6}	8×10^{-4}	5×10^{-10}	
PWR 7	4×10^{-5}	2×10^{-5}	8×10^{-10}	
PWR 8	4×10^{-5}	1×10^{-4}	4×10^{-9}	
PWR 9	4×10^{-4}	1×10^{-7}	4×10^{-11}	
BWR 1	1×10^{-6}	0.4	4×10^{-7}	
BWR 2	6×10^{-6}	0.9	5×10^{-6}	←
BWR 3	2×10^{-5}	0.1	2×10^{-6}	←
BWR 4	2×10^{-6}	8×10^{-4}	2×10^{-10}	
BWR 5	1×10^{-4}	6×10^{-11}	6×10^{-15}	

← points out dominant contribution according to this indicator.

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*This "indicator" is intended to be an approximate measure of the risk (i.e., probability × consequences) posed by each release category of WASH-1400.

contributors to risk. The EPRI work succeeded in dividing the consequences among the different categories (see section 2.2) by actually repeating the calculations of WASH-1400. The ability to identify the relative importance of the various release categories leads to identification of the dominant accident sequences, the first step in improvement of reactor safety.

An alternative question which may be asked is, ignoring the question of which accident sequences contribute most of the risk, how important are small consequence versus large consequence accidents? This information is not apparent in WASH-1400. All of the tables and graphs indicate clearly that the large consequence accidents are improbable as compared with the smaller ones, which is undoubtedly correct. This, again, is not a risk indicator. However, it is possible to break down the graphical information and associate probabilities with small consequences intervals (or vice versa) after which it is trivial to identify where the important risk arises. For example, figures 2-4 and 2-6 display the probability of early and latent death, respectively. Breaking this information down roughly, we have extracted, as an example, the following information: the overall risk for early deaths is 4×10^{-3} per year and for latent deaths is 9×10^{-2} per year per year. (The strange unit "per year per year" is due to WASH-1400's specification of the yearly incidence of latent fatalities after a specific accident, rather than the total due to the accident.*) These results are to be compared with the WASH-1400 results of 3×10^{-3} and 7×10^{-2}, respectively, from table 2-9, giving some confidence that our crude extraction of this result is not misleading.

As to where most of the fatalities arise, we find that half of the early fatalities occur from accidents which cause a minimum of 400 early deaths. It can be determined that the class of accidents causing at least 400 deaths has a probability of 3×10^{-6} per year (for 100 reactors) and appear to cause a minimum of 650 cancers per year or 20,000* cumulative. (On the other hand, it is not this class of accidents which causes the bulk of cancer deaths; the risk from cancer is concentrated in the higher probability region, as we shall indicate.) 85% of early deaths come from accidents which cause a minimum of 100* early deaths.

*To minimize the confusion caused by presentation of the rate at which latent effects occurs rather than the cumulative number, we have also given the latter number (which gives the actual commitment of deaths from the accidents considered) which may then be compared directly with the early fatalities.

On the other hand, half of the cancer deaths occur from accidents which cause a minimum of 80 cancer deaths per year (or 2400* cumulative). This corresponds to accidents with a probability of 3×10^{-4} per year (for 100 reactors). From a rough comparison of the data, it appears that the 2400* cancer death accident itself (as distinguished from those which exceed this number) typically causes no early fatalities at all. It is interesting that a release that would cause no early deaths would cause thousands of lethal cancers, based on the WASH-1400 dose response relationship. Finally, 95% of the risk from cancer deaths arises from accidents which cause a minimum of 10 deaths per year (300* cumulative).

This simple analysis is revealing (if not confusing) and gives results which are contrary to the impression which WASH-1400 conveys, that it is the small accidents that contribute most of the risk. The study, of course, does not directly make such a statement, but in its consideration of the factors which are important to the risk, it speaks primarily of the higher probability of the small accidents, leading the reader to think that these are the accidents which are most important.

A related question is whether there are any features of the WASH-1400 methodology which would inherently affect the shape of the curve of probability versus consequences. Such features might not affect the accuracy of the overall average risk, but might still alter the balance between large and small consequence accidents. For example, WASH-1400 uses a set of population distributions, each constructed from the total information on all sites that are associated with one of the "typical" sites. There is, to some extent, an averaging process involved in this approach that could reduce the apparent significance of very large accidents. However, it is possible that the method used to construct the population distribution may avoid this difficulty. Another methodological approach which could influence the balance between consequence sizes is the treatment of common mode failures. A basic question is whether the occurrence of common mode failures unanticipated by the study could affect the shape of the curve; a change of a factor of 10 or 100 at the high end of the curve would drastically change the importance of large accidents for early fatalities. The relative importance of high versus low consequence accidents is highly relevant to siting and emergency planning.

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*See footnote, previous page.

We are thus led to a consideration of how the results of risk assessment studies, such as WASH-1400, may be applied. We can lump applications into several areas: first, of course, is simply to provide an assessment of risk that may be used in making overall policy decisions, either on the acceptability of nuclear power *per se* or on its acceptability as compared with the alternatives. Use of these studies for this, their most fundamental purpose, is not necessarily straightforward. For example, it is not clear how to weight the importance of very large consequence accidents. WASH-1400, despite its disclaimers, does attempt to force a judgment that the risk from nuclear power is low. As to the possibility of making comparisons with the alternatives to nuclear power, no equivalent studies have been performed for other technologies.

A second area of application has been discussed in the previous sections, i.e., to identify areas where design might suitably be changed. This is the main thrust, presumably, of the effort of the NRC to extend the WASH-1400 methodology to several other versions of the LWR. However, the point of view is to apply it to the initial design stage and, indeed, to judgments on the research program. There is a limitation to this approach that is quite visible in the case of the ECCS: WASH-1400 assumed functionability, then went on to show that its results were not extremely sensitive to the dependability of the ECCS system, within limits. Since a basic purpose of the safety research program is to verify functionability of the ECCS system for large LOCAs, a study that presumes such functionability is not an appropriate guide to alterations in the research program. In spite of this, results of the probabilistic treatment, properly applied, can be useful for design or research decisions. It is to be expected that the work being supported by EPRI would have this emphasis.

Finally, we return to the narrower question of how results from a study such as WASH-1400 may be used for siting decisions. We have indicated that the results are not broken down into enough detail to be used in certain applications. To be applied in siting decisions, the basic assumptions of the calculation must be known, i.e., assumptions about population distributions, local meteorology, and probable effectiveness of emergency planning. At the present time, the only way to determine the dependence of consequences on these assumptions is to repeat the calculations.* Moreover, for broader considerations, such as what kind of emergency planning should exist around nuclear facilities, the display of results obscures the relative importance of small and large consequence accidents, information that would be valuable for planning.

We should emphasize that the studies discussed in this report direct their attention only to the nuclear power plant. Hence they do not provide a basis for consideration of other aspects of the nuclear fuel cycle, such as reprocessing or waste management. Nor do they treat associated questions, such as the possibilities for sabotage of nuclear plants or the potential for diversion of nuclear materials to weapons production, possibilities which are difficult to treat in any analytical framework. These studies have concentrated on the potential risks from accidental releases of radioactivity from the central component of the nuclear fuel cycle, the power plant itself.

Because WASH-1400 presents a public framework for assessment of reactor-related risks, it has been the focus of public debate and of this discussion. Both the Nuclear Regulatory Commission and outside reviewers are making substantial efforts to criticize and improve this framework and its results, even though the point of view of the NRC and the outsiders has often been different. It is apparent, however, that the interaction has led to greater accuracy in the study's results.

Some improvements which might be made to increase the accuracy and utility of results from the WASH-1400 methodology have been indicated above. In a more substantial way, the APS study made numerous recommendations related to what we have called safety assurance, i.e., design adequacy, based on an expanded research effort, and quality assurance. The probabilistic methodology can ultimately be involved in the safety assurance process. In either of the general areas we have discussed, safety assurance and risk assessment, the interaction between the Nuclear Regulatory Commission and other elements of the citizenry

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*The necessity of performing one's own calculations requires simplifications which might be avoided otherwise. For example, the APS study group constructed a simple, but very revealing dispersion model. However, in the absence of a careful reading, one might conclude from their discussion that the total latent deaths from a given accident is strictly proportional to the total distance from the plant that is considered, a result that would be roughly true only in the absence of depletion of the plume.

is extremely important, as the recent past has shown. It remains to be seen whether the complementary approaches to safety, research and design versus risk assessment, and the complementary points of view, that of the Nuclear Regulatory Commission and other government entities versus some members of the general population, can lead to a satisfactory resolution of the issues important to nuclear reactor safety.