NUREG-0438

PLAN FOR RESEARCH TO IMPROVE THE SAFETY OF LIGHT-WATER NUCLEAR POWER PLANTS

A Report to the Congress
of the United States of America

April 12, 1978

Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission

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TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

SUMMARY

This is the U.S. Nuclear Regulatory Commission's first annual report to Congress on recommendations for research on improving the safety of light-water nuclear power plants. In the preparation of this report, a research review group was formed, composed of NRC staff members. Consultants to the group represented various points of view, including those of national laboratories, universities, public interest groups, NRC research contractors, industry, and the U.S. Department of Energy.

Suggestions for reactor safety research were identified in, or received from, various sources, including the Advisory Committee on Reactor Safeguards, the NRC regulatory staff, and the consultants to the Research Review Group. After an initial screening to eliminate those not related to improved reactor safety, all the suggestions were consolidated into research topics.

The research topics were examined against a set of judgmental criteria consisting of the breadth of technical support, the potential for reducing reactor accident risks, the generic applicability of potential improvements, and the estimated cost of implementation.

It is recommended that the following research projects be carried out:

A. Alternate containment concepts, especially vented containments

B. Alternate decay heat removal concepts, especially add-on bunkered systems

C. Alternate emergency core cooling concepts

D. Improved in-plant accident response

E. Advanced seismic designs

Because of the limited time available for the preparation of this report, only the research topics that appeared most likely to lead to significant risk reductions are recommended as research projects. The remaining research topics, along with others that may be identified, should be examined in scoping studies. Furthermore, it is recommended that a project be undertaken to improve the methodology of value/impact analyses for use in selecting future research projects and in aiding decisions on the implementation of potential safety improvements in nuclear power plants.

The recommended research projects, if approved, will produce safety-system design and performance requirements and value/impact analyses associated with their implementation in plants. Actual implementation will require the establishment of regulatory criteria or rules by NRC; development by industry of the details of design, manufacture, and installation; and an NRC review of industry proposals to ensure compliance with regulatory requirements.

No funds have so far been budgeted for these research projects. At the proposed level of effort, the implementation of this plan would require \$14.9 million over a 3-year period from the time work is started.

1.0 INTRODUCTION

This is the U.S. Nuclear Regulatory Commission's (NRC) first annual report to Congress on research related to improving the safety of light-water nuclear power plants. It is responsive to NRC's amended research charter in the Fiscal Year 1978 Budget Authorization Act[1] for the NRC. This Act amended the Energy Reorganization Act of 1974,[2] requiring the NRC to develop a long-range plan for the development of new or improved safety systems for nuclear power plants. As discussed in Section 1.2, this requirement represents a significant and useful change in NRC's charter for the conduct of its safety research program.

This report presents recommendations for research projects and studies directed toward improving the safety of light-water nuclear power plants. The recommendations have been reviewed by the Advisory Committee on Reactor Safeguards, which "concurs in these choices and believes that these studies should be undertaken even though their risk reduction potentials are not yet clearly known. These studies and the follow-on programs will serve to place in perspective the extent and suitability of possible safety improvements."[3] Because of the short time available for the preparation of this first report, the scope of its recommendations should be regarded as preliminary and will be expanded in the future to provide longer range coverage and greater completeness.

The proposed NRC research program, described in detail in Chapter 4, consists of five research projects to be started as soon as funds are available. These five projects were selected from a list of sixteen research topics, which represented a consolidation of more than 200 suggestions for research. Scoping studies to identify more clearly whether any of the remaining eleven research topics should be considered for research projects in a longer range program should also be conducted as part of establishing the long-range program requested by Congress. These studies could also be useful in providing technical bases for closing out further consideration of topics not initially selected. Finally, it is recommended that studies be performed to improve the methodology of value/impact analyses for use in assessing proposed research topics as well as to aid in evaluating the need for implementing the results to be achieved in this program.

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[1]Nuclear Regulatory Commission Authorization for Appropriations for Fiscal Year 1978, Public Law 95-209 (91 Stat 1481), December 13, 1977.

[2]Energy Reorganization Act of 1974, Public Law 93-438.

[3]Advisory Committee on Reactor Safeguards, "Proposed Research on Systems to Improve Safety" (Stephen Lawroski, Chairman, to Hon. Joseph M. Hendrie, Chairman, U.S. Nuclear Regulatory Commission), March 13, 1978. This ACRS report is reproduced in Appendix A.

At the proposed level of effort, the program would require funding of $14.9 million over a 3-year period from the time work is started. Additional funding is likely to be required in future years because the scoping studies recommended above may identify more projects that should be undertaken.

1.1 BACKGROUND

The NRC research program is part of the NRC regulatory program for ensuring the safety of nuclear power plants, which are owned and operated by electric utility organizations. The principal responsibility of the NRC is to ensure that public health, public safety, and the environment are adequately protected. The NRC performs this function by defining conditions for the use of nuclear power and ensuring by technical review and inspection that they are met. The NRC research program provides technical information, independently of the nuclear industry, to aid in discharging these regulatory responsibilities.

1.1.1 TECHNICAL APPROACH TO REACTOR SAFETY

The much discussed defense-in-depth approach is a major factor in ensuring the safety of nuclear power plants. The NRC staff uses this approach in its review of a broad spectrum of possible events in order to define safety design and operating requirements for plant components, systems, and structures. The range of potential occurrences in a nuclear power plant can be categorized into three major groups:

- Anticipated operational events, which may occur with moderate frequency (i.e., several times per year) and would not result in significant releases of radioactivity.

- Events with a low probability of occurrence (in the range of 1 chance in 10 to 1 chance in 100 per year) and the potential for small releases of radioactivity.

- Events with a very low probability of occurrence (in the range of 1 chance in 1,000 to 1 chance in 10,000 per year) for which engineered safety features are provided to mitigate potentially severe consequences.

Since events in the first group may occur with moderate frequency during plant operation, it is the NRC's objective to ensure that they do not involve radioactive releases above the stringent requirements governing normal operations. Analysis of these moderate-frequency events provides an opportunity for detecting and correcting weaknesses in specific plant designs that might otherwise contribute to more serious failures.

Analyses of events in the second group must show that, if they were to occur, the plant design would be such that NRC's regulations covering exposure of the public to radioactivity would not be exceeded. In this way, assurance is gained that these low-probability events would entail little or no risk to public health and safety.

To provide additional defense in depth, events in the third group are postulated in spite of their very low likelihood. These postulated events are evaluated using very conservative assumptions. "Conservative" in this sense means deliberately underestimating the performance of a safety feature and overestimating the consequences of its failure. An example is the assumption of degraded performance of safety systems concurrent with a major rupture of a pipe in the reactor coolant system (a loss-of-coolant accident). The plant, including engineered safety features, must be designed to ensure that the calculated consequences of these very low probability events are well within the NRC regulations. The engineered safety features include such systems as emergency core cooling systems, radioactivity-removal systems, and containment heat removal systems.

The spectrum of events considered in the NRC review does not include <u>all</u> events that could conceivably occur in a nuclear power plant. It is not necessary, nor even possible, to design nuclear power plants (or any other man-made system) for all conceivable eventualities. Thus, at the very low probability end of the spectrum of events there is a residuum of accident sequences that could, if they occurred, lead to radiological consequences in excess of NRC regulations. Such accidents may involve sequences of failures (i.e., accident sequences), each one of which is, in itself, relatively unlikely. In the early years of the regulatory program, the probabilities of accidents involving such sequences were considered in a highly judgmental way to assist in regulatory decision-making. More recently, additional information and improved methods for estimating failure probabilities have become available. The NRC staff continues to use judgment and available information in determining whether potentially severe events should be formally considered in the regulatory process (i.e., require special design features) or, alternatively, whether they are so unlikely as to be regarded an acceptable risk.

1.1.2 REACTOR SAFETY REGULATORY PROCESS

Before a nuclear power plant can be built at a particular site, a construction permit must be obtained from the NRC. A major part of a construction permit application is the preliminary safety analysis report (PSAR) This document describes the design of the proposed plant and presents comprehensive data on the proposed site. It evaluates a broad range of potential occurrences, including a set of design-basis accidents and the safety features provided to prevent them or, if they should occur, to mitigate their effects on both the public and the facility's employees.

The NRC staff reviews the design of the proposed plant and site to ascertain that adequate provisions to protect the health and safety of the public and the environment are included. Design methods and calculation procedures are examined to establish their validity. Checks of actual calculations and other procedures of design and analysis are made by the staff to establish the validity of the applicant's design.

When the NRC staff has completed the first phase of a safety review, the Advisory Committee on Reactor Safeguards (ACRS), an independent committee established by Congress to advise the NRC on nuclear safety, reviews in public sessions each application for a construction permit and reports in a public letter to the Chairman of the NRC on the acceptability of the plant.

The law requires that, before a construction permit is issued for a nuclear power plant, a public hearing be held to allow for full public participation in the NRC decision-making process. The public hearing is conducted by a three-member Atomic Safety and Licensing Board appointed from the NRC's Atomic Safety and Licensing Board Panel. Interested parties opposed to the plant have the right to participate in these hearings. The Board considers all the evidence presented in the hearing, together with proposed findings of fact and conclusions of law filed by the parties, and issues an initial decision.

The Board's initial decision is subject to review by an Atomic Safety and Licensing Appeal Board on its own motion or in response to exceptions filed by any party to the proceeding. The decision may also be reviewed by the Nuclear Regulatory Commissioners. The final NRC decision regarding a licensing action is subject to judicial review in the Federal courts.

When the construction of a nuclear facility has progressed to such a point that final design information and plans for operation are ready, the applicant submits a final safety analysis report (FSAR) in support of an application for an operating license. The FSAR furnishes pertinent details on the final design of the plant and supplies plans for operation, procedures for coping with emergencies, and security provisions for protection against sabotage. This information is also reviewed in detail by the NRC staff and then independently evaluated by the Advisory Committee on Reactor Safeguards in open sessions. The ACRS advice is provided to the Commission by public letter.

Throughout the construction and lifetime of the plant, periodic inspections are conducted to audit safety and compliance with license conditions.

1.2 SCOPE OF NRC RESEARCH PROGRAM

The NRC's original research charter was defined in Section 205 of the Energy Reorganization Act of 1974, which gave the NRC's Office of Nuclear Regulatory Research responsibility for (1) developing recommendations for research deemed necessary for the performance of NRC's licensing and related regulatory functions and (2) performing, or contracting for, research deemed necessary for the performance of NRC's licensing and related regulatory functions.

In discussing reactor safety research, it is necessary to consider three types of research, distinguished by the different goals embodied in each:

- Research for improved safety.
- Confirmatory research.
- Developmental research.

Research for improved safety is research on concepts, systems, and processes believed to have potential for improving the safety of commercial nuclear power plants. Its purpose is to investigate the feasibility, benefits, and costs of implementing these concepts. Research for improved safety can in principle be carried out by industrial and government organizations. This type of research is the subject of the present report.

Confirmatory research is research needed to provide a basis for evaluating applications for regulatory decisions, or to provide a basis for regulatory requirements or policy, or to provide NRC with the physical or judgmental capability to regulate the use of nuclear power and materials. It is carried out, independently of the nuclear industry, by NRC and by comparable organizations in other countries. This type of research comprises the current NRC research program and has made significant contributions toward improving the safety of nuclear power plants.

Developmental research is research conducted to evaluate the safety of materials, processes, and equipment likely to be proposed by an applicant for an NRC license. It includes research performed in the process of developing and designing a proposed facility, as well as any research needed to provide information in support of a safety assessment. This type of research is not performed by the NRC.

The Conference Report[1] associated with the Energy Reorganization Act of 1974 made it clear that the NRC was to engage only in confirmatory research. This requirement kept the NRC's research program in a principally reactive mode and left the NRC little initiative to conduct research in areas that could lead to the development of improved reactor safety systems.

The Fiscal Year 1978 Budget Authorization Act for the NRC modifies Section 205 of the Energy Reorganization Act to require that the NRC prepare a long-term plan for the development of new or improved safety systems for nuclear power plants. The NRC believes this change in its charter to be very useful; it will permit the exploration and evaluation of the many suggestions that have been made for improving the safety of nuclear power plants and may indeed lead to improvements in their safety.

In addition, the associated Conference Report[2] amplifies the Fiscal Year 1978 Budget Authorization Act as follows:

- The basic purpose of this research is the improvement of reactor safety and not the enhancement of the economic attractiveness of nuclear power versus alternative energy sources.

- The plan is to include brief descriptions of the projects which are proposed, the need for each project, a timetable for its implementation, the cost of the project, and other pertinent information.

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[1]Conference Report, Energy Reorganization Act of 1974, Report No. 93-1445, October 8, 1974.

[2]Conference Report, Nuclear Regulatory Commission Authorization for Appropriations for Fiscal Year 1978, Report No. 95-788, November 1, 1977.

- The Commission, in developing the plan, should coordinate with the Department of Energy or other agencies which are conducting similar efforts and any actions taken to implement such a plan should take into account related activities in progress at other agencies.

- This plan is to be updated annually and submitted to the Congress by February 1 of each year.

This new charter for NRC research on improved safety is the basis for the activities proposed in this report. Future reports will present a more comprehensive plan; they may also recommend some tasks that will be more appropriately funded and directed by organizations other than the NRC. The scoping studies planned as part of the research program described herein would include development of this more comprehensive planning. One type of task already foreseen as inappropriate for NRC is detailed system design; it is discussed later in this section.

Although the focus of NRC's ongoing program is on confirmatory research, it is evident that some of this research has the potential for improving reactor safety. Examples are improved techniques for the surveillance of reactor system integrity, alternate configurations for emergency core cooling systems, and improved fire protection methods; these will continue as part of the confirmatory research program. The new program of research for improved safety has been formulated to recognize such items and avoid duplication of effort.

Clearly, the principal concern with reactor safety is related to potential accident risks and, to some extent, to radioactive exposures of plant operating and maintenance personnel. The component of risk that arises from reactor accidents was treated in the recent Reactor Safety Study.[1] Accident sequences that are significant contributors to risk were found to involve:

1. An initiating event (e.g., a transient disturbance or a pipe break).

2. Failure of one or more safety functions (e.g., reactor shutdown or emergency core cooling).

3. Release and dispersion of radioactivity (e.g., failure of the reactor containment building).

Risk can therefore be reduced by diminishing (1) the probability of initiating events, (2) the probability of failure of safety systems provided to control the course of events, and (3) the probability of failure of safety systems and structures provided to inhibit releases of radioactivity. The proposed NRC research program considers all of these types of improvement.

To many people, the concept of risk reduction is closely associated with concern about how low the resultant risk should be--that is, "how safe is safe enough?"

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[1] Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400, U.S. Nuclear Regulatory Commission, Washington, D.C., October 1975.

The concern stems from the knowledge that zero risk in any activity is unattainable and that a competent basis is needed for defining the point beyond which further reductions in risk need not be made. While risk can be objectively defined as the product of the probability of a set of events and their consequences, it is also necessary, in decision-making on a broad policy level, to consider the subjective value judgment that society may place on risks arising from various activities. It is also necessary, in making such decisions, to weigh benefits against risks. Such decisions have generally been made in a qualitative way because many of the factors involved in benefit/risk assessments could not be well quantified.

While reasonably good techniques, such as those used in the Reactor Safety Study, are available to quantify risks, there appear to be many aspects of benefits that are exceedingly difficult to quantify. Furthermore, quantification of the value-judgment portion of the risk equation and the comparison of benefits and risks in commensurate terms appear to be equally difficult. This suggests that the development of techniques to quantify benefits versus risks would require a long-term effort.

The U.S. Environmental Protection Agency and the Natural Resources Defense Council have suggested that NRC should develop quantitative criteria for acceptable risks from potential nuclear power plant accidents. It was also the opinion of many of the consultants who advised the NRC in the preparation of this report that, since its charter has been expanded to cover research on improved reactor safety, the NRC should study the establishment of acceptable levels of risk. Such an effort has already commenced in NRC's confirmatory research program and will be continued there. However, the NRC sees no need to wait for completion of that effort before it undertakes the research work recommended in this report.

The Congressional requirement for the "development of new or improved safety systems" for nuclear power plants could cause some difficulty if the NRC were to develop the design of an improved safety system in sufficient detail to permit direct incorporation into commercial nuclear power plants. Such a course would place the NRC in the position of reviewing and approving, as part of its licensing process, designs that it had itself developed. This potential conflict of interest can be avoided if the NRC does not develop detailed system designs but rather obtains the basic data and develops analytical models for the analyses needed in its regulatory process. Such an approach would enable the NRC to determine the feasibility of achieving particular safety improvements, to evaluate the safety significance of proposed changes, and to specify regulatory requirements where implementation is determined to be desirable, without preparing detailed designs.[1] Problems requiring detailed design work could be defined in coordination with the U.S. Department of Energy (DOE); the latter could carry out the work, and the NRC could use the results to evaluate the improved-safety potential. Preliminary discussions have been held between the NRC and DOE staffs to initiate the necessary coordination foreseen in the Congressional mandate, and NRC conferred with DOE representatives during the preparation of this report. As the program plan matures, the appropriate coordination will be implemented.

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[1]The Advisory Committee on Reactor Safeguards shares the views of NRC on this matter (see Appendix A).

To improve the safety of nuclear power plants, the design, construction, or operation of safety-related structures, systems, or components must be modified in some specific way. Research is the first step in the process leading to such modifications; to achieve the improvement in safety, the research results must then be implemented. Implementation, some aspects of which are discussed in Section 1.4, is not included in the NRC research program described in this report.

1.3 PREPARATION OF THE RESEARCH PLAN

This plan for research was developed principally by the NRC Research Review Group (RRG) on Improved Reactor Safety.[1] The RRG was organized specifically to consider NRC's amended charter for conducting research on reactor safety improvement and consists of members representing NRC's Offices of Nuclear Regulatory Research, Nuclear Reactor Regulation, and Standar Development. Consultants to the RRG represent various points of view, includi… those of NRC research contractors, industry, national laboratories, universities, public interest groups, and DOE.

As discussed in greater detail in Chapter 2 (see also Appendix C), a systematic review was conducted of recommendations published in several significant critical reviews and reports on nuclear reactor safety. Because of its long and detailed involvement in reactor safety, special attention was given to the recommendations of the Advisory Committee on Reactor Safeguards, including its report[2] to Congress on NRC's safety research program. Comments from the NRC regulatory staff were also solicited and considered. In addition, suggestions and criticisms were obtained from the consultants to the RRG. The result was a list of improved-safety concepts recommended by one or more of these groups.

In identifying research topics that warrant immediate consideration, the NRC staff applied a set of criteria in a relatively judgmental way. The criteria used were the following:

- Breadth of Support: Degree of consensus among the sources of suggestions for research as well as the technical expertise of the various sources.

- Risk-Reduction Potential: The risk-reduction potential of each concept, estimated principally on the basis of insights derived from the Reactor Safety Study.[3]

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[1]The members of, and consultants to, the RRG are listed in Appendix B.

[2]Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, NUREG-0392, Washington, D.C., 1977.

[3]The term "risk" is generally defined as the product of the probability and the consequences of a reactor accident.

- Generic Applicability: The number of existing and future plants in which the projected improvements could be implemented. This criterion includes consideration of the possible times involved in implementation.

- Cost: A rough estimate of the cost of implementing the improved concept or system.

These criteria were applied to the list of suggested research topics reviewed in Chapter 2 and resulted in the recommended projects outlined in Chapter 4. Chapter 3 describes in detail how these criteria were applied.

1.4 IMPLEMENTATION

The principal products of the research program to improve reactor safety would be new analytical tools or new concepts for safety-related structures, systems, and components. A typical project would involve assessing the risk-reduction potential of the concept, developing calculational models for predicting its performance, conducting experiments that may be necessary to validate its technical capabilities, and preparing preliminary cost estimates of putting the concept into use. Application of these research results to nuclear power plants would involve their use by the nuclear industry in plant design and operation, and by the NRC reactor licensing staff in safety requirements for, and safety reviews of, individual plants.

The results of each research project for improving reactor safety would be factored into the regulatory process in the same general way as the results of the NRC's confirmatory research projects are processed. A formal procedure has been established for transmitting NRC research results to user organizations in NRC such as the licensing, standards development, and inspection staffs; the Advisory Committee on Reactor Safeguards; the Atomic Safety and Licensing Board Panel and the Atomic Safety and Licensing Appeal Panel; and the public. When a significant piece of research is completed, a formal document, referred to as a Research Information Letter (RIL), is transmitted to the applicable NRC offices, with a distribution encompassing those organizations just indicated as well as others. In much the same way, on completion of a research project to improve nuclear power plant safety, a Research Information Letter would be used to transmit the results to the appropriate NRC office(s) and would be made available to the public in the NRC Public Document Room.

When the Research Information Letter reaches the licensing or standards office, for example, consideration is given to using its results to develop a new or revised regulatory requirement. The criteria to be used in deciding whether or not to implement a particular safety improvement would depend on circumstances at that time. The decisions would require a formal value/impact analysis of the consequences of implementing the requirement as well as review by the Advisory Committee on Reactor Safeguards and the Commission. If a positive decision were made, new or revised regulations or regulatory guides would have to be developed and issued for public comment. Rulemaking hearings might also be necessary.

In response to possible future regulatory criteria covering the safety improvements developed in, or resulting from, this research program, industry would have to provide detailed designs and predicted performance data for the features proposed for implementation. These proposals would be reviewed in the NRC reactor licensing process to ensure that they comply with regulatory requirements. The industry would then have to procure equipment and install it at appropriate times relative to plant operation and regulatory requirements. Moreover, the NRC inspection and enforcement programs would be applied to supervise and audit construction and operation, as is currently done for all safety-related matters.

The time when a concept for improving reactor safety can be used depends on when the relevant research results become available; how long it takes to design, manufacture, and install the necessary hardware; and how early in the reactor design, construction, and operation process the concept must be adopted to be effective. Some concepts may be used in any light-water reactor, even an operating unit; these may be used as soon as 4 or 5 years after start of the research project. Other concepts may require such major changes in plant configuration that they can be adopted only in the early stages of plant design. Because the latter would be applicable only to newly proposed plants, the improvement in reactor safety would not be realized for at least 12 years (the current plant lead time) after the start of the research project.

1.5 FUTURE REPORTS

The Congress has asked the NRC to prepare a long-term plan for research projects directed toward improving the safety of nuclear power plants and to submit an annual report on this subject. As indicated earlier herein, this is the first such report. While it evaluates sixteen research topics and recommends some for pursuit during the next 3 years, it does not represent a complete assessment of all the research effort that may be warranted by these topics. This assessment would be completed in the coming year and may result in additional recommendations. It is also planned during the coming year to conduct a broader review of pertinent literature, further meetings of the Research Review Group, and further meetings with the Advisory Committee on Reactor Safeguards. Furthermore, there will be continuing follow-on over the years of trends and changes in reactor designs, sizes, siting, and safety requirements; this effort can also result in additional research recommendations. All of these inputs will be considered in next year's report and should result in a longer term view of research on improvements in the safety of nuclear power plants.

1.6 SCOPE OF THIS REPORT

Chapter 2 describes the process and criteria used in selecting and evaluating candidate topics for research on improved-safety concepts. In Chapter 3, the entire set of candidate topics is analyzed and evaluated; all recommendations received or identified in selected reports are listed in Appendix C. Chapter 4 translates the recommended research topics into specific projects and studies, with associated schedules and costs. Background information on ongoing reactor safety research is presented in Appendix D.

2.0 CONCEPTS FOR IMPROVING THE SAFETY OF LIGHT-WATER NUCLEAR POWER PLANTS

In developing this plan for improvement-oriented safety research, it was necessary to select from many suggestions candidates for further evaluation. The suggestions were then consolidated into sixteen research topics and evaluated to identify those that warrant immediate consideration as research projects. This chapter lists the sources of suggestions, discusses the criteria used in evaluation, and presents brief descriptions of the research topics.

2.1 SOURCES OF SUGGESTIONS

In collecting suggestions for screening, it seemed wise to turn to organizations or persons who have had extensive experience in reactor safety or who had studied reactor safety in some depth. Appendix C contains a summary of suggestions obtained from the following sources:

- Advisory Committee on Reactor Safeguards
- NRC Regulatory Staff
- Consultants to the NRC Research Review Group on Improved Reactor Safety
- Report to the American Physical Society by the Study Group on Light-Water Reactor Safety[1]
- Report of the Nuclear Energy Policy Study Group, sponsored by the Ford Foundation[2]
- ECCS Acceptance Criteria[3]
- Environmental Quality Laboratory, California Institute of Technology[4]

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[1]Published in Reviews of Modern Physics, Vol. 47, Supplement No. 1, Summer 1975.

[2]Published as Nuclear Power - Issues and Choices, Ballinger Publishing Company, Cambridge, Mass., 1977.

[3]Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, U.S. Atomic Energy Commission, Washington, D.C., Docket No. RM-50-1, December 1973. This rulemaking hearing record includes recommendations by the Consolidated Intervenors, which included, among others, the Union of Concerned Scientists.

[4]F. C. Finlayson, Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors, EQL Report No. 9, Environmental Quality Laboratory, California Institute of Technology, May 1975.

These sources provided many suggestions for improved-safety research. However, as indicated in Appendix C, there were a number of suggestions that covered matters other than improvement-oriented safety research. These included suggestions for (1) confirmatory research, (2) developmental research, and (3) licensing activities. Suggestions in classes 2 and 3 were not considered further in the context of this report; suggestions in class 1 will be considered in planning the NRC program in confirmatory research. Appendix C indicates the disposition of each of the comments contained therein.

For purposes of this report, the suggestions assessed by the staff as relevant to research for improved reactor safety were then consolidated into sixteen research topics (see Table 3-1). These research topics therefore incorporate all of the relevant suggestions that were derived from all of the sources listed above.

Although the above-mentioned sources of suggestions for improved safety were believed to be the most useful at present, the list is not exhaustive. Further study in the next year will cover other sources of information and suggestions not included in the initial survey. Moreover, additional recommendations are to be anticipated as the work described herein progresses.

2.2 EVALUATION AND SELECTION CRITERIA

The next step was the development and application of a set of criteria for use in evaluating research topics and in selecting those that appeared to warrant immediate consideration as research projects. Four criteria were used:

- Breadth of support
- Risk-reduction potential
- Generic applicability
- Cost of implementation

In evaluating a research topic, each of these criteria was used by assigning values of high, medium, or low. In some cases, because each topic included a variety of specific suggestions, more than one value was assigned for a single criterion.

As discussed in Chapter 4, it is recommended that these criteria be reexamined in detail in order to improve the methodology to be used in evaluating research topics and in aiding decisions on implementing the results to be achieved in this program.

2.2.1 BREADTH OF SUPPORT

The first criterion was breadth of support among the sources of suggestions for research. Some topics were supported by only two sources, whereas others were supported by all seven sources. Judgment was used by the staff in assessing the apparent degree of consensus among sources and the technical expertise of the various sources. Thus the staff evaluation of the breadth of support was more than just a "head count." This criterion was weighted heavily in the selection process.

2.2.2 RISK-REDUCTION POTENTIAL

An assessment of the risk-reduction potential inherent in the research topics was a significant factor in their evaluation and selection. The assessment was on a per plant basis and was based principally on engineering insights gained from the Reactor Safety Study, but a quantitative estimate of this potential was not performed for this first report. Research Project F, described in Chapter 4, should provide improved methods for the quantitative evaluation of the risk-reduction potential in future selections of projects.

In assessing the risk-reduction potential, it was necessary to consider the potential accident sequences in the Reactor Safety Study that were significant contributors to the overall risk associated with the operation of nuclear power plants. In general, these accident sequences involved two types of initiating events: either pipe breaks or operating disturbances (transients). If the probability of just one type of initiating event were to be substantially reduced, the overall accident risk would not be significantly changed because the other type would still remain as a contributor to risk at about the same level of probability. Reducing the probability of both types of initiating events below their current low levels could have a much more significant risk-reduction potential; however, additional accident sequences that had not been considered significant contributors to overall risk would have to be reexamined to determine the actual risk reduction that could be achieved.

In the same way, risks could be reduced by reducing the failure probability of safety features provided to mitigate the course of events that can follow initiating events. The specific accident sequences that can contribute significantly to potential risks have to be examined to evaluate the importance of reducing the failure probabilities of various safety features (i.e., decay heat removal systems and containment buildings).

There is also a potential for risk reduction in decreasing the probability of the radioactive releases that could be caused by accident sequences. This could be accomplished, for instance, by changing the failure probability of reactor containment buildings.

In many cases, an improvement in a system or feature would affect the plant's behavior in a variety of situations. It is likely that detailed examination of the risk-reduction potential of a particular system will disclose unforeseen effects, which may be favorable or unfavorable or both, on other systems and postulated events. Tradeoffs that enhance overall safety may be available. Such considerations will be addressed in the research program on improved safety.

2.2.3 GENERIC APPLICABILITY

The generic applicability of research results, the third criterion, was difficult to apply because it covers more than one consideration. The principal one is the number of reactors in which the research results could be implemented.

Even if a particular result (e.g., seismic decoupling) were applicable to all types of plants, it could not be assigned a high generic applicability index unless it could also be backfitted into existing plants. On the other hand, results that could be backfitted into existing plants and those under construction, as well as applied to future reactors, would have high generic applicability. Recent trends toward the standardization of plant designs[1] emphasize the importance of generic applicability. The timeliness of implementation of a particular research result also had to be considered since the longer the time required for implementation, the lower the generic applicability.

For topics that involve several potential areas of research, such as alternate emergency core cooling concepts, it was difficult to predict whether particular solutions could or should be backfitted; such topics were therefore judged to have medium generic applicability. The vented-containment concept, on the other hand, might be applicable to existing as well as to future plants and would therefore have high applicability.

Because generic applicability was difficult to evaluate, it was weighted somewhat less heavily than breadth of support and risk-reduction potential. However, in next year's report its weight may be increased if the methodology studies recommended herein provide better means of assessing generic applicability.

2.2.4 ESTIMATED COST OF IMPLEMENTATION

In assigning values for this criterion, low cost was considered to be less than \$10 million, medium cost was considered to be in the range of \$10 to \$50 million, and high cost over \$50 million, on a per plant basis. The cost would depend a great deal on whether or not backfitting was involved and whether or not plant downtime and power replacement costs were included. Because sufficient information for precise cost estimates was not available, this criterion was not weighted heavily in the overall selection of recommended research projects.

2.3 RESEARCH TOPICS

As stated earlier, suggestions for improved-safety research were consolidated into sixteen research topics, which were then evaluated in terms of the criteria described above to identify those that warrant immediate consideration as research projects. The research topics are briefly described in this section; more detailed descriptions and evaluations are presented in Chapter 3.

2.3.1 PLANT SURVEILLANCE AND OPERATION

The first group of research topics falls under the general heading of "Plant Surveillance and Operation" and is related to improvements in systems and procedures used for monitoring plant performance, during both normal operation and post-accident conditions, to reduce the likelihood of accidents or mitigate their consequences. This group includes four topics:

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[1]U.S. Nuclear Regulatory Commission, "Standardization of Nuclear Power Plants - General Statement of Policy," Federal Register, Vol. 42, No. 128, pp. 34395-34396, July 5, 1977.

1. Nondestructive Examination and On-Line Monitoring: the development of equipment and systems to perform surveillance of the primary system (i.e., the reactor pressure vessel, piping, pumps, and heat exchangers) and operating components during both inspections and operation. These techniques would be used to detect the initiation and growth of flaws in steel and other abnormal conditions that might result in accident initiation.

2. Improved Plant Controls: the use of more advanced control systems and improved operator/control interfaces during normal operation to (a) prevent or forestall accident-initiating events that originate in plant control systems and (b) to control plant disturbances and preclude their development into accidents.

3. Improved In-Plant Accident Response: the application of improved monitoring and accident diagnostic systems, and operator response during accident conditions to mitigate the consequences of accidents.

4. Reduced Personnel Exposure: the application of procedures or improved equipment to reduce the radiation exposure of plant personnel performing testing and maintenance operations.

2.3.2 SAFETY SYSTEMS

The second group, "Safety Systems," consists of suggestions for new or improved equipment, structures, and systems specifically designed to reduce the consequences of initiating events that could lead to a plant accident. These safety systems, both active and passive, provide the barriers for controlling the release of radioactivity. This group consists of the following seven research topics:

5. Alternate Emergency Core Cooling Concepts: the investigation, with emphasis on hardware, of new or improved emergency core cooling (ECC) systems to provide greater assurance that the cooling water injected into the primary system would reach and flood the core in a timely and effective way.

6. Alternate Decay Heat Removal Concepts: the use of an add-on system providing a high degree of separation and independence, passive systems, backup feedwater supplies, and other alternate concepts to improve the overall reliability of decay heat removal.

7. Alternate Containment Concepts: the development of new or improved containment concepts that would reduce the likelihood of containment failure in the event of an accident or mitigate its consequences.

8. Improved Reactor Shutdown Systems: the modification of existing systems and the development of new systems to enhance the reliability of reactor shutdown during an operating disturbance.

9. Reactor Vessel Rupture Control: the consideration of systems or design concepts that would mitigate the consequences of reactor vessel failure.

10. Core Retention Measures: the consideration of design concepts to retain and cool molten core material in the event of a core meltdown accident.

11. Equipment for Reducing Radioactivity Releases: the development of systems to further reduce routine releases of radioactivity resulting from anticipated operating disturbances.

2.3.3 PLANT CONFIGURATION AND DESIGN

The third group, "Plant Configuration and Design," covers improvements in plant design to reduce the likelihood of accidents or to mitigate their consequences. This includes reducing susceptibility to site-wide accident-initiating events and providing greater protection for components during an accident. This group consists of the following three topics:

12. Advanced Seismic Designs: the development of new plant design concepts to decouple or reduce the response of the reactor system to earthquake loads.

13. Improved Plant Layout and Component Protection: the consideration of designs that would reduce the likelihood of individual component failures causing further component or system failures and resulting in a more severe accident sequence.

14. Protection Against Sabotage: the consideration of improvements in plant design that would enhance protection against sabotage.

2.3.4 SITING AND EMERGENCY RESPONSE

The fourth group, "Siting and Emergency Response," covers suggestions for improved siting practices and offsite emergency plans to further mitigate the consequences of severe accidents. It consists of two topics:

15. New Siting Concepts: the consideration of improved siting practices, including remote plant siting and nuclear power parks.

16. Improved Offsite Emergency Response Planning: the consideration of more effective methods for evacuating or protecting the population in the vicinity of a reactor plant in the event of a major accident.

Chapter 3 presents detailed evaluations of these research topics in terms of the criteria described in Section 2.2.

3.0 EVALUATION OF RESEARCH TOPICS AND SELECTION OF RESEARCH PROJECTS

The sixteen research topics encompassing suggestions for research on reactor safety improvement are described and evaluated in this chapter. A qualitative assessment of each topic in terms of the criteria used in evaluation is presented in Table 3-1. The objective of the evaluation was to identify projects for the proposed NRC research program for improving the safety of light-water nuclear power plants.

Since the selection of topics for research projects necessarily involved a good deal of judgment, and because research related directly to improved safety represents a new area of endeavor for the NRC, it was decided to recommend only those projects that appeared to be most certain to have the potential for improving the safety of nuclear power plants. The scoping studies to be carried out in the initial research program (see Chapter 4) should provide the basis for developing a comprehensive, long-range program plan in the future.

3.1 SUMMARY OF EVALUATIONS

Several recommendations for improved-safety research emerged from the detailed evaluations of sixteen research topics described in this chapter and summarized in Table 3-2. Six topics (research topics 1, 4, 8, 14, 15, 16) are accommodated by current NRC programs but should be included in the scoping studies proposed in the program described herein. Five topics (2, 9, 10, 11, 13) deserve scoping studies to determine whether research programs are needed or appropriate. The scoping studies would be the principal vehicle for developing the comprehensive, long-range plan for improved-safety research that is mandated by Congress.

Seven research projects have been found to warrant undertaking:

- Research Project A: Alternate Containment Concepts (Research Topic 7).
- Research Project B: Alternate Decay Heat Removal Concepts (Research Topic 6).
- Research Project C: Alternate Emergency Core Cooling Concepts (Research Topic 5).
- Research Project D: Improved In-Plant Accident Response (Research Topic 3).
- Research Project E: Advanced Seismic Designs (Research Topic 12).

TABLE 3-1. CONCEPTS FOR THE IMPROVEMENT OF REACTOR SAFETY

RESEARCH TOPIC	SOURCE: ACRS NUREG-0392[a]	NRC STAFF	CONSULTANTS	APS STUDY GROUP REPORT[b]	FORD FOUNDATION STUDY[c]	ECCS ACCEPTANCE CRITERIA[d,e]	ENVIRONMENTAL QUALITY LABORATORY[e,f]		QUALITATIVE ASSESSMENT MATRIX: RISK REDUCTION POTENTIAL H	RISK REDUCTION POTENTIAL M	RISK REDUCTION POTENTIAL L	GENERIC APPLICABILITY H	GENERIC APPLICABILITY M	GENERIC APPLICABILITY L	ESTIMATED COST OF IMPLEMENTATION H	ESTIMATED COST OF IMPLEMENTATION M	ESTIMATED COST OF IMPLEMENTATION L
PLANT SURVEILLANCE AND OPERATION																	
1. NDE and On-Line Monitoring		O	O	O						O —	→	O					O
2. Improved Plant Controls		O	O	O	O						O	←	— O				O
3. Improved In-Plant Accident Response		O	O	O					O —	→		O					O
4. Reduced Occupational Exposure	O	O	O							O —	→	←	— O				O
SAFETY SYSTEMS																	
5. Alternate Emergency Core Cooling Concepts	O	O	O	O	O	O	O			O			O			←	— O
6. Alternate Decay Heat Removal Concepts	O	O	O	O					O			←	— O			O	
7. Alternate Containment Concepts	X	O	O	O	O				O			O —	—	→	←	— O	
8. Improved Reactor Shutdown Systems			O		O					O			O —	→		O	
9. Reactor Vessel Rupture Control	R				O						O			O	←	— O	
10. Core Retention Measures	O		O	O							O		O —	→		O —	→
11. Equipment for Reducing Radioactivity Releases	R	O	O								O	←	— O			←	— O
PLANT CONFIGURATION AND DESIGN																	
12. Advanced Seismic Designs	O	O	O						O —	→			←	— O	←	— O	
13. Improved Plant Layout and Component Protection	O	O	O							O —	→		←	—		←	— O
14. Protection Against Sabotage	O	O	O	O	O								O			←	— O
SITING AND EMERGENCY RESPONSE																	
15. New Siting Concepts	X	O	O	O						O			←	— O		O —	→
16. Improved Off-Site Emergency Response Planning	O		O	O	O					←	— O	O					O

[a] The symbol R indicates recommendations made in ACRS reports but not identified in NUREG-0392; X indicates that recommendation could be interpreted as being in more than one concept category.

[b] Published in Reviews of Modern Physics, Vol. 47, Supplement No. 1, Summer 1975.

[c] Published as Nuclear Power-Issues and Choices, Ballinger Publishing Company, Cambridge, Mass., 1977.

[d] Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, U.S. Atomic Energy Commission, Washington, D.C., Docket No. RM-50-1, December 1973.

[e] ECCS was the principal focus of this document

[f] F.C. Finlayson, Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors, EQL Report No. 9, Environmental Quality Laboratory, California Institute of Technology, May 1975.

TABLE 3-2
SUMMARY OF RESEARCH TOPIC EVALUATIONS

No.[a]	Research Topic[b]	Breadth of Support[c]	Risk-Reduction Potential	Generic Applicability	Cost	Disposition
7	Alternate Containment Concepts	H	H	H	M	Initiate Research Project A[d]; study further
6	Alternate Decay Heat Removal Concepts	H	H	M	M	Initiate Research Project B; study further
5	Alternate Emergency Core Cooling Concepts	H	M	M	L	Initiate Research Project C; study further
3	Improved In-Plant Accident Response	H	H	H	L	Initiate Research Project D; study further
16	Improved Offsite Emergency Response Planning	H	L	H	L	Accommodated in current program; evaluate further
14	Protection Against Sabotage	H	e	M	L	Accommodated in current program; evaluate further
12	Advanced Seismic Designs	M	H	L	M	Initiate Research Project E; study further
1	NDE and On-Line Monitoring	M	M	H	L	Accommodated in current program; evaluate further
15	New Siting Concepts	M	M	L	M	Limited effort needed; accommodated in current program; evaluate further
13	Improved Plant Layout and Component Protection	M	M	L	L	Evaluate further
2	Improved Plant Controls	M	L	M	L	Evaluate further
4	Reduced Occupational Exposure	L	M	M	L	Partly accommodated by current program; evaluate further
8	Improved Reactor Shutdown Systems	L	M	M	M	Accommodated in current program; evaluate further
10	Core Retention Measures	L	L	M	M	Evaluate further
11	Equipment for Reducing Radioactivity Releases	L	L	M	L	Evaluate further
9	Reactor Vessel Rupture Control	L	L	L	M	Evaluate further

[a] Index for Table 3-1.

[b] The order of topics in this table is governed by the staff's judgment regarding breadth of support, from highest to lowest.

[c] H = High, M = Medium, L = Low; the values shown in this table correspond to the circles and not the arrows in Table 3-1.

[d] See Chapter 4 for project descriptions.

[e] Not known at present.

- Research Project F: Improved methodology for evaluating research topics.

- Research Project G: Scoping studies.

These research projects are described in Chapter 4, which also presents information on costs and schedules.

3.2 PLANT SURVEILLANCE AND OPERATION

RESEARCH TOPIC 1: NONDESTRUCTIVE EXAMINATION AND ON-LINE MONITORING

Description

Surveillance programs are conducted at nuclear power plants in order to ensure the adequacy of the quality achieved in plant construction and maintained during the lifetime of the plant and to aid in identifying incipient failures that could be accident-initiating events. Surveillance also operates to reduce the likelihood of accidents.

The types of surveillance currently in use or under development are nondestructive examination of the primary system pressure boundary, in-core irradiation of fracture mechanics specimens, on-line systems for detecting the initiation and growth of flaws in steel, and on-line monitoring for abnormal behavior that might indicate an incipient failure. Surveillance is not restricted to the primary system pressure boundary; it can also cover loose parts, vibrations, reactor vessel internals, and other plant equipment.

The NRC is currently supporting research to improve the characterization of flaws in steel by ultrasonic techniques. These efforts are centered on the use of computers to provide a more precise and three-dimensional reconstruction of flaws by the synthesis of multiply reflected pulses to permit better evaluations of their severity and significance to component integrity. Adaptive learning, a relatively new technique of flaw detection, is being investigated by the Electric Power Research Institute. This method uses a computerized system that can recognize the ultrasonic pattern produced by flaws.

To better detect the appearance and growth of flaws after components have been placed in operation, systems for on-line monitoring are also being developed. The objective of on-line monitoring systems is early detection of the initiation and growth of small flaws. To this end, the NRC is currently studying the use of acoustic emission to detect flaw growth. One reactor vendor and the Department of Energy are already testing an on-line system based on these principles. In addition, the NRC is studying the use of acoustic monitoring for the detection of flaws introduced during welding.

The NRC is also studying on-line systems to monitor for unusual vibrations, loose parts, or indications of abnormal neutronic or thermal-hydraulic conditions. These systems are intended to recognize incipient failure capable of initiating an accident before it occurs.

Evaluation

Improvement in surveillance systems and techniques has been recommended by the NRC staff, by the consultants to the NRC Research Review Group, and by the American Physical Society's (APS) Study Group on LWR Safety. However, the Advisory Committee on Reactor Safeguards has indicated that the present level of research is adequate. Recommendations for improved surveillance represent primarily a recognition that reducing the probability of accident-initiating events is an important aspect of reactor safety. All specific topics that were recommended are currently receiving some level of research support.

Quantitative risk analyses indicate that accident sequences initiated by pipe breaks contribute to the total risk associated with the operation of nuclear power plants. However, it is not likely that significant gains in the reduction of pipe-break probabilities can be demonstrated in the near future, even though significant work in this area is going on in confirmatory research. It will be easier to demonstrate risk reduction by improvements in the reliability of safety systems in accident sequences that are initiated by pipe breaks. Since much of the unreliability of such systems is due to personnel performance in testing and maintenance, as discussed in Research Topic 3, that area is being given high priority as part of this research program.

It must also be recognized that improved nondestructive-examination techniques will have a somewhat limited effect on risk reduction because not all pipe breaks are attributable to the propagation of flaws. Some breaks can be caused by unanticipated loads, such as water hammer, and would not be prevented by improved nondestructive examination. Nor can improved nondestructive examination or on-line monitoring systems be expected to be completely effective in detecting flaws. The risk-reduction potential resulting from improved flaw detection is therefore evaluated as medium. Similarly, loose-part monitors or vibration detectors that would probably be able to identify an incipient failure would not be of very great significance in risk reduction because only a small portion of such failures might be accident-initiating events. The risk-reduction potential resulting from the use of such equipment is hence evaluated as medium to low.

The cost of nondestructive examination or on-line monitoring equipment for a plant would be low. The equipment and techniques would have high generic applicability and after a period of development could be implemented rapidly.

Although the risk-reduction potential of plant surveillance is not as great as that of some other concepts, its relatively low cost and high generic applicability indicate that further development is warranted. The present NRC confirmatory research program is covering this area adequately and will be continued, perhaps with some modification as it is reexamined in the next year.

RESEARCH TOPIC 2: IMPROVED PLANT CONTROLS

Description

This topic is concerned with (a) preventing or forestalling accident-initiating events that originate in the plant control systems, and (b) controlling plant disturbances to prevent their development into accidents. The term "plant control" embraces the controllers and other devices usually thought of in this connection as well as the people involved in the control function. The plant operators in the control room are a primary consideration; however, operating and maintenance personnel in other areas of the plant also can affect the reliability and effectiveness of the plant control function.

Inadequate performance of the control system could affect reactor safety mainly by causing a somewhat increased demand for operations of automatic reactor shutdown systems. If the control system cannot maintain process system perturbations within specified levels, the reactor shutdown system is called on to shut down the reactor. This may occur because the perturbation exceeds the capability of the control system, because of spurious operation of the control system, or because of operator actions.

Improvements in plant control systems would not be likely to reduce by more than a small amount the frequency of operating disturbances requiring reactor shutdown and hence demands on the reactor protection system.

Recommendations have been made for instrumentation and limit controls to prevent plant parameters from exceeding operating limits. For example, controls could be used to prevent the primary system pressure from exceeding the pressure-temperature limit curves during startup, thus providing a means of reducing the frequency of overpressurization incidents. It has also been suggested that more advanced control systems are available than are currently used in nuclear power plants.

Evaluation

Plant control systems have been identified by the NRC staff, by the consultants to the NRC Research Review Group, in the APS Study Group Report, and by the Ford Foundation study[1] as candidates for safety improvement. No recommendation for improvements has been made by the Advisory Committee on Reactor Safeguards. Furthermore, risk analyses have not indicated that control system inadequacies or malfunctions are large contributors to risk. However, human errors may be more important. By improving the performance of control systems, the need for operator actions and the associated likelihood of operator errors can be reduced.

It is believed that only a small reduction in risk could result from improvements in plant controls. The potential benefits of adopting such improvements would also have to be carefully evaluated to determine that safety would be actually improved.

The cost would be low. Furthermore, since economic as well as safety gains might result, the industry may explore this area voluntarily. Improved control systems would have medium to high generic applicability, including implementation

[1]Nuclear Power - Issues and Choices, Report of the Nuclear Energy Policy Study Group, Ballinger Publishing Company, Cambridge, Mass., 1977.

in existing plants. It does not seem that additional research is needed for evaluation purposes. This area should be examined further to assess its potential value and impact more completely.

RESEARCH TOPIC 3: IMPROVED IN-PLANT ACCIDENT RESPONSE

Description

This topic deals with what the plant operators can and should do during a developing accident situation. Factors involved include the plant-status information available to the operators from instruments and indicators, operator training and procedures related to such situations, and human responses under stress conditions.

Human factors are believed to have a major influence on the availability of safety systems to operate when needed. The actions of operators and other personnel in testing and maintenance can be important in this regard. While a redundant (i.e., backup) component of a safety system is off line for testing or maintenance, the extra margin of safety inherent in the principle of redundancy could be reduced. Thus the frequency of testing and maintenance and the offline time could affect the availability of the system when needed. The computer code FRANTIC[1] has been developed by the NRC to optimize the testing interval for availability. Such analyses can be used to establish regulatory requirements that would improve the reliability of existing safety systems. Analyses have also shown that, in the performance of test and maintenance operations, components may be left in an unavailable state through personnel carelessness, improper training, use of improper procedures, or failure to follow procedures.

Improved accident-monitoring systems have been suggested as a means of improving safety. Computerized processing of data is potentially a useful diagnostic tool to assist the operator during an accident. Improvements can also be made in control-room layout and data presentation. In the design of annunciators, warning lights, and display panels, greater attention could be given to the human factors that influence the operator's ability to understand the condition of the plant and to make proper decisions. In addition, improved instrumentation could be developed to aid the operator in evaluating the condition of the plant (e.g., to measure the water level in the reactor core).

Research is currently being undertaken to gain greater understanding of human errors and to develop methods for reducing their likelihood. A major program supported by the Electric Power Research Institute has been studying approaches to improving the design and layout of control rooms. Studies supported by the NRC in the confirmatory research program are attempting to determine the principal causes of human errors, with emphasis on a better understanding of the man/machine interface. Human-error data are being evaluated to quantify the likelihood of errors and identify the dominant contributing factors. Systematic approaches are also being developed to quantify the effects of specific human errors on system performance and accident response.

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[1]W. E. Vesely and F. F. Goldberg, "Time-Dependent Unavailability of Nuclear Safety Systems," IEEE Transactions on Reliability, R6(4), 257-260 (1977).

Recommendations have also been made to improve the training of operators to respond to accident conditions. The qualifications and training requirements for operators are generally recognized to be stringent. Existing training programs use simulators that teach operators how to respond to abnormal plant behavior under quite realistic conditions. The extent to which the present level of training can be improved has not been investigated.

Evaluation

In-plant accident response has been identified as an area of potential improvement by the NRC staff, by the consultants to the NRC Research Review Group, and in the APS Study Group report. Risk analyses indicate that human errors in testing and maintenance are a significant cause of the unavailability of safety systems when needed to respond to accidents. A reduction in the frequency of these errors could therefore have a significant impact in reducing accident risks. Errors introduced by the operator in responding to an accident are indicated by risk analyses to make a significant contribution to the frequency of some accident sequences. Improvement in the human response to accidents or in increased automation of accident response could lead to a high to medium reduction in risk.

The cost of improvements in control room design, test and maintenance procedures, training, and equipment for accident monitoring or controls should be low. The techniques and equipment should have high generic applicability.

Because of its high to medium potential for risk reduction, low cost, and general applicability, this area has been assigned a high priority for research (see Chapter 4, Research Project D). This research would begin by evaluating the results of past and current studies to determine what additional analyses may be appropriate in future years.

RESEARCH TOPIC 4: REDUCED OCCUPATIONAL EXPOSURE

Description

Title 10, Part 20, of the Code of Federal Regulations requires that, in addition to complying with quantitative limits for permissible radiation exposures, licensees "make every reasonable effort to maintain radiation exposures...as low as is reasonably achievable" (ALARA). While such exposures are outside the bounds of considerations pertinent to reactor accidents, it should be noted that the overall annual exposures received by plant personnel performing test and maintenance operations are significantly larger than the average low-level doses that would be received by the public as a result of accidents. The commonest sources of such exposures are unanticipated special maintenance operations, such as changeouts of steam generators.

Lower occupational exposures could be achieved by designing plants so as to reduce the radiation levels in maintenance areas or to decrease the time required for performing test and maintenance operations. Remotely operated equipment, such as remote welders, could also be developed to reduce exposure. The Department of Energy is currently testing techniques for periodically reducing the contamination of the primary system, which would help to reduce personnel exposure. In its confirmatory program, the NRC supports research to identify the sources of radiation exposure received by plant operating and maintenance personnel.

Evaluation

The desire to reduce occupational exposure has been expressed by the Advisory Committee on Reactor Safeguards, by the NRC staff, and by the consultants to the NRC Research Review Group. While such exposures are not related to accident considerations and the overall risk to plant personnel is considered to be small, it seems worthwhile to study this area further because the overall radiation doses received in such exposures are higher than those postulated for reactor accidents and the value of medium to low that has been assigned to the risk-reduction potential of this topic is quite judgmental.

Some reduction in occupational exposure should be possible at low cost. The techniques that could be developed would have medium to high applicability to various types of reactors.

Additional effort should be expended in confirmatory research and in scoping studies to identify in more detail the direct causes of occupational exposure and to evaluate the means by which the principal contributors could be reduced.

3.3 SAFETY SYSTEMS

RESEARCH TOPIC 5: ALTERNATE EMERGENCY CORE COOLING CONCEPTS

Description

In the event of a pipe-break accident in both boiling and pressurized water reactors, the emergency core cooling (ECC) system must operate to cool the reactor core and prevent a serious accident. Concern about ECC effectiveness has been related mainly to the difficult and complex calculations needed for analyzing the performance of ECC systems in large-pipe-break accidents in pressurized water reactors.

An extensive experimental and theoretical research program related principally to ECC systems of current design is being conducted as part of the NRC confirmatory research program. Research in this area is also being conducted by the U.S. nuclear industry and by several foreign countries.

Suggestions have been made for various sorts of improvements to current systems to increase the assurance that the emergency core cooling water injected into the primary system would reach and flood the core in a timely and effective way.

In addition, suggestions have been made to seek new and different approaches to the implementation of the ECC function, so as to reduce the present reliance on complex calculations of system performance.

Tests with the Semiscale[1] experimental apparatus, which simulates pipe-break accidents in pressurized water reactors, have indicated the potential value of several advanced ECC concepts involving alternate locations for the injection of ECC water. Additional tests and analyses are needed to determine the extent to which the safety margin in ECC system performance could be increased with the various new concepts that have been proposed.

Modifications have also been suggested that might improve the performance of ECC systems in large-pipe-break accidents in boiling water reactors.

Evaluation

The development of improved ECC concepts has been recommended by all of the sources listed in Appendix C. The principal focus of these recommendations relates to the development of alternate ECC concepts that would provide adequate safety margins and whose adequacy could be demonstrated by simpler calculational methods than those required for current systems. This focus relates principally to large pipe breaks since there is very much less uncertainty concerning the safety margins available for small pipe breaks. The operational reliability of ECC systems for small pipe breaks can be significantly affected by the performance of test and maintenance personnel and will be explored as part of Research Topic 3. Since large pipe breaks do not contribute significantly to accident risks and since it is not clear that the operational reliability of ECC systems can be easily improved, the risk-reduction potential of improved ECC systems has been evaluated as medium.

Advanced ECC concepts could be implemented at medium cost. Some concepts could be incorporated into existing designs; others might be applicable to new plant designs only.

Based mainly on the breadth of support from the sources listed in Appendix C, advanced ECC concepts related to system performance in large-pipe-break accidents should be given high priority for research. A program should be initiated (see Chapter 4, Research Project C) to evaluate the risk-reduction potential of the various concepts that have been proposed and any others that appear promising. These analytical studies will also require data from experiments in existing, but modified, facilities (i.e., the Semiscale facility).

RESEARCH TOPIC 6: ALTERNATE DECAY HEAT REMOVAL CONCEPTS

Description

Even after the nuclear chain reaction in a power reactor has been stopped, heat is produced by the radioactive decay of the fission products in the core. The reactor core must therefore be cooled for an extended period of time. This is

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[1]Semiscale is an experimental facility operated for the NRC at the Idaho National Engineering Laboratory to study loss-of-coolant accident behavior.

the function of the decay heat removal system. Because of the moderate frequency of transient events that require plant shutdown, these systems must have high reliability.

Among the suggestions that have been made for improving the reliability of decay heat removal systems are the following:

1. An add-on system providing a high degree of separation and independence from the remainder of the plant to decrease the likelihood of common-mode failures. This system could also have power supplies, water supplies, and heat sinks that would be independent of the plant. It could also have improved seismic design features, optimum fire protection, and housing in a bunker to protect it from external events that could affect its operability.

2. Increased separation and improved protection for the normal complement of plant decay heat removal systems to improve reliability for the hot- and cold-shutdown phases of operation (see Research Topic 13).

3. More emphasis on passive systems or systems with minimal dependence on active components.

4. Provision of, or improvement in, the ability of disconnecting the plant from its load and continuing to operate.

5. The use of backup feedwater supplies such as fire trucks.

The heat that is removed from the core must be released to the environment through an ultimate heat sink. Frequently the ultimate heat sink for a nuclear power plant is a river, canal, or lake. Since the effectiveness of the heat sink may be affected by the climate, natural events, and man-made events, it has been recommended that additional heat sinks, such as deep water wells, be used to provide additional reliability.

Evaluation

The investigation of alternate decay heat removal concepts was recommended by most of the sources reviewed, including the Advisory Committee on Reactor Safeguards. Risk analyses also indicate the importance to reactor safety of reliable decay heat removal systems. It has been shown that transient accidents involving the failure of decay heat removal systems are significant contributors to risk, and hence this research topic is considered to have a high risk-reduction potential. The cost of typical improvements in decay heat removal systems is expected to be medium. Since many of the concepts that have been recommended would be applicable to existing plants, this topic is judged to have medium to high generic applicability.

Since alternate decay heat removal concepts have a high risk-reduction potential and many recommendations have been made for new approaches, research in this area should be given high priority (see Chapter 4, Research Project 3). It should include scoping studies of candidate alternate systems and a study of design criteria and risk tradeoffs for an add-on system, with the degree of bunker protection as a parameter to be studied further.

RESEARCH TOPIC 7: ALTERNATE CONTAINMENT CONCEPTS

Description

The containment building, which surrounds the reactor and the primary coolant system, is provided to inhibit the release of radioactivity to the environment. Analyses of severe accidents that involve melting of the reactor core indicate that the radioactivity-removal processes that occur within the containment can be significant in limiting the consequences of accidents of this type.

The objective of alternate containment designs is to reduce the probability of containment failure and subsequent releases of airborne radioactivity. Some of the concepts that have been proposed include passive containment systems as well as containments with larger volumes or higher pressure-retention capability; the use of lower initial pressures has also been suggested. A preliminary study indicates that controlled containment venting to the atmosphere is a promising approach; this concept involves the use of appropriate filters to trap the radioactivity.

Underground siting has also been suggested as a concept for reducing the risk associated with the operation of nuclear power plants. A number of approaches to underground siting have been considered; they differ in the depth of burial, the types of soil or rock selected for burial, the types of plant feature located above or below ground, and the type of penetration to the surface. Some of the proposed approaches offer the possibility of improved resistance to earthquakes, reduced accident consequences, and reduced susceptibility to sabotage.

The containment might also be modified to provide additional protection against the consequences of energetic fuel-water interactions and the rapid burning of hydrogen in accidents involving core melting. Concepts for the retention of a molten core are discussed separately under Research Topic 10. These concepts are very closely related to improved containment capability, however, and will be so evaluated.

Evaluation

Research on alternate containment concepts has been recommended by almost all of the sources listed in Appendix C. The risk-reduction potential of alternate containment designs may be higher than that of any other concept for improved safety. However, some of the alternate containment concepts would be high in cost and would be applicable only to plants not yet designed. The vented-containment concept appears especially attractive because its cost would be moderate and because it could be implemented in many existing plants. Underground siting, on the other hand, would be applicable only to plants not yet designed. The estimated costs of different approaches to underground siting vary substantially.

Alternate containment concepts appear to be a promising area for safety improvements and have been evaluated as having high priority for research (see Chapter 4, Research Project A). An analytical program should be undertaken to evaluate the potential value of different approaches to filtering and venting. If necessary, an experimental program to test the efficacy of various filter materials should be undertaken. The effect of venting on the response of the plant should be examined for a variety of accident sequences, including some involving core melting and the spectrum of non-core-melt events. Experimental requirements or deficiencies of data required to complete a value/impact analysis would be identified in the study.

Additional research on underground siting should be deferred until the results of the California study[1] have been completed and have been reviewed by the NRC. Preliminary evaluations of other alternate containment concepts should be undertaken to identify the most promising approaches.

RESEARCH TOPIC 8: IMPROVED REACTOR SHUTDOWN SYSTEMS

The importance of reliable shutdown systems is recognized in the NRC General Design Criteria, which require diverse systems in nuclear power plants. Each light-water reactor has shutdown systems that insert the control rods and inject a boron solution into the coolant to reduce the reactivity. Because of the moderate frequency of operational disturbances (e.g., loss of electrical load to the turbine generator) that require their operation, reactor shutdown systems must have high reliability. Both modifications to existing designs and new concepts for reactor shutdown have been suggested. The reliability of reactor shutdown systems could be improved by providing greater independence among the banks of control rods or by developing new diverse systems with faster response.

Evaluation

Improvements in reactor shutdown systems were recommended by only two of the sources reviewed. Risks analyses have shown that transient accident sequences involving failure to scram are a significant contributor to the risk calculated for the boiling water reactor, but not for the pressurized water reactor. Improvements in reactor shutdown systems are therefore evaluated as having medium potential for risk reduction.

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[1]A multicontractor study, being performed in response to an act of the State Legislature, to investigate the possible requirement in California of placing future nuclear power plants underground.

The cost of implementing improved reactor shutdown systems for new plants is expected to be medium; backfitting costs would be medium to high. (Overall, the cost has been assigned a value of medium.) Although some concepts could be backfitted into existing plants, the generic applicability of improved reactor shutdown systems is considered to be medium to low.

Over the past few years, the NRC staff has been conducting a thorough investigation of the adequacy of reactor shutdown systems in current light-water-reactor designs. The Regulatory position on anticipated transients without scram is expected to be released in the near future. The results of this study should provide a better understanding of the need for improved reactor shutdown systems. A scoping study is recommended to identify needed improvements.

RESEARCH TOPIC 9: REACTOR VESSEL RUPTURE CONTROL

Description

Because existing plant safety features would not be capable of performing their design functions for some of the ways in which reactor vessels might fail, additional features have been suggested to mitigate the consequences of such failure. Some of the concepts that have been proposed would provide protection against missiles generated from a burst vessel; others would provide a supplemental flooding system, such as a system for flooding the reactor cavity, which might be able to cool a molten core inside the reactor vessel.

Current practice emphasizes ensuring the reliability of reactor vessels rather than mitigating the consequences of their failure. Considerable research has been completed, and some is still under way in NRC's Heavy Section Steel Technology Program, to gain better insight into the mechanisms that could potentially cause reactor vessel failure and to improve the quality of reactor vessel steels. Quality control in the fabrication of the vessel and nondestructive examination are important procedures that are needed to substantiate the quality of vessels and to reduce the probability of their failure.

Evaluation

Research on reactor vessel rupture control was recommended by the Ford Foundation study and by the Advisory Committee on Reactor Safeguards in reports submitted in 1965 and 1974. However, the recent ACRS report NUREG-0392 does not recommend any research on this topic. Surveys of the history of vessel failure indicate that the probability of reactor vessel failure is small. Risk analyses have shown that, at the expected probability of failure or at probabilities 100 times greater, the contribution of reactor vessel failure to potential accident risks would be small.

The cost of implementing a vessel rupture control device or a cavity-flooding system is expected to be medium to high. These concepts are probably applicable only to new plant designs and are thus considered to have low generic applicability.

Reactor vessel rupture control has been assigned a relatively low priority for research. Scoping analyses should be performed to determine the range of design parameters that might characterize a cavity-flooding system or an energy-absorption system and also provide preliminary value/impact analyses for such approaches.

RESEARCH TOPIC 10: CORE RETENTION MEASURES

Description

The function of core retention measures would be to cool, and thus to retain within containment, the molten core materials that could result from accident sequences in which the reactor core would melt. Successful retention of molten core materials could reduce their potential for interacting with concrete and penetrating the containment floor. It could also reduce the release of radionuclides into groundwater below the containment base. Various concepts for core retention have been suggested in the past. These include the use of ceramic barriers, low-melting-point materials for heat-absorption purposes, types of concrete that would minimize the generation of noncondensable gases on exposure to the molten materials, and core retention devices with active cooling systems attached. All of these past concepts have raised questions as to how much confidence could be placed in such measures without more knowledge of core meltdown phenomena and the mechanisms involved in retaining large quantities of molten fuel and steel.

Evaluation

Research into the development of core retention measures was recommended in the APS Study Group report, and further studies in this general area have been recommended by the Advisory Committee on Reactor Safeguards (ACRS). In NUREG-0392, the ACRS recognized the substantial efforts currently under way in NRC's confirmatory research program to explore the behavior of molten materials. The ACRS also recognized that the development of definitive models for core-melt phenomena would be a relatively long term project (e.g., 3 to 5 years) and that this work should continue on a priority basis. Thus in-depth assessments of core retention measures would also be relatively long term efforts since they would depend on the information being generated in the present confirmatory research program.

In assessing the potential utility of core retention devices in reducing accident risks, it is necessary to compare the risks associated with the release of large amounts of airborne radioactivity (in the event of containment rupture by overpressurization after a core-melt accident) and those associated with the core's melting through the containment floor. The latter risks have been estimated to be a small fraction of the former. Thus the principal usefulness of a core retention device would be to reduce the probability of containment rupture by overpressurization. However, there are a number of potential accident sequences (i.e., those involving steam or hydrogen explosions) that could cause containment rupture even in the presence of a core retention device. Because of this, the risk-reduction potential of core retention devices is believed to be low. It should also be noted that the ACRS restated in NUREG-0392 its 1974 recommendation that "means of retaining molten core or ameliorating the consequence" be studied. Research Topic 7 (alternate containment concepts) is related to amelioration of the consequences and has been assigned a high priority for research because of its high risk-reduction potential.

Based on the values assigned through evaluation criteria, core retention measures should be given relatively low priority for research. As already stated, there is a close relationship between this research topic and alternate containment concepts (Research Topic 7). Therefore these topics and the potential tradeoffs between them should be considered in context. Additional study should be given to core retention concepts, with emphasis on those that may potentially reduce the effect of molten core-concrete interactions on the containment pressure response. The study would be integrated with the work performed on Research Topic 7.

RESEARCH TOPIC 11: EQUIPMENT FOR REDUCING RADIOACTIVITY RELEASES

Description

In the safety review of nuclear power plants, the potential release of radionuclides is estimated for a broad range of postulated incidents, up to and including those accidents the plants are designed to mitigate. Realistic analyses of a variety of such accident sequences indicate that the potential releases would be much smaller than those resulting from accidents involving melting of the core. However, the anticipated frequency of these events can be expected to be significantly higher than that of core-melt accidents. It has thus been suggested that the aggregate risk from such relatively small accidents may be significant, and systems that would further reduce the consequences of these accidents have been proposed.

Evaluation

The Advisory Committee on Reactor Safeguards, the NRC staff, and the consultants to the NRC Research Review Group have recommended investigation of concepts that would reduce the consequences of non-core-melt accidents, including the possibility of releases from a plant that is shut down. A study is currently being undertaken by the NRC to better understand the contribution to potential risks of accidents that do not involve core melting. The results of this study should aid in determining the need for, and the potential value of, design modifications for reducing radioactivity releases in such accidents. Preliminary analyses have indicated that, for current plant designs, non-core-melt accidents make a relatively small contribution to the overall risk. The cost of implementing equipment for reducing releases of radioactivity in such accidents would be low to medium. Improvements in this area would have medium to high generic applicability and could probably be backfitted if required.

Because of its low risk-reduction potential, this topic should have low priority for research. The current NRC confirmatory research program is covering this area adequately and will be continued, perhaps with some minor modifications.

3.4 PLANT CONFIGURATION AND DESIGN

RESEARCH TOPIC 12: ADVANCED SEISMIC DESIGNS

Description

Nuclear power plants are designed to withstand the effects of the so-called operating-basis and safe-shutdown earthquakes, which are defined on the basis of past seismic activity in the region of specific sites. However, at some level of probability, it is possible to postulate an earthquake that could exceed the design margins of the plant and result in failures of structures and equipment, leading to possible core-melt consequences. Advanced concepts for plant design have been recommended that could potentially increase seismic safety margins. Suggested approaches to advanced seismic design include the following:

1. Increased energy-absorption capability. This is an extension of existing design practice and is based on permitting small amounts of inelastic deformation in materials to absorb some of the seismic energy.

2. Component isolation from seismic motions. A limited survey of current methods was conducted in fiscal year 1977. The effectiveness of isolation for very large components and interactions among mechanically connected components with different responses need further investigation.

3. Isolation of the foundation. It is theoretically possible to isolate large structures from horizontal and, to a lesser extent, vertical seismic motions by using crushable materials or damped springs in the foundations. However, the use of low-friction bearings and lateral restraints to isolate large structures from horizontal earthquake motions at the foundation is generally regarded as more feasible. Some conventional structures employing isolation systems have been designed.

4. Flotation in a fluid-filled basin. Flotation may isolate a structure from most horizontal and vertical seismic motions. Several conceptual designs have been proposed, and some licensing experience is available in the evaluation of the proposed offshore floating nuclear plants.

In addition, it has been proposed that seismic resistance be tested by means of large shaker tables, shaker devices attached to completed structures, and explosives. It has also been suggested that safety margins could be increased by increasing the peak-acceleration values and/or extending the spectrum shape used in establishing the bases for seismic design.

Evaluation

Recommendations for advanced seismic designs were made by the Advisory Committee on Reactor Safeguards, the NRC staff, and the consultants to the NRC Research Review Group. In risk analyses earthquakes are considered as potential causes of common-mode failure of safety systems occurring simultaneously with an accident-initiating event. Although methods of probabilistic analysis for quantifying the risk contribution of earthquakes are still under development, advanced seismic designs are considered to have a high to medium risk-reduction potential.

The cost of advanced seismic designs could vary substantially with the specific concept, but has been ssigned a value of medium to high. Most concepts would be applicable only to new plants, and hence this topic is considered to have low to medium generic applicability.

Primarily because of the recommendations for research, advanced seismic designs have been assigned a high priority (see Chapter 4, Research Project E). A preliminary evaluation should be performed of the concepts that have been recommended. Work in the first year of this project would identify research needed to support a later value/impact analysis of any concepts found to be feasible and advantageous.

RESEARCH TOPIC 13: IMPROVED PLANT LAYOUT AND COMPONENT PROTECTION

Description

The arrangement of systems and components in the plant affects their potential for interaction with one another as well as the potential for concurrent adverse effects from a common event or environment. This has the potential for adversely affecting the availability of safety functions in accident situations. For example, the rupture of high-pressure piping could cause pipe whip and the possible resulting failure of neighboring pipes or other equipment. The impact of the jet of water emitted from a break could also damage equipment located nearby.

One aspect of research related to improved plant layout and design is to perform a series of overall plant layouts and consider various possible alternatives that may improve safety. In this approach a set of criteria would be established so that tradeoffs in designs may be made; that is, different designs would be measured against the criteria and jugments made on the relative importance of meeting each criterion. An example would be the tradeoff of minimizing the length of piping to reduce the chances of a pipe break versus the need to separate structures for inspection and maintenance.

Recommendations have been made for improved plant layouts that would provide increased protection for components from such influences and thus reduce the potential for common-mode system failures. The suggested improvements include increased separation of plant components, increased separation of redundant components of a system, improved criteria for the use of piping restraints, and the use of energy-absorbing materials to protect vital equipment.

Many of the concepts that have been recommended could provide additional protection against other potential sources of common-mode failures such as fires or cavity flooding.

Evaluation

Recommendations for improved plant layout and component protection have been made by the Advisory Committee on Reactor Safeguards, by the NRC staff, and by the consultants to the Research Review Group. Improvements in this aspect of plant design were judged to have medium to low risk-reduction potential. Although common-mode contributors are particularly important to reactor risk because of the existence of redundancy in safety systems, risk analyses indicate that human error is a greater contributor to common-mode failure than physical proximity.

Many of the concepts would be applicable to all types of plant designs, but some, such as those involving increased physical separation, could be implemented only in new plants. Hence this topic is considered to have low to medium generic applicability. In new plants the cost of implementing changes in plant layout or component protection would be low to medium.

During the scoping studies, recommendations in this area should be reviewed in greater depth to identify the most promising concepts.

RESEARCH TOPIC 14: PROTECTION AGAINST SABOTAGE

Description

In recent years, concern for the protection of nuclear facilities against sabotage has risen in response to the increased use of terrorist tactics in international politics. During this time significant improvements have been made to reduce the vulnerability of nuclear power plants to sabotage. It should be noted that many of the concepts proposed for improved plant configuration and design are also applicable to protection against sabotage. However, sabotage protection has been retained as a separate research area because it involves some unique design considerations and because of the importance of ensuring that the concepts for improving protection against sabotage receive the level of consideration that is warranted.

Extensive NRC studies involving improvements in safeguards have already been undertaken in a number of areas; these include programs for various fuel cycle facilities, the transportation and export of special nuclear materials, and commercial nuclear power reactors. Some of the studies have been completed and have led to modifications in the regulatory criteria and requirements for commercial nuclear power plants. Other ongoing programs are evaluating additional concepts for the protection of the plant against internal and external threats.

Important confirmatory research efforts are under way to develop methods for evaluating the effectiveness of safeguards systems in the areas of physical protection for facilities, transportation, and materials control and accounting.

Evaluation

The sources listed in Appendix C are nearly unanimous in recommending enhanced protection against sabotage. It is very difficult to evaluate sabotage-related risk to the public because the frequency and targets of terrorist activities have varied so widely in history; hence no value can be assigned to the potential for risk reduction. The generic applicability of concepts for improved sabotage protection is medium, and the cost of implementation is thought to be low to medium.

A scoping study should be undertaken to identify any new improvements that might appear promising.

3.5 SITING AND EMERGENCY RESPONSE

RESEARCH TOPIC 15: NEW SITING CONCEPTS

Description

The potential consequences of a major reactor accident are sensitive to the size of the population that is exposed to radiation. Remote siting of nuclear power plants, offshore siting, and the colocation of facilities in nuclear power parks have been suggested as possible means of reducing public risk. Risk tradeoffs among safety system design, containment capability, and siting seem to be potentially attractive and have been suggested.

To date, regulations for the siting of nuclear power plants have not been based on quantitative methods of risk analysis. The implementation of many of the concepts that have been proposed could require major changes in regulations and the redirection of national policy. For example, many utilities do not have truly remote sites within their service areas.

The evaluation of the potential benefits of new siting concepts would best be performed by risk analysis techniques. Some improvements in the modeling of accident consequences, such as improved meteorological models, might be needed to make such evaluations more meaningful.

A multiyear interoffice program is under way in NRC to review the siting policy and regulations, and to propose needed changes. Ongoing research programs in this area include the improvement of meteorological models and risk studies.

Evaluation

Recommendations for new siting concepts have been directed primarily at the use of more remote sites. The potential contribution to risk reduction is evaluated as medium. In the event of a severe accident, latent health effects are predicted to occur over a large area. Therefore, this component of risk is not particularly sensitive to the size of the nearby population unless the site is extremely remote. By contrast, the early effects of a severe accident, such as early fatalities, would be more affected by siting practices. Since new siting concepts would be applicable only to new plants, their generic applicability is considered to be low to medium, and the associated cost is judged to be medium to low.

Very little new research would be required to evaluate the potential benefits of new siting concepts or to implement the concepts. Policy decisions might be made on the basis of existing data and risk analyses. The present NRC confirmatory research program covering this area will be continued, perhaps with some minor modifications.

RESEARCH TOPIC 16: IMPROVED OFFSITE EMERGENCY RESPONSE PLANNING

Description

Nuclear power plants are required to have emergency response plans that are coordinated with state and local authorities to ensure that, in the event of an emergency, adequate warning is given to the public, instructions are provided to the public for evacuation or other protective measures as necessary, and adequate health services are available. Recommendations related to emergency planning include activities that would protect the public during the initial period of exposure and others that would be undertaken later to reduce long-term exposures.

Current NRC research programs on offsite emergency response planning are covering three general areas. The first is concerned with developing a set of accident scenarios for planning purposes. The results should provide planners with a realistic description of a spectrum of potential accidents for evaluating the suitability of emergency response plans. The second area identifies and evaluates protective measures and strategies that could be taken to mitigate the public consequences of potential radioactive releases. The various strategies that have been evaluated include evacuation, sheltering, medical prophylaxis, and selective relocation. The last area combines the first two in order to evaluate the effectiveness of employing the various protective strategies over the spectrum of accidents. The objective of this work is to gain further insight into the relative merits of implementing one strategy or a combination of strategies.

Evaluation

A number of recommendations have been made for improved offsite emergency response. Risk analyses indicate that the effectiveness of emergency response at the time of the accident, such as evacuation, does not have a major effect on risk. Activities taken after the accident, to relocate individuals or to restrict the consumption of supplies of food and water, could, however, have a major effect on the total doses received by the public. These activities do not require a great deal of detailed preplanning, however. Overall, this topic was judged to have low to medium risk-reduction potential. The cost of implementing improved offsite emergency response plans would be low, and the guidelines for plans would be generally applicable to all reactors.

Many of the types of study that have been recommended are being undertaken in the existing confirmatory research program, which will be continued, perhaps with some minor modifications.

4.0 RECOMMENDED RESEARCH PROGRAM FOR IMPROVING NUCLEAR POWER PLANT SAFETY

4.1 INTRODUCTION

Of the sixteen research projects identified in Chapter 2 and evaluated in Chapter 3, five are recommended for pursuit at this time because they emerged quite clearly as having significant potential for improving the safety of light-water nuclear power plants. Furthermore, to help in selecting research projects for the comprehensive long-range program mandated by Congress, an effort directed at improving the methodology used in value/impact analyses is recommended as part of the research program proposed herein. It is also recommended that scoping studies be conducted on the remaining eleven research topics to determine which topics warrant additional research and to obtain preliminary value/impact analyses of the various concepts proposed for improving the safety of nuclear power plants. The following research projects are recommended for the initial phase of the program:

Research Project A: Alternate Containment Concepts

Research Project B: Alternate Decay Heat Removal Concepts

Research Project C: Alternate Emergency Core Cooling Concepts

Research Project D: Improved In-Plant Accident Response

Research Project E: Advanced Seismic Designs

Research Project F: Improved Methodology for Evaluating Research Topics

Research Project G: Scoping Studies

As mentioned in Chapter 3, six of the remaining eleven topics (nondestructive examination and on-line monitoring, reduced occupational exposure, improved reactor shutdown systems, protection against sabotage, new siting concepts, and improved offsite emergency response planning) are already receiving significant attention in NRC's regulatory process and confirmatory research program; these should be reexamined for completeness as part of the proposed scoping studies. The remaining five topics (improved plant controls, reactor vessel rupture control, core retention measures, equipment for reducing radioactivity releases, and improved plant layout and component protection) should also be covered by scoping studies to determine whether research projects are warranted in NRC's future program for improved-safety research.

It should be noted that most of the research projects proposed herein will require 1 to 2 years for completion. Work on alternate emergency core cooling concepts and advanced seismic designs is anticipated to take 3 years. It is already clear that the development of improved value/impact criteria would involve continuing studies for several years. It is also possible that the proposed scoping studies may identify topics that warrant inclusion in future research programs. These topics, as well as any others that may be suggested or identified, will be covered in next year's report.

The remaining sections of this chapter describe and summarize the research projects recommended for implementation. They also discuss estimated costs and approximate schedules for each project. An overview of the proposed research program is presented in Table 4-1.

4.2 RESEARCH PROJECT A: ALTERNATE CONTAINMENT CONCEPTS (RESEARCH TOPIC 7)

The research proposed here would cover two areas. The first is a specific study of the use of vented containments to reduce the probability of large airborne releases of radioactivity in the event of an accident involving core melting. It is expected that such a study would be performed in sufficient detail to determine its feasibility, to define system performance and safety design requirements, and to complete a value/impact analysis. The second would be a scoping study to develop further information regarding the usefulness of the other alternate containment concepts that have been proposed.

Research on vented containments would cover the following aspects:

- Various conceptual system configurations and associated filter materials to determine feasibility, sizing, and cost.
- The potential reduction in the probability of large radioactive releases, including the resultant impact on overall accident risks.
- Impact on potential accidents that do not involve core melting.
- System performance and safety design requirements.
- A quantitative value/impact analysis.
- Experiments on filter performance if found to be needed.

The scoping study of other containment concepts would be more general than that of vented containments and would involve only enough effort to arrive at preliminary value/impact findings. It would cover such concepts as passive containment systems, containments with larger volumes, higher pressure capabilities, reduced initial operating pressures, and improved capability to deal with the consequences of energetic fuel-water interactions or the rapid burning of hydrogen.[1]

The vented-containment study could be completed in about 2 years at an estimated cost of $600,000. Experimental data on filter materials, if found to be needed, could be generated in 1 year at an estimated cost of $500,000.

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[1]The study of underground containment has been suggested, but its start will be delayed until next year to permit evaluation of the California study on underground siting, soon to be released.

A scoping study of other containment concepts could be performed in about 1 year at a cost of approximately $500,000.

4.3 RESEARCH PROJECT B: ALTERNATE DECAY HEAT REMOVAL CONCEPTS (RESEARCH TOPIC 6)

The research proposed here would cover two areas. The first is a specific study related to the usefulness of installing an additional, or add-on, decay heat removal system in existing nuclear power plants to improve the overall operational reliability of decay heat removal. Such a study would entail reviewing the detailed design of a decay heat removal system in order to assess the potential improvement in reliability.[1] The study would also produce suggested system performance and safety design criteria as well as a value/impact analysis. The second area would involve a scoping study to develop further information regarding the usefulness of other alternate concepts proposed for decay heat removal systems.

Add-On Decay Heat Removal System. The study of an add-on decay heat removal system would include a reliability assessment of the overall decay heat removal system of a plant or plants in which an add-on decay heat removal system had been installed. In designing an add-on system for high reliability, consideration would be given to such features as maximum independence of existing systems to reduce the potential for common-mode failures, including the use of independent power and water supplies as well as an independent heat sink. Other factors to be considered include the following:

- Increased separation of, and improved protection for, existing decay heat removal systems to improve reliability for both hot- and cold-shutdown phases of operation.
- Use of redundancy, separation, and fire protection techniques, independent electric power and water supplies, and independent heat sinks.
- The impact on risk-reduction potential of design features such as bunkering to protect the system from earthquakes, severe-weather phenomena, sabotage, and accidents.

Other areas to be covered in this research project are system performance and safety design requirements as well as a quantitative value/impact analysis.

Alternate Decay Heat Removal Concepts. The study on other alternate decay heat removal concepts would be scoping in nature and would involve only enough effort to arrive at preliminary value/impact findings. It would cover such concepts as improved systems that permit the nuclear power plant to remain operational in the event of a turbine trip, improved backup feedwater systems, passive decay heat removal systems, and additional ultimate heat sinks such as deep wells.

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[1] It is proposed that the actual system design be performed under the auspices of the U.S. Department of Energy to meet objectives stated by NRC.

The study of an add-on heat removal system could be completed in about 2 years at an estimated cost of approximately \$600,000. Additional funds would be expended by the U.S. Department of Energy for the detailed design of one or more add-on systems.

The scoping study of alternate decay heat removal concepts could be completed in about 1 year at a cost of about \$600,000.

4.4 RESEARCH PROJECT C: ALTERNATE EMERGENCY CORE COOLING CONCEPTS (RESEARCH TOPIC 5)

The proposed research would develop experimental and analytical data for determining whether significant improvements could be made in the performance of emergency core cooling (ECC) systems. The recommended program is intended to augment, on an accelerated schedule, present confirmatory research plans by delineating and evaluating ECC systems with significant potential for improved core-cooling capability, less complex analytical descriptions, and broad applicability. Existing computer codes would be modified as needed for analyses of improved ECC concepts, and a variety of experiments would be performed to test some of the suggested concepts.

The study on alternate ECC concepts would include:

- Modifying the latest versions of the appropriate best-estimate computer codes as needed to model alternate ECC concepts.
- Using these codes to perform a variety of calculations for various BWR and PWR system designs involving alternate ECC systems. Suggested improvements include alternate locations for fluid injection, devices to divert or restrict fluid flow, and increasing the volume or pressure of the available fluid. A search would also be undertaken for other improved ECC system configurations.
- Accelerating the schedule of the Semiscale facility at the Idaho National Engineering Laboratory to permit experimental testing of selected alternate ECC system configurations.
- Performing preliminary value/impact evaluations of promising concepts for alternate ECC system configurations.

It is expected that computer code and model updating could be performed in about 2 years at a cost of approximately \$400,000. This work could, to some extent, proceed in parallel with experimental design work at the Semiscale facility.

The analyses of various alternate ECC concepts would take about 3 years and are estimated to cost approximately \$1.5 million. Some of this work could be performed concurrently with the experiments.

The acceleration of the design, procurement, and fabrication of the Semiscale facility modifications would require about 1 year and cost approximately \$2 million.

4.5 PROJECT D: IMPROVED IN-PLANT ACCIDENT RESPONSE (RESEARCH TOPIC 3)

The proposed project would review studies completed or in progress on the following topics in order to establish the need for additional research:

- Human errors in testing and maintenance.
- Monitoring and diagnostic systems to assist the operator under accident conditions.
- Operating and emergency procedures for responding to accident situations.
- Improved use of simulators in studying operator response to accident situations and for related training.
- Man/machine interface, information presentation, pattern recognition, control-room design, and automatic controls for safety systems.
- Human initiation of accidents.

The initial study is estimated to take 1 year and cost approximately $600,000. Studies of automatic monitoring and diagnostic systems could be performed in 2 years at a cost of $1,000,000.

4.6 RESEARCH PROJECT E: ADVANCED SEISMIC DESIGNS (RESEARCH TOPIC 12)

This research project would entail a study of various concepts for improved seismic resistance. The suggestions listed in Chapter 3 and any others that may be appropriate would be reviewed to determine feasibility, risk-reduction potential, cost, and other relevant value/impact components. These studies would be closely coordinated with the ongoing confirmatory research programs on seismic design and seismic risk. In future years, research may be appropriate on one or more concepts that are shown in this project to be promising candidates for application.

The study on advanced seismic designs would include:

- Preliminary value/impact analyses of the following concepts suggested in Chapter 3 and others as appropriate:
  - Increased energy-absorption capability
  - Component isolation from seismic motions
  - Isolation of the foundation
  - Flotation in a fluid-filled basin
  - Testing of seismic resistance by means of large shaker tables, shaker devices attached to complex structures, and explosives.

- Increasing the peak-acceleration values and/or extending the spectrum shape used as bases for seismic design

- Preliminary definition of the design requirements of candidate concepts.

- Development of information for evaluating the risk-reduction potential, including a probabilistic assessment of the behavior of the plant and its components under earthquake conditions.

- Improving analytical models and conducting appropriate experiments pertinent to these evaluations.

The scoping, value/impact, and research definition studies could be completed in about 1 year at a cost of approximately $600,000.

Concurrent work on the initial system design requirements and model development (including probabilistic studies of the behavior of components and plants) could probably be completed in 3 years at a cost of approximately $3 million.

The experimental program would be defined after the foregoing work has been completed.

4.7 RESEARCH PROJECT F: IMPROVED METHODOLOGY FOR EVALUATING RESEARCH TOPICS

The scope of work proposed for this project would be directed toward developing more objective and precise methods for making value/impact assessments. The tasks involved include the following:

- Development of methods for arriving at a reasonably quantitative way of evaluating the risk-reduction potential of proposed concepts to improve the safety of nuclear power plants as well as the overall change in risk level that would be achieved by proposed changes if research results were to be implemented.

- Development of methods for combining various factors such as breadth of technical support, risk-reduction potential, generic applicability, and cost of implementation in a more quantitative formulation of value (risk reduction, breadth of application, timeliness) and impact (increases in plant complexity, effect on other systems and functions, cost).

The program effort to develop evaluation methodology is very important in planning the future safety research program and, as such, is foreseen to be a continuing one. The level of effort proposed for the first year would cost approximately $500,000.

4.8 RESEARCH PROJECT G: SCOPING STUDIES

The evaluations described in Chapter 3 identified eleven research topics that warrant scoping studies to identify new concepts worthy of research support, to determine which topics covered by current NRC programs require additional research, and to obtain preliminary value/impact analyses of the various alternatives proposed for improved safety.

The scoping studies would include a review and evaluation of the following research topics, identified by the numbers used in Chapters 2 and 3:

1. Nondestructive examination and on-line monitoring systems (to identify new concepts worthy of support).

2. Improved plant controls (to determine whether additional research is required on advanced control systems or other safety controls).

4. Reduced occupational exposure.

8. Improved reactor shutdown systems.

9. Reactor vessel rupture control (to scope the requirements and the potential for improving reactor safety).

10. Core retention measures (to determine whether any significant control of airborne radioactivity is possible).

11. Equipment for reducing radioactivity releases under both normal and abnormal conditions.

13. Improved plant layout and component protection (to determine whether additional safety protection can be achieved).

14. Protection against sabotage (to determine whether the current confirmatory research program is adequate).

15. New siting concepts (to determine whether safety improvements can be effected through new siting schemes).

16. Improved offsite emergency response planning (to determine whether significant risk reduction can be achieved in this manner).

Six of the research topics listed above (Nos. 1, 4, 8, 14, 15, and 16) are covered by current NRC programs, which may require some modification as a result of the scoping studies proposed herein.

The scoping studies would take 1 to 2 years at an estimated cost of $1,500,000. It is expected that, in future years, an ongoing effort in this area would be pursued at a much lower level of effort and cost.

4.9 CONCLUSION

The NRC staff believes, and the Advisory Committee on Reactor Safeguards concurs, that the research projects described herein will serve to place in better perspective the extent and suitability of potential improvements in the safety of light-water nuclear power plants. It is recommended that these studies be undertaken even though their overall risk-reduction potentials are not fully known. Furthermore, the scoping and methodology-development studies described in this plan should also be performed to provide the basis for a longer term effort in this area.

Table 4-1 contains a list of the recommended research projects along with estimates of funding, schedules, and personnel resources needed to carry them out. The total resources required would be $14.9 million and eight professional staff members. The cost estimates for the program described in this report cover a 3-year period, with $7.5 million for the first year, $4.9 million for the second year, and $2.5 million for the third year. As a result of scoping studies and other efforts recommended herein it is likely that additional resources will be needed in the future.

TABLE 4-1

SUMMARY OF RECOMMENDED RESEARCH PROJECTS

	Research Project	Scope	Program Support (millions of dollars)	Duration (years)
A.	Alternate Containment Concepts	Vented-containment studies	0.6	2
		Other concepts	0.5	1
			1.1	
B.	Alternate Decay Heat Removal Concepts	Add-on system	0.6	2
		Alternate concepts	0.6	1
			1.2	
C.	Alternate Emergency Core Cooling Concepts	Code modification	0.4	2
		Analyses	1.5	3
		Semiscale upgrade	2.0	1
			3.9	
D.	Improved In-Plant Accident Response	Review	0.6	1
		Automatic monitoring and diagnostic system	1.0	2
			1.6	
E.	Advanced Seismic Designs	Preliminary evaluation	0.6	1
		Definition and modeling	3.0	3
			3.6	
F.	Improved Methodology for Evaluating Research Topics	Model development	0.5	1[a]
G.	Scoping Studies	Review of research topics	1.5	1 to 2
		SUBTOTAL	13.4	1 to 3
		Eight additional staff members	1.5[b]	
		TOTAL	14.9[c]	

[a]This would involve continuing studies for several years.

[b]Covers a 3-year period.

[c]Approximately $7.5 million in the first year, $4.9 million in the second year, and $2.5 million in the third year.

GLOSSARY

This glossary provides brief and informal explanations of selected terms used in the main text of the report to aid the nontechnically oriented reader.

ACRS	See "Advisory Committee on Reactor Safeguards."
ALARA	An objective of design and operation to limit releases of radioactivity to levels that are as low as reasonably achievable (ALARA).
APS	American Physical Society.
accident-initiating event	An occurrence (e.g., a pipe break) that has the potential to cause an accident if additional equipment provided to cope with such events also were to fail.
accident sequence	The order of events in a postulated or real accident leading to some particular outcome (e.g., release of radioactive materials).
Advisory Committee on Reactor Safeguards	An independent group established by law to advise the Nuclear Regulatory Commission on its regulatory activities, principally those related to nuclear safety.
BWR	See "boiling water reactor."
backfitting	A process of making changes to plants that are already designed or built.
best estimate codes	Computer codes employing the currently available state of the art in realistic modeling of physical processes.
boiling water reactor	A nuclear heat source cooled by boiling ordinary water, the steam being used directly in a turbine generator to produce electricity.
bunker	As used in this report, the term "bunker" refers to a structure or special protective hardware designed to resist the effects of severe external events such as fire, earthquakes, or sabotage.

cold shutdown	The condition in which a nuclear plant is taken down to essentially ambient temperature and pressure with no power (except decay heat) being produced.
common-mode failure	Failure of two or more components or systems caused by the same event, environment, or defect.
confirmatory research	Research needed to provide a basis for evaluating applications for regulatory decisions, or to provide a basis for regulatory requirements or policy, or to provide NRC with the physical or judgmental capability to regulate the use of nuclear power and materials.
containment building	A structure designed to inhibit the release of radioactive material to the environment in the event of potential reactor accidents.
coolant	A fluid used to remove heat. In the context of this report, the coolant, which is relatively pure water, is used to remove the heat generated in the core of the reactor.
core	The region of a nuclear reactor in which the controlled nuclear fission process occurs and in which most of the energy of fission is released as heat. The core consists of the fuel, fuel cladding, control rods (special rods designed to control and stop the fission process), various core support structures, and coolant.
core melt	Melting of components of the core (principally the fuel rods) as a result of insufficient cooling.
core retention measures	Concepts that provide for holding within the containment building the materials from a reactor core in the event of a core meltdown.
DOE	U.S. Department of Energy.
decay heat removal system	A system that provides long-term cooling for the core after plant operation is stopped but while heat is still being produced by the radioactive decay of fission products in the core. Such systems are used in the course of routine plant operation.

design-basis accident (DBA)	A defined accident used as an assumption or basis for the design of various plant engineered safety features.
developmental research	Research conducted to evaluate the safety of materials, processes, and equipment likely to be proposed by an applicant for an NRC license.
ECC	Emergency core cooling or coolant.
ECCS	See "emergency core cooling system(s)."
EPRI	See "Electric Power Research Institute."
Electric Power Research Institute	An organization founded in 1972 by U.S. electric utilities to develop and manage a technology program for improving electric power production, distribution, and utilization.
emergency core cooling system(s)	Special safety system(s) designed to cool a reactor core primarily in the event of a loss-of-coolant accident.
engineered safety feature	A system or feature of a nuclear power plant that has been designed to perform a specific safety function; in particular, to control a design-basis accident or mitigate its consequences.
fracture mechanics	The study of crack propagation and arrest in materials such as steel.
heat sink	A place for disposing of the unused heat within the nuclear power plant to the environment. This place may be a river, well, or pond; sometimes referred to as the "ultimate heat sink."
hot shutdown	The condition in which a nuclear plant is not operating, but the coolant is still at approximately operating temperature and pressure.
improved safety research	See "research for improved safety."
LOCA	See "loss-of-coolant accident."
LWR	See "light-water reactor."

light-water reactor	A generic term for both pressurized water reactors and boiling water reactors, which are cooled and moderated by ordinary (i.e., "light") water.
loss-of-coolant accident	Accident in which a break is postulated to occur in a reactor pipe, thus causing a loss of the high-temperature, high-pressure water that normally cools the core.
NDE	See "nondestructive examination."
NRC	U.S. Nuclear Regulatory Commission.
nondestructive examination	A means of analyzing a system or component without altering its configuration or composition.
occupational exposure	Radiation dose received by a person assigned to work in a radiation environment. Usually such exposure comes about through maintenance, testing, and inspections routinely performed during the lifetime of the plant.
PWR	See "pressurized water reactor."
primary system	A general term encompassing the reactor pressure vessel, piping, pumps, and heat exchangers in which the normal coolant is circulated.
pressurized water reactor	A nuclear heat source cooled by water under pressure; the heated water is used to boil water in a secondary system for driving the turbine generator to produce electricity.
radionuclide	Radioactive atomic form of an element.
reactor core	See "core."
Reactor Safety Study (RSS)	A detailed probabilistic study published by the U.S. Nuclear Regulatory Commission in October 1975 under the title <u>Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants</u> (report numbers WASH-1400 and NUREG-75/014).

reactor shutdown system	A means (such as neutron-absorbing control rods) for stopping the nuclear fissioning in the core. Backup shutdown systems also usually exist, and these contain concentrations of neutron-absorbing material in a liquid form.
reactor vessel	A steel pressure vessel containing the reactor core and associated structures. The reactor vessel provides the pressure boundary for containing and directing the reactor coolant to and from the core.
redundan	A term describing the independent duplication of functions.
regulatory research	Originally this term referred to the confirmatory research conducted by NRC; as described in this report, regulatory research may also include research for improved safety.
research for improved safety	Research on advanced concepts, systems, and processes believed to have potential for improving the safety of nuclear power plants.
routine release of radioactivity	Discharge of radioactive effluents during normal operation with appropriate filtering or retention to ensure that the effluent meets NRC's ALARA criteria.
seismic design	The design of a plant to withstand earthquakes or tremors that have or may be experienced in the region where the plant is located.
Semiscale	A small-scale nonnuclear test facility simulating a pressurized water reactor and used to provide data for the development and/or verification of computer models of engineered safety features.
transient	A disturbance or change in a nuclear power plant which causes variations with time of such plant parameters as power, pressure, flow, etc.
ultimate heat sink	See "heat sink."

value/impact analysis	A technique for numerically weighing the usefulness of some change against its possible negative effects.
vented containment	A containment building with a system designed to control the pressure in the containment and the release of gases with suitable filtering of the radioactivity in those gases.

APPENDIX A

REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS ON THE PROPOSED NRC RESEARCH PROGRAM

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 13, 1978

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: PROPOSED RESEARCH ON SYSTEMS TO IMPROVE SAFETY

Dear Dr. Hendrie:

The Advisory Committee on Reactor Safeguards has reviewed the proposed research program on systems to improve the safety of nuclear power plants, as embodied in the draft report, "Report to the U.S. Congress on NRC Plans for Research Directed Toward Improving the Safety of Light-Water Nuclear Power Plants," dated March 3, 1978. This report was reviewed by the full Committee at its 215th meeting, March 9-10, 1978. The proposed program was reviewed by a Subcommittee at a meeting on February 23, 1978. In addition, members of the Committee Staff attended the meetings of the NRC Research Review Group on January 10 and February 10, 1978.

The proposed program has been developed in response to the requirement by Congress in the FY 78 Budget Authorization Act for the NRC. Although the pertinent section of the Act bears the subheading, "Improved Safety System Research," the wording of new subsection (f) refers to "...projects for the development of new or improved safety systems..." The NRC Staff has recognized, and pointed out in its report, that the requirement for "development," if interpreted literally, could compromise the position of the NRC as an impartial judge of safety systems incorporated into nuclear plants. The NRC Staff has proposed, therefore, that its program be limited chiefly to the evaluation of new concepts for improving reactor safety. The Committee agrees with this approach. In its recent report to the Congress (NUREG-0392), the Committee stated:

> "...The ACRS believes that the development, testing, and proof of efficacy of new or improved safety systems should not be the responsibility of the NRC, but should be conducted by the nuclear industry or DOE. However, the ACRS believes that it is a proper and even necessary function of the NRC to perform or sponsor research on concepts that, if developed and implemented by the appropriate bodies, could lead to improvements in safety."

The NRC Staff has recommended five research projects as having the greatest prospect of leading to improved safety. They are:

A. Alternate containment concepts, especially vented containments.

B. Alternate decay heat removal systems, especially bunkered systems.

C. Alternate ECCS concepts.

D. Improved accident response.

E. Advanced seismic design.

The Committee concurs in these choices and believes that these studies should be undertaken even though their risk reduction potentials are not yet clearly known. These studies and the follow-on programs will serve to place in perspective the extent and suitability of possible safety improvements.

The NRC Staff has stated in its report that most of these research projects will require only one to two years for completion, the possible exceptions being Projects A and E. Although these five projects in themselves would not appear to represent the sort of "long-term plan" requested by the Congress, the NRC Staff has proposed that two additional programs be undertaken, as follows:

F. Improvement of the methodology for evaluating research topics and alternate plant designs.

G. Scoping studies of the eleven additional research topics that have been suggested.

These programs can be expected to provide a basis for a longer term effort.

The Committee believes that Project F on the development of better methods for evaluating concepts proposed to improve safety is essential to the success of this new effort. Although there will always be a large subjective or judgmental element in the selection of research projects on improved safety, these selections should be made on as quantitative and factual a basis as practical. It seems evident also that it will be extremely difficult to provide a suitable methodology without at some point addressing the question of how safe is safe enough.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

APPENDIX B

MEMBERS OF AND CONSULTANTS TO THE NRC RESEARCH REVIEW GROUP

APPENDIX B

MEMBERS OF AND CONSULTANTS TO NRC RESEARCH REVIEW GROUP

The principal inputs to the development of the research program described in this report came from the NRC Research Review Group on Improved Reactor Safety. The Research Review Group was organized specifically to consider NRC's amended charter for conducting research on improved reactor safety and consists of members representing NRC's Offices of Nuclear Regulatory Research, Nuclear Reactor Regulation, and Standards Development.

Suggestions and criticisms were obtained from the consultants listed on the next page, who represented various points of view, including those of national laboratories, universities, public interest groups, industry, NRC research contractors, and the U.S. Department of Energy.

Members of NRC Research Review Group

Chairman

Mr. Saul Levine, Director
Office of Nuclear Regulatory Research (RES)

Members

Dr. Gary L. Bennett, Chief
Research Support Branch
Water Reactor Safety Research (WRSR)

Dr. Stephen H. Hanauer, Technical
Advisor, Office of the Executive
Director for Operations

Dr. Thomas E. Murley, Director
Division of Reactor Safety
Research (RSR)

Mr. James A. Norberg
Senior Reactor Engineer
Engineering Methodology Branch
Office of Standards Development

Mr. L. S. Rubenstein, Chief
Research Analysis Section
Program Support Branch
Office of Nuclear Reactor Regulation

Mr. Ronald M. Scroggins, Chief
Systems Engineering Branch
Water Reactor Safety Research (WRSR)

Mr. M. A. Taylor
Senior Reactor Safety Engineer
Probabilistic Analysis Staff, RES

Dr. L. S. Tong, Assistant Director
for Water Reactor Safety
Research (RSR)
Chief Scientist, RES

Consultants

Mr. Eric S. Beckjord, Acting Director
Division of Nuclear Power Development
U.S. Department of Energy

Mr. Dale G. Bridenbaugh
MHB Technical Associates

Dr. George F. Brockett
Vice President
Intermountain Technologies, Inc.

Dr. William D. Corcoran, Director
Performance Analysis
Combustion Engineering, Inc.

Dr. Richard S. Denning, Manager
Nuclear & Flow Systems Section
Battelle Columbus Laboratories

Dr. David E. Dorfan
Associate Professor of Physics
University of California at Santa Cruz

Dr. Fred C. Finlayson, Manager
Nuclear Projects Energy Systems Group
The Aerospace Corporation

Mr. Carl J. Hocevar
Energy Engineering Group, Inc.

Dr. Herbert J. C. Kouts, Chairman
Department of Nuclear Energy
Brookhaven National Laboratory

Mr. Milton Levenson
Director for Nuclear Power
Electric Power Research Institute

Dr. S. Levy
S. Levy, Inc.

Mr. David McCloskey, Manager
Nuclear Fuel Cycle Safety Research Department
Sandia Laboratories

Dr. Nathan Newmark
Professor Emeritus of Civil Engineering
University of Illinois

Mr. Warren Owen
Vice President, Design Engineering
Duke Power Company

Mr. Andrew J. Pressesky
Acting Assistant Director for Safety and Quality Assurance
Division of Nuclear Power Development
U.S. Department of Energy

Dr. Donald H. Roy, Manager
Plant Design Section
Babcock & Wilcox Company

Dr. Romano Salvatori, Manager
Projects Development
Westinghouse Electric Corporation

Dr. Glen G. Sherwood, Manager
Safety and Licensing Operation
General Electric Company

Mr. John E. Ward, Chairman
Committee on Reactor Licensing and Safety
Atomic Industrial Forum, Inc.

Mr. James O. Zane, Deputy Director
Water Reactor Research Directorate
EG&G Idaho, Inc.

APPENDIX C

REVIEW OF SUGGESTIONS ON IMPROVING THE SAFETY OF LIGHT-WATER NUCLEAR POWER PLANTS

APPENDIX C

REVIEW OF SUGGESTIONS ON IMPROVING THE SAFETY OF LIGHT-WATER NUCLEAR POWER PLANTS

In order to identify concepts and research projects for improved reactor safety, the NRC Research Review Group reviewed a number of books, reports, and letters in which recommendations have been presented for improving the safety of nuclear power plants. The principal documents that have been reviewed are reports[1] by the Advisory Committee on Reactor Safeguards, the ECCS Acceptance Criteria,[2] the Report to the American Physical Society by the Study Group on Light-Water Reactor Safety,[3] Assessment of ECCS Effectiveness for Light-Water Nuclear Power Reactors,[4] and the Ford Foundation study Nuclear Power - Issues and Choices.[5] Recommendations from these references are abstracted in this appendix (see Table C-1). In addition, suggestions were solicited from the NRC regulatory staff and from a number of consultants representing the views of national laboratories, universities, public interest groups, the U.S. Department of Energy, NRC research contractors, and the nuclear industry (see Appendix B).

Some of the recommendations that have been made were judged to fall outside the scope of concepts for improved safety. A few of these were procedural recommendations on how to identify or evaluate important concepts. Other suggestions pertained to confirmatory research. In particular, a number of suggestions were directed at a better understanding of the safety margins of current reactor designs. These suggestions have been included in this appendix but were not used in developing the categorization of concepts for improved safety that is discussed in Chapter 3. All recommendations received are listed in Table C-1, which also indicates the manner in which they have been addressed.

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[1]Including the recent Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program - A Report to the Congress of the United States of America, NUREG-0392, December 1977.

[2]Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, U.S. Atomic Energy Commission, Washington, D.C., Docket No. RM-50-1, December 1973. This rulemaking hearing record includes recommendations by the Consolidated National Intervenors, which included, among others, the Union of Concerned Scientists.

[3]Published in Reviews of Modern Physics, Vol. 47, Supplement No. 1, Summer 1975.

[4]F. C. Finlayson, Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors, EQL Report No. 9, Environmental Quality Laboratory, California Institute of Technology, May 1975.

[5]Published as Nuclear Power - Issues and Choices, Ballinger Publishing Company, Cambridge, Mass., 1977.

C.1 RECOMMENDATIONS BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (ACRS) has issued a series of letter reports summarizing its comments on the NRC's Water Reactor Safety Research Program. These reports are issued on a regular basis and have suggested areas of research that could lead to improved reactor safety. As the Safety Research Program has evolved, the Committee has recognized the progress made and has reflected this progress in its comments and recommendations.

Generic safety issues on unresolved safety questions have been identified annually by the ACRS. These are issues for which the NRC staff "has made a determination that the safety significance of the issue does not prohibit continued operation or licensing actions while the longer term generic review is under way." In a recent report,[1] the NRC has outlined a program for the resolution of these generic safety issues. This program has grouped the generic technical activities into priority categories based on a set of uniform criteria. Detailed task action plans discussing the generic issues have been developed for items with the highest priority. These plans define the problem, describe the staff's approach to its resolution, and provide manpower, funding, and schedule estimates for completing the task. These generic or unresolved safety issues are not reviewed in this appendix because they will be included in the confirmatory research program. Plans for their resolution are discussed in NUREG-0410.

Another source of recommendations was the ACRS report entitled Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program--A Report to the Congress of the United States of America (NUREG-0392, December 1977). This report reviews and evaluates the NRC safety research program. In addition, it makes recommendations for additional confirmatory research in certain areas and for research that has the potential for leading to the development of improved-safety concepts.

C.2 RECOMMENDATIONS BY NRC STAFF MEMBERS

Members of the NRC research, licensing, and standards staffs were requested to submit recommendations for research related to the improvement of reactor safety. They responded with numerous suggestions, which are listed in Table C-1.

C.3 RECOMMENDATIONS BY CONSULTANTS

Recommendations for improved safety concepts and safety research were solicited from a group of consultants representing the views of national laboratories, universities, public interest groups, NRC research contractors, the nuclear industry, and the U.S. Department of Energy. The consultants, who are listed in Appendix B, submitted specific recommendations at open meetings or in subsequent letters.

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[1]U.S. Nuclear Regulatory Commission, NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants, Report to Congress, NUREG-0410, January 1, 1978.

C.4 REPORT OF THE APS STUDY GROUP ON LIGHT-WATER REACTOR SAFETY

The American Physical Society's (APS) Study Group on Light-Water Reactor Safety examined the safety issues related to the accidental release of radioactivity from commercial light-water reactors. The report of the APS Study Group provides a technical analysis that identifies some of the critical technical matters and reviews the safety research and development program.

C.5 FORD FOUNDATION STUDY

The report of the Nuclear Energy Policy Study Group, sponsored by the Ford Foundation, was published in 1977 under the title Nuclear Power - Issues and Choices. Its objective was an examination of the issues involved in the debate on nuclear power in the United States and abroad. The study participants sought to develop a framework for assessing the difficult problems related to nuclear power that are now before the U.S. Government. Many questions were considered in assessing the role of nuclear power, and several observations and recommendations concerning improved reactor safety are made.

C.6 ATOMIC ENERGY COMMISSION - ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS

In this document, the U.S. Atomic Energy Commission announced its decision in the rulemaking hearing concerning acceptance criteria for emergency core cooling systems for light-water nuclear power reactors. The document reviews the history of the rulemaking hearing, explains the principal reasons underlying the key elements of the decision, and summarizes the technical context of the issues presented. The changes in the interim acceptance criteria are identified and the reasons for them are discussed in detail. The recommendations for research contained in this document are related principally to the emergency core cooling system.

C.7 ENVIRONMENTAL QUALITY LABORATORY, CALIFORNIA INSTITUTE OF TECHNOLOGY (EQL REPORT NO. 9)

The objective of this report was to assess one aspect of the safety of light-water reactors: the functional effectiveness of the emergency core cooling system (ECCS). The principal problem areas are defined, and the options available for at least partially resolving some of the apparently unresolved issues are evaluated. The report also attempts to weigh the technical evidence presented directly or indirectly in connection with the rulemaking hearings on acceptance criteria for emergency core cooling systems. It recommends accelerated research needed to provide a quantitative basis for assessing the margin of safety in ECCS operation and identifies design concepts for improving the coolability of light-water reactors.

TABLE C-1

DISPOSITION OF COMMENTS, WITH CROSS REFERENCES TO CHAPTERS 3 AND 4

This table lists the suggestions received or identified in the sources discussed on the preceding pages and indicates their disposition, with cross references to Chapters 3 and 4 and Appendix D. As described in Chapter 1, safety research may be either confirmatory, developmental, or related to improvements. In some cases there is an overlap in these categories. Some suggestions fall into the area of licensing or technical assistance work related to licensing activities. Only those suggestions that were clearly related to research for improved safety were factored into the evaluation leading to the definition of research topics. The following key is used to describe what research (or licensing) category a particular suggestion may fall into:

CR - Confirmatory research

IR - Improvement-oriented research

DR - Developmental research

L - Licensing (e.g., technical assistance, promulgation of regulations, standards, guides, etc.)

Research topics are identified by number (see Table 3-1)
Research projects are identified as A, B, C, D, E, or F (see Chapter 4)

		Disposition	
Research Suggestion and Source	Suggestion Category	Research Topic and Project	Other Disposition
ACRS Report - 11/24/65			
Reactor vessel rupture control	IR	9	
ACRS Report - 4/14/67			
Fuel failure, fuel distortion effects of ECCS partial malfunction	CR		App. D
Fuel-coolant interactions, meltdown	CR		App. D
Large-scale melting	IR	7-A	
ACRS Report - 1/17/68			
Seismic research	IR, CR	12-E	App. D

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
ACRS Report - 3/20/69			
Reactor designs to deal with meltdown	IR	7-A, 10	
Adequate core cooling	IR	5-C	
Verification of scaling extrapolation of ECCS	CR		App. D
Effect of water chemistry on residual heat removal reliability in long term	CR		
Antiseismic design	IR	12-E	
Containment of molten core	IR	10	
Reduction of routine releases	IR	11	
ACRS Report - 1/11/71			
Core retention and meltdown phenomena	IR, CR	10	App. D
ACRS Report - 5/13/71			
Seismic studies of eastern U.S.	CR		App. D
ACRS Report - 2/10/72			
Improved ECCS	IR	5-C	
ACRS Report - 4/18/73			
Turbine missiles	CR, IR	13	App. D
ACRS Report - 11/20/74			
Improved ECCS	IR	5-C	

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
ACRS Report - 11/20/74 (cont'd)			
Reiteration of meltdown recommendations	IR	10	
Improved means of nondestructive examination	IR, CR	1	App. D
Control of reactor pressure vessel rupture	IR	9	
Assessment of ECCS reliability	CR		App. D
Better understanding of LOCA/ECCS	CR		App. D
ACRS Report - NUREG-0392			
Decay heat removal	IR	6-B	
Alternate ECCS concepts	IR	5-C	
Seismic research in Pacific Northwest	CR		New program started
Liquid pathway study	L		
Emergency response plans	IR	16	
Probabilistic studies of earthquakes	CR, IR	12-E	NRC programs under way
Retention of molten cores	IR	7-A, 10	
New concepts in siting	IR	15	
Development of ALARA criteria for plant personnel exposure and whole fuel cycle	L		NRC Standards
Decommissioning research	L, DR		NRC Standard issued
Health effects of alternate fuel cycles	CR, L		Some work under way (App. D)

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
ACRS Report - NUREG-0392 (cont'd)			
Design concepts to make sabotage more difficult	CR, IR	14	
Alternate locations of irradiated fuel storage pool	L		
Bunkered dedicated shutdown system; physical separation of redundant safety-related facilities	IR	6-B	
ACRS Working Group Meeting - 2/2/78			
Development of reliability criteria that are more realistic than single-failure criteria	L		
Use of reliability techniques in the licensing process	L		Some work under way
Evaluation of improvement in seismic safety obtained by simply designing for an even higher earthquake loading	IR	12-E	
Use of large shakers in seismic design and qualification of safety equipment	IR	12-E	
Improved ECCS that addresses both small and large breaks, recognizing the different concerns in each size: small breaks - improved reliability of ECCS; large breaks - improved functionability of ECCS	IR	5-C	
Value/impact methodology should include long-range task to develop complete systems model for evaluating the overall safety improvement to be expected from a suggested safety item	IR	F	

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
ACRS Meeting - 3/9/78			
Loss of critical operating dynamic functions	CR		Will include in risk assessment
Quantitative value/impact model should include time of implementation and its effect on total risk	IR	F	
Address: Can one vent a vented containment? (Need to know distribution of radionuclides)	IR	7-A	
Effects of vented system on different types of containment	IR	7-A	
Effect of proposed alternate containment systems on performance of other safety systems	IR	7-A	
Method of selecting reactor operators	IR	3-D	
NRC Staff Members			
Improved surveillance capability for pipe leaks or breaks and incipient failures	IR	1	
Improved nondestructive examination techniques	CR, IR	1	App. D
Automatic control systems, including error-correcting systems	IR	2	
Broad study of noise analysis and reactor control	IR	1	
Improved contamination control	IR	4	
Improved diagnostic tools or display for accidents	IR	3-D	
Improved operator training on accident prevention and handling	IR	3-D	

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
NRC Staff Members (cont'd)			
Post-accident instructions	IR	3-D	
Man-machine interactions	IR	3-D	
Improved reliability of all valves	L		As needed
Improved reliability of electric relays and trip signal measurements	L		As needed
Containment vent valves	IR	7-A	
Alternate ECCS concepts	IR	5-C	
Ultimate heat removal capability	IR	6-B	
Methods to control hydrogen offgas	DR		Industrial safety
Bunkered decay heat removal systems	IR	6-B	
Reduced susceptibility to sabotage	IR, CR	14	Confirmatory research under way
Improved material and fabrication techniques	DR		Vendors
Ways to isolate plants	IR	15	
Seismic early warning systems	IR		As needed
Alternate siting and containment concepts	IR	7-A, 15	
Systems analysis of design/fabrication	L		Inspection and Enforcement
Improved availability of safety systems	L, CR		Work under way (e.g., FRANTIC code)
Full-scale core spray tests	CR		Industry
Reduced pellet-cladding interactions	DR		App. D

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
NRC Staff Members (cont'd)			
Reduced operational personnel exposures; use of respirators	IR, CR	4	To be considered in ongoing programs
Evaluation of need for requiring more equipment to be qualified in Seismic Category I	L		
Testing of fuel failure detectors	CR	1	Some work under way (App. D)
Development of process for qualification testing of safety equipment	DR		Demonstration of performance required by NRC
Use of water wells as ultimate heat sink	IR	6-B	
Evaluation of seismic decoupling concepts	IR	12-E	
Development of consistent earthquake criteria	L		
Evaluation of dual containment systems	IR	7-A	
Use of more realistic source terms	L		
Dry containments for BWRs	IR	7-A	
Evaluation of CANDU containment	IR	7-A	
More accurate data on failure rates	CR		Data collection under way (App. D)
Dedicated decay heat removal systems	IR	6-B	
Effect of surface hardening treatment for stud bolts	DR		Industry

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
NRC Staff Members (cont'd)			
Hazard of using chemical ice-melting materials around a nuclear plant	L		Need for research not evident
Multipurpose facility for seismic testing	IR	12-E (if needed)	
Methods for quantitative prediction of failure rates	CR		App. D
Reduced radioactivity release on secondary side of PWRs	IR, L	11	
Special materials to reduce potential for hydrogen offgas explosions	DR		Industrial safety
Investigation of vanadium metal powder for rapid atmosphere inerting (subset of above suggestion)	DR		Industrial safety
Energy-absorbing materials to protect against turbine missiles	IR	13	
Improved steam-generator designs	IR, DR	11	
Decommissioning procedures	L		NRC standard issued
Improved BWR and PWR water chemistry	DR		Industry
Improved electric transmission capabilities	DR		
Improved emergency electric power supplies	IR	6-B (partly)	
Improved stability and reliability of electric transmission grids	DR		
Individual Consultants			
Data on actual radioactive releases	L		Normal reporting practice

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
Individual Consultants (cont'd)			
Siting research	IR	15	
Data for improved weather models	CR, L		To be studied further by NRC
Analysis of LOCAs from realistic pipe break times, disturbed initial conditions, and steady-state operating conditions	CR		Vendors have done some work
Routine emissions and alleviation	IR	11	
Vented-containment concepts	IR	7-A	
Improved existing ECCS	IR	5-C	
Evaluation of older plants, including review of the effect of aging on safety performance and operability	L, CR		Part of Standard Evaluation Plan
Reduced occupational doses	IR	4	
Improved plant availability	DR		Industry
Assurance that field work matches design	L		Inspection
Quantification of low failure rates	CR		Data collection under way (App. D)
Expanded human error studies (in the control room and in the rest of the plant)	IR	3-D	
Study of check valves to improve ECCS	IR	5-C	
Advanced containment concepts	IR	7-A	
Steam explosions	CR		App. D
Continuation of realistic modeling	CR		App. D

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
Individual Consultants (cont'd)			
Development of overall national approach to energy safety to guide allocation of Federal funding	Policy matter		
Methods to realistically combine loads	L		
Tradeoff analysis on bunkered decay heat removal system	IR	6-B	
Availability of ultimate heat removal	IR	6-B	
System effects such as interaction between human factors and transients; systems interactions; potential for pattern recognition	IR	3-D	
Identification of improved concepts from review of WASH-1400 or study of foreign reactors	IR	F	This report
Loading information for a jet plume and cavity pressurization; flow discharge and flow/structural interaction; blowdown loads	CR		NRC programs (Marviken, etc.)
Reduction of radioactivity release with emergency condensing systems	IR	11	
Establishment of trip points	IR	2	
Definition of "undue risk"	CR		Planned under risk assessment (App. D)
Diagnostic tools for operator	IR	3-D	
Alarm priorities	IR	3-D	
Improved operator training for accidents	IR	3-D	
Simplified hardware designs	IR	2	

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
Individual Consultants (cont'd)			
Safety systems that are easier to analyze	IR	5-C	
More work on safety margins	CR		App. D
Optimized safety systems	CR		Some work under way (App. D)
Use of WASH-1400 to determine areas of research	IR	F	This report
Diagnostics for incipient failures	IR	1	
Operator response to accidents	IR	3-D	
Materials improvement	DR		
Use of pipe sleeves to prevent pressure buildup	IR	13	
Broadening data base on seismic loads in order to evaluate anti-seismic designs	IR, CR	12-E	Some work under way
Correlated meteorology program	CR		Some work under way in risk assessment
Potential negative aspects of plant changes such as the suggested improved safety systems	L		
Evaluation models should be compared with best estimate models, which incorporate statistical error techniques	CR, L		App. D
Study of evacuation (including relocation) and development of emergency action plans	IR	16	Also App. D (risk assessment)
Aging problem for equipment	CR		App. D (impact) OCTAVIA code

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
Individual Consultants (cont'd)			
Reduced susceptibility to sabotage	CR, IR	14	Confirmatory research programs under way
Instrumentation to detect incipient abnormal behavior	IR	1	
Siting policy considering core melt	IR	15	
Effectiveness of distributing potassium iodide tables in emergency	IR	16	
Comprehensive underground siting study	IR	7-A	
Optimized testing frequency	L, CR		NRC FRANTIC code
Evaluation of vent valves in PWR core barrels	IR	5-C	
Evaluation of high-pressure rupture disks in divider baffle of PWR steam generator	IR	13	
Use of check valves at bottom of BWR jet pumps	IR	5-C	
Provision of more ECC water to lower plenum	IR	5-C	
Improved vent pipe discharge geometry for BWR suppression pools	IR	7-A	
Improved design of vessel foundation walls for reduced loadings	IR	13	
Improved relief valves for steam and steam-water	CR		NRC following German research
New designs for reactor shutdown systems	IR	8	

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
Individual Consultants (cont'd)			
Use of fire-truck water for backup supply	IR	6-B	
Control of contaminated sump water	IR	10	
Use of refractory cements below reactor vessel to minimize hydrogen evolution and aerosol generation	IR	10	
Unique siting concepts such as offshore siting	IR, CR	15	NRC has done some work
Mitigation of meltthrough effects	IR	10	
Optimization of total system safety	IR		
More in situ qualification testing	L		
Fragility levels of components	IR	12-E	
Determination of "how safe is safe enough?"	CR		Planned under risk assessment (App. D)
Energy absorption designs for improved seismic response	IR	12-E	
Increased separation of components	IR, L	13	
Core catchers as an element of advanced containment systems	IR	7-A, 10	
Explicit identification of containment failure modes for conventional vs. other proposed containment concepts	IR	7-A	
Development and evaluation of the capabilities of various modes of real-time diagnostic/control aids in the power plant control center	IR	2	

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
Individual Consultants (cont'd)			
Operator reliability investigation through use of simulators that can model multicomponent system failures	IR	3-D	
Improvement of safety component integrity as a means of preventing accidents and hence of actually improving safety	IR	1	
Development of better in-core instrumentation for defining more precisely nuclear, thermal, and hydraulic state of the core at any given time. Development of more accurate instrumentation and ability to predict instrument performance during its service life are important prerequisites to developing systems for maintaining the reactor in the safest configuration given single control system failures or other abnormal conditions	DR		
Resolution of generic problems	L		Under way
Definition of rules and development of criteria for backfitting of plants under construction or in operation	L		
Design verification program for use in advance in plant construction	L		
Mandatory expanded failure feedback program, including failure analysis	L		Under review
Comprehensive program to include resident inspectors during construction, startup, and critical phases of operational plants	L		Inspection and Enforcement
Criteria for inspection and certification of nuclear suppliers	L		

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
Individual Consultants (cont'd)			
Criteria for seismic information for construction permits. Complete mapping to determine possible interties of the Hosgri fault with other known fault systems at its southerly/northerly ends	L, CR		
Emergency licensing criteria	L		
Program to improve regulatory effectiveness in quality assurance, assessment of the cumulative impact of regulatory, generic, and backfit deficiencies and in the documentation of deviation in older plants	L		Older plants being reviewed
Improvement program to evaluate importance of siting to accident consequences and of emergency response to consequence mitigation	IR	15, 16	
Program to evaluate the consequences and avoidance of accidents during reactor shutdown	IR	4, 11	
Program to evaluate improved separation of both control and hardware systems	IR	13	
Prevention versus mitigation	L		Defense in depth
Improved seismic design	IR	12-E	
Increased knowledge of the capability of the pressure suppression concept	CR, IR	7-A	App. D (vendor research)
Increased fuel safety research and testing	CR		App. D
Specific criteria for safety R&D program selection	IR	F	

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
Individual Consultants (cont'd)			
Program for on-line flaw detection in piping, control components, and vessels. Loose-parts detection for reactor vessels, steam generator, pumps, condensers, etc.	IR	1	
Programs for design of alternate shutdown systems should be developed with demonstration of their performance	IR	8	
Alternate ECCS designs that simplify computational problem	IR	5-C	
Development of consensus standards to be applied to interpretative and advisory hardware and software	L		
Systems that significantly improve the safety margins and reduce extent of analysis and justification required during licensing reviews and public hearings	IR		To be evaluated as goal for improved safety research
Waste disposal, plutonium, and reprocessing	CR, DR		Work under way
Removal of some design-basis events if safety improvements work out	IR		Part of value/ impact analyses
Correlation of proposed programs to generic issues identified in NUREG-0410, 0138, and 0153	L		
Development of probabilistic approach to earthquake criteria and more statistical information on component behavior under seismic loads	IR, L	12-E	
Addressing the possibility that incidents could occur when a plant is not operating, and these accidents might expose plant personnel to radiation	L, IR	4	

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
Individual Consultants (cont'd)			
Control of contamination	IR	4	
Effect of the balance of plant on safety	L		
Research into dispersion of pollutants	CR		Some work under way
Alternate ECCS that also increases the reliability for reactor cooling after shutdown in the absence of a primary system rupture	IR	5-C	
Improved fire protection	CR		App. D
Development of methodology for systematic basis for (1) value/impact analysis of regulatory change and (2) quantification of an acceptable risk target	IR	F	
Pressure relief and alternate shutdown systems for anticipated transients without scram	IR	8	
Study of failure modes for conventional vs. advanced containment concepts	IR	7-A	
Review of problems in the licensing process to determine problem areas	L		
Evaluation of improved separation of both control and hardware systems	IR	6-B	
APS Study Group Report			
Improved inspection and testing	L		

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
APS Study Group Report (cont'd)			
Improved containment designs	IR	7-A	
Human engineering of controls	IR	3-D	
Controlled venting	IR	7-A	
Improvement of biological data base and development of decontamination techniques	CR, IR	16	
Improved decay heat removal systems	IR	6-B	
Increased automation	IR	2, 3-D	
Alternate ECCS concepts	IR	5-C	
Vented-containment designs	IR	7-A	
Underground siting	IR	7-A (will review California study)	
Core retention devices	IR	10	
Offsite emergency response	IR	16	
Reduction of steam explosion probability	IR	7-A	
Mitigation of groundwater contamination	IR	10	
Evacuation criteria	IR	16	
Remote siting	IR	15	
Improved inspection and test techniques for primary piping	IR	1	
Accidents other than design-basis accidents	L		
Simultaneous transient and massive electrical failure	L		

TABLE C-1 (Continued)

		Disposition	
Research Suggestion and Source	Suggestion Category	Research Topic and Project	Other Disposition
APS Study Group Report (cont'd)			
Increased effectiveness of containment devices	IR	7-A	
Major improvements in containment design (including controlled venting and re-examination of core catchers)	IR	7-A	
Human engineering of reactor controls	IR	3-D	
Automation of reactor controls	IR	3-D	
Increased operator training	IR	3-D	
Quantification of QA effectiveness	L		
Improved protection against sabotage	IR	14	
Quantification of ECCS safety margin	CR		App. D
- more easily analyzed ECCS concepts - more effective ECCS concepts - large-scale experiments - standardized reactors	IR	5-C	
Remote siting	IR	15	
Underground siting	IR	15	
Nuclear-park siting	IR	15	
Uncertainties in consequences estimates	CR		App. D
Mitigation techniques, especially for decontamination and human biological response	IR	16	
Meteorology and dispersion information	CR		Some work under way
Better understanding of core melt	CR		Under way

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
Ford Foundation Study			
ECCS with superior reliability	IR, DR	5-C	App. D
Energy absorption of fragments from burst reactor vessel	IR	9	
Controlled venting of containment	IR	7-A	
Improved control syst ms	IR	2	
Improved control rod independence	IR	8	
Underground siting	IR	7-A (will review California study)	
ECCS Acceptance Criteria			
Steam binding	IR	5-C	
Determination of cladding temperature at which embrittlement ceases to be simply a function of oxidation	CR		App. D
More extensive data base on heat transfer coefficients, cladding oxidation, decay heat, thermal-hydraulics, fuel behavior, pumps	CR		App. D
Rate of cladding/water reaction	CR		App. D
Effects of cladding swelling	CR		App. D
Models for flow redistribution during blowdown	CR		App. D
More sophisticated models for refill-reflood	CR		App. D
More experiments on heat transfer from Zircaloy cladding	CR		App. D

TABLE C-1 (Continued)

Research Suggestion and Source	Suggestion Category	Disposition: Research Topic and Project	Disposition: Other Disposition
Environmental Quality Laboratory			
Reduction in linear power rating	L		ECCS criteria
Thickened rod cladding	L		ECCS criteria
Oxidation of cladding (preoperational)	L		ECCS criteria
Increased reflood rates	IR	5-C	
Reduced steam binding	IR	5-C	
Expanded and accelerated large-scale system testing (e.g., LOFT)	CR		App. D
Fission-product decay heat studies	CR		NRC program completed in 1977 (see App. D)
Large-scale critical break flow	CR		Marviken program (third phase)
Additional FLECHT (reflood heat transfer) tests	CR		App. D
Statistics of LOCA-ECCS analyses	CR		NRC program started at Sandia Laboratories
Accelerated BWR-LOCA models	CR		App. D

APPENDIX D

SUMMARY OF CURRENT REACTOR SAFETY RESEARCH

CONTENTS

APPENDIX D

SUMMARY OF CURRENT REACTOR SAFETY RESEARCH

D.1 CONTEXT OF REACTOR SAFETY RESEARCH

This appendix presents a brief description of existing light-water reactor (LWR) safety research programs, to help establish a framework for assessing the need for additional research to improve safety. While the largest single reactor safety research effort is sponsored by the NRC, an appreciable amount of research is being funded by the reactor industry, the Electric Power Research Institute (EPRI), and other countries. To date, the NRC research has been confirmatory, whereas some of the other research sponsors have on occasion funded developmental research.

Some perspective on the magnitude of current funding for light-water-reactor safety research in and outside the United States may be obtained from the following table:

Sponsor	Amount (millions of dollars) (1978 levels unless noted)
NRC	87
U.S. reactor industry	65*
EPRI	35
Federal Republic of Germany	50-60
Japan	50-60
France	20-30
United Kingdom	15-25
Euratom	10-15

*Amount estimated for 1976; includes both safety and reliability research. The amount for "confirmatory safety research" may have been about $26 million.

A quantitative comparison with the NRC confirmatory safety research budget is difficult to make because (1) the non-NRC numbers may include developmental or improvement-oriented research, (2) the methods of developing and reporting funding levels differ from one organization to another, and (3) some funding information (particularly for the U.S. reactor industry) is not publicly available.

The NRC's confirmatory reactor safety research program[1,2] is directed primarily at providing a capability for an independent confirmatory assessment of the safety of nuclear power plants under postulated accident conditions. The research data and the analysis methods are iteratively applied to the assessment of potential nuclear plant accidents to gain confidence that the margins of safety required in the licensing process are adequate.

The NRC confirmatory research program consists of seven principal elements: primary system integrity, thermal-hydraulic tests, fuel-rod behavior, computer code development, reactor operational safety, site safety, and risk assessment.[3,4]

The U.S. Department of Energy (DOE) is in the process of establishing a developmental safety research program for light-water reactors. Studies are planned in fiscal year 1978 to identify areas in which additional developmental research may be warranted. Initially, DOE is focusing on

- Developing improved-safety systems of unique or novel design.
- Investigating the human/machine interface (specifically data processing) to reduce risk.
- Developing risk-based analytical methods to select research topics from the foregoing items and to guide the research in general.

Through approved reprogramming of funds in fiscal years 1978 and 1979, DOE will probably be funding a research program[5] at a level of approximately $4 million by fiscal year 1980. The personnel of DOE and NRC are regularly exchanging information on their respective programs.

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[1]L. S. Tong and G. L. Bennett, "NRC Water-Reactor Safety-Research Program," Nuclear Safety, Vol. 18, No. 1, January-February 1977.

[2]T. E. Murley, L. S. Tong, and G. L. Bennett, Summary of LWR Safety Research in the USA, NUREG-0234, U.S. Nuclear Regulatory Commission, Washington, D.C., May 1977.

[3]U.S. Nuclear Regulatory Commission Annual Reports for 1975, 1976, 1977 (available from the U.S. Government Printing Office, Washington, D.C.).

[4]Reactor Safety Research Program, NUREG-75/058, U.S. Nuclear Regulatory Commission, Washington, D.C., June 1975.

[5]In addition to establishing a developmental safety research program, DOE has been providing facilities for some of the NRC's reactor safety confirmatory research. In fiscal year 1978, DOE budgeted $28.1 million for these facilities; in fiscal year 1979, DOE is planning to budget $10 million for this activity.

Reactor safety research programs are sponsored by the four U.S. reactor vendors (Babcock & Wilcox, Combustion Engineering, Inc., the General Electric Company, and the Westinghouse Electric Corporation) and EPRI. Exxon Nuclear Corporation sponsors work on computer codes and fuel rods. Much of the reactor-vendor research is proprietary because its developmental nature relates to commercial interests and competitive advantage, and will be discussed only in general terms; EPRI, however, is working under funding by the electric power utilities and has a large, open research program that both parallels and complements the NRC program.[1]

The NRC maintains regular contact with EPRI so that the two programs complement each other as much as is appropriate. In two areas, NRC and EPRI are jointly funding research with the reactor industry. The day-to-day management of these two projects is accomplished by a program management group consisting of one member from each of the sponsoring organizations.

A number of other nations, most notably the Federal Republic of Germany, France, and Japan, are sponsoring reactor safety programs. These programs include both developmental and confirmatory research and in many cases complement or supplement NRC-sponsored programs. The NRC has entered into reactor safety research agreements with some 14 other countries or organizations, including the Federal Republic of Germany, France, and Japan.[2] Through these agreements, NRC exchanges information on research programs of mutual interest.

In reviewing reactor safety research being conducted in other countries, it is important to keep in mind its objectives and the particular socioeconomic conditions that may prevail there. For example, the Federal Republic of Germany is densely populated, and this fact significantly influences the emphasis of its reactor safety research program, particularly in the areas of siting and containment. The United States also considers population density in its siting practices and emphasizes containment measures. In France, the current reactor safety effort is directed toward pressurized water reactors. The goal of French reactor safety research is to develop a reactor that is as "clean" as possible. As a result France is heavily emphasizing studies on preventing fuel failures, maintaining reactor vessel integrity, and performing nondestructive examinations. The United States is also working in this area.

The Japanese reactor safety research has as its goals nuclear safety and the protection of the public from radiation exposure. The Japanese employ a defense-in-depth design philosophy similar to that used in the United States. During the period 1976-1980, Japan is emphasizing research on reactivity-initiated accidents, loss-of-coolant accidents, fuel behavior, structural safety, radioactive effluent control, probabilistic safety assessments, and seismicity.

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[1]Electric Power Research Institute, Research and Development Projects, November 3, 1977.

[2]G. L. Bennett, A. H. Spano, and S. A. Szawlewicz, "NRC International Agreement on Reactor Safety Research," Nuclear Safety, Vol. 18, No. 5, September-October 1977.

The principal reactor safety research under way in other countries is briefly summarized in the sections that follow.

D.2 CATEGORIES OF REACTOR SAFETY RESEARCH

The existing reactor safety research programs, including those of the NRC, may be divided into the following general categories:

- Thermal-Hydraulic and System Interaction Tests: Experiments designed to further elucidate the physical phenomena that occur in postulated accidents. These experiments help the model developers confirm and improve techniques for analyzing the safety systems of commercial nuclear power plants.

- Fuel Behavior Tests: Experiments designed to provide a better physical understanding of the behavior of nuclear fuel rods under normal and abnormal conditions. These experiments are also used in the development of analytical models.

- Primary System Integrity Studies: Experimental and analytical efforts designed to improve the understanding of the metallurgical and mechanical response of the primary system pressure boundary of a reactor during normal conditions and accident conditions.

- Computer Code Development: An analytical program designed to provide better mathematical models and computer codes for calculating the behavior of nuclear power plants under postulated accident conditions.

- Reactor Operational Safety Research: A research effort on the operational safety aspects of nuclear power reactors.

- Site Safety Research: An experimental and analytical effort designed to provide a better understanding of the influence of siting on the safety of nuclear power plants.

- Risk Assessment: Primarily an analytical program in which techniques are developed and used to assess the risk associated with the use of nuclear power.

D.3 CURRENT REACTOR SAFETY RESEARCH

This section describes the kinds of research being conducted under each of the seven categories defined above. As already noted, most of this research is confirmatory in nature. There is, of course, some duplication, but this is helpful in providing checks on the research results. It should be borne in mind that the research information presented in this appendix is based only on the information formally provided to NRC under existing information exchange procedures and as such may not be complete. The U.S. reactor vendor information has been taken from special annual briefings and is presented here in a nonproprietary form.

D.3.1 THERMAL-HYDRAULIC AND SYSTEM INTERACTION TESTS

NRC Program

The NRC focuses particular attention on the performance of the emergency core cooling systems (ECCS) because these systems are designed to keep the nuclear fuel rods cool in the event of a loss or reduction in the primary system coolant. Such coolant losses or reductions are included in the set of design-basis accidents used by NRC in assessing the effectiveness of the various engineered safety features, such as the ECC systems. The NRC thermal-hydraulic test program places special emphasis on the loss-of-coolant accident (LOCA) because this is the severe accident the ECC systems were designed to mitigate.

The NRC thermalhydraulic test program consists of two major parts: (1) LOCA separate-effects experiments and (2) LOCA integral tests. The separate-effects tests cover the specific phases and phenomena of a postulated LOCA (such as the blowdown and reflood experiments and containment response); the integral tests synthesize all these effects in a single test facility [such as the Loss-of-Fluid Test (LOFT) or Semiscale facility].

Information developed in the NRC thermal-hydraulic test program elucidates the basic phenomena and provides an improved basis for developing and verifying better analytical models to describe the accident environment and the resulting effects. The results obtained to date indicate that there is considerable margin for safety in the performance of ECC systems.

Reactor-Vendor Programs

Babcock & Wilcox completed construction of a new 6-MWe heat transfer facility in December 1976. This facility is being used to study the onset of critical heat flux (related to reduced heat transfer) under conditions of different thermal heat flux shapes and simulated fuel-rod-bundle designs.

Combustion Engineering, Inc. (C-E) has programs under way in the areas of reactor flow model testing and evaluation, hot-loop studies of reactor components, critical heat flux (CHF) measurements, and reactor coolant pump model tests. The flow model testing is aimed primarily at providing useful hydraulic and flow distribution design information for the System 80 (3800-MWt) reactor. The hot-loop studies have so far been directed at determining fuel-rod-bundle flow behavior. The CHF studies have been used to improve the C-E model of CHF. The reactor coolant pump model tests are being run under a joint EPRI/C-E program. The NRC will utilize these data in developing improved reactor coolant pump models for LOCA analysis.

The General Electric Company sponsors work on blowdown/ECCS interactions, blowdown heat transfer (i.e., the amount of heat transferred from the fuel rods during the initial depressurization, or blowdown, phase of a LOCA), counter-current flow limiting behavior, and upper plenum phenomena. Some of this work is jointly sponsored by NRC and EPRI. A $17 million high-flow hydraulic test facility to study vibration and other flow-induced phenomena has been almost completed.

In the nonnuclear blowdown/ECCS interaction studies, General Electric is studying integral system effects, including full bundle height and four separate ECC systems. Work has been completed on the BWR/4 fuel bundle, and tests are under way on the newer BWR/6 design. The experimental data show the licensing calculations to be very conservative. The blowdown heat transfer experiments provide an evaluation of a thermal margin calculation and indicate the types of boiling phenomena present. A series of basic countercurrent flow limiting (CCFL) tests have been made on single adiabatic and electrically heated rod bundles. In addition to a study of flow-path mixing in the reactor upper plenum, a more detailed CCFL program is being proposed, this one to include system effects.

General Electric is performing experiments in its pressure-suppression test facility to evaluate the effects of a postulated LOCA on its MARK II containment system. Recent experiments have examined the phenomena occurring during steam condensation and associated dynamic load conditions on the suppression pool and vent system boundaries. The extent of thermal stratification in the suppression pool has also been determined.

The Westinghouse thermal-hydraulic testing program includes reflood heat transfer tests, blowdown heat transfer tests, and a reactor coolant pump program. The reflood heat transfer program (cooperatively funded by EPRI, NRC, and Westinghouse) is designed to obtain a better understanding of reflood heat transfer during a LOCA. The blowdown heat transfer tests are being run in the Westinghouse J-loop test facility in a cooperative program with EPRI.

Using mock-up model tests, Westinghouse has obtained experimental data on the temperature in the upper head of a reactor. In addition, Westinghouse is sponsoring a reactor coolant pump test program in France to determine the behavior of a 0.382-scale model Westinghouse pump operated under various steady-state conditions of steam, water, and air.

EPRI Program

EPRI has sponsored state-of-the-art reports on aspects of a LOCA as well as blowdown heat transfer work at Combustion Engineering. It has also sponsored studies on the mixing of emergency core coolant water with steam in scaled facilities at Westinghouse. In addition, EPRI has funded a number of reports on factors affecting critical heat flux and two-phase (steam/ water) flow modeling.

Foreign Programs

The Federal Republic of Germany has run LOCA tests on an electrically heated bundle containing 340 simulated fuel rods [pressurized water reactor (PWR) conditions] and on two parallel bundles containing 49 rods each [boiling water reactor (BWR) conditions]. Future LOCA/ECCS experiments will address processes taking place in the reactor core and upper plenum during reflood and refill. The NRC is planning to cooperate in this program. Pilot LOCA-type experiments have been carried out in a 1/4-scale PWR dry containment system. Experimental thermal-hydraulic studies have been run on a BWR pressure-suppression tank.

In addition to being involved in the ISPRA blowdown tests in Italy and the ROSA II tests in Japan, France has several thermal-hydraulic facilities of its own:

OMEGA: A loop to measure the behavior of up to 36 rods during blowdown.

ERSEC: A reflood loop capable of working at pressures of up to 6 bars with 36 rods (∿3.5 meters in length).

EPIS: A facility to study steam/water interactions.

France is also studying the behavior of 1/3-scale pumps in two-phase flow (EVA program).

Japan has sponsored a number of integral system tests in its Rig-of-Safety Assessment (ROSA) facility, which is analogous to the NRC Semiscale facility. In addition, separate-effects experiments have been run on the blowdown and reflood phases of a LOCA. Japan is planning a large-scale core-reflood test program (NRC and the Federal Republic of Germany are participating in this program).

In Sweden, a number of nonnuclear experimental studies on containment response to LOCA-type conditions have been performed in the Marviken facility. The NRC has participated in this program and has obtained valuable information for use in analyzing the behavior of containment systems used in the United States.

D.3.2 FUEL BEHAVIOR TESTS

NRC Program

The NRC fuel-rod behavior program element is an experimental and analytical program designed to provide a more detailed understanding of the response of nuclear fuel assemblies to abnormal or accident conditions.

The fuel-rod cladding is the first barrier to the release of radioactive materials. Because of the importance of the cladding, it is necessary to understand its coolability and how it can be affected by the course of an accident.

In order to assess the behavior of fuel cladding under abnormal or accident conditions, it is necessary to understand the cladding environment. This makes it necessary to study the internal conditions of the fuel rod, as well as such external conditions as the heat transfer to the reactor coolant water and the effect of possible mechanical deformation of adjacent fuel rods. If it is postulated that a fuel rod does release radioactive material, then it becomes necessary to obtain information on this material and how it can be transported. Finally, should the postulated accident result in fuel melting, then information is needed on how this molten fuel interacts with, and is in turn influenced by, its environment so that some understanding may be obtained on the containment of the radioactivity. From this work additional measures may be established to further reduce the potential risk to the public.

To meet these needs, the fuel-rod behavior program element consists of experimental and analytical efforts in four major areas: (1) studies of fuel-rod components; (2) in-reactor tests of fuel rods, (3) development and verification of fuel analysis codes, and (4) experimental and analytical studies of fuel meltdown and fission product release.

The information generated in the fuel-rod behavior program is used to develop physical models that are incorporated into fuel analysis codes and fission product transport codes. These codes are then verified through integrated in-reactor tests.

The information developed to date has supported the conservatism of the licensing assumptions in this area. The decay heat available to raise the temperature of the cladding in a postulated LOCA is now known much more precisely than it was when the licensing assumptions were established. As postulated, it is less than that assumed in licensing assessments.

The cladding oxidation rate during a LOCA is now better defined than it was when the licensing criteria were established. As postulated, the licensing assessments are conservative.

Concurrently, in-reactor tests have shown the basic integrity of the cladding under a variety of accident simulations, including power-coolant mismatch, reactivity-initiated accidents, and loss of coolant.

Reactor-Vendor Programs

Fuel-rod behavior research at Babcock & Wilcox covers fuel-rod irradiation tests and fuel-rod development. In a fairly extensive fuel-rod irradiation and examination program, Babcock & Wilcox is studying fuel behavior under normal and power ramp conditions, cladding creepdown and collapse, and the fuel centerline temperature. Special emphasis is being placed on a new fuel-rod bundle design. With EPRI support, Babcock & Wilcox is investigating creep and collapse for different types of Zircaloy cladding. Studies on rod bowing are also in progress.

Combustion Engineering research covers fuel and poison rod bowing, iodine spiking tests, fuel irradiation experiments, fuel-element performance improvement, and cladding studies. A fuel model development program, principally directed at phenomena other than the LOCA, is under way. The main purpose of the program appears to be to obtain more flexibility in the operation of nuclear power plants.

Combustion Engineering has studied the behavior of 50,000 operating fuel rods to obtain data on bowing and is analyzing the phenomenon of iodine spiking, which occurs when there is a power change. Its fuel irradiation program is still in an iterative stage with respect to model development and testing. The fuel-element performance improvement work is aimed at preventing fuel-cladding failures from pellet-cladding mechanical interactions. Combustion Engineering is planning some fuel-cladding damage tests at Kraftwerk Union (KWU) in Germany. These tests will complement NRC-funded multirod burst tests at the Oak Ridge

National Laboratory. The effects of oxidation on the performance of the Zircaloy cladding are also being studied.

At General Electric, fuel behavior work is generally related to eliminating the possibility of cladding failures from pellet-cladding interactions. Some modeling work is also under way.

Through experimental studies and evaluations of actual fuel performance, General Electric has established a set of preconditioning interim operating management recommendations to preclude failures from pellet-cladding interactions. These recommendations are essentially concerned with reducing cladding strain. Coupled with new designs, these recommendations have resulted in greatly reduced fuel-rod failure. Some of the pellet-cladding interaction remedy work is funded by EPRI. The new General Electric fuel model is being prepared for submittal to the NRC for review. When approved by the NRC, this model will be used in the thermal-mechanical design of fuel rods and in establishing the initial conditions for LOCA analysis.

Westinghouse has pursued some model development to account for observed fuel-rod bowing. However, the basic aim of its fuel behavior program is product improvement.

EPRI Program

In addition to the fuel research it is sponsoring, EPRI has programs to study decay heat (which complemented and provided independent corroboration of the NRC program results), cladding oxidation, pellet-cladding interactions, and fuel-rod behavior under reactor operating conditions. Studies on the use of plutonium fuel in light-water reactors have been supported. Under an EPRI program, information has been developed on the densification of fuel in light-water reactors. Thermal and structural studies of fuel rods have also been sponsored.

Foreign Programs

Considerable research has been done in the Federal Republic of Germany on simulated core meltdown accidents. The general conclusion is that all of the processes that can lead to containment damage require considerably more time than has been assumed to date.

France has a number of fuel behavior studies in progress, including a cooperative program with Sweden on LOCA-related cladding studies. One goal of this research is to develop a fuel failure detector. PHEBUS, a French in-reactor loop test facility, will be used to run LOCA-type loop tests on fuel rods. Some research on fission product release and fuel meltdown is in progress.

Fuel behavior research in Japan covers operational, transient, and accident conditions. Fission product transport and radiation control are also being studied. Under the Nuclear Safety Research Reactor (NSRR) program, Japan is studying the threshold value for fuel rupture during a reactivity-initiated accident (RIA). The energy released during an RIA and its effects on reactor components will also be studied. Computer codes will be developed to model RIAs. In addition, Japan is sponsoring research to develop the means of decreasing the release of certain fission products during normal operation and during an accident.

Norway manages an international fuel research program at its Halden reactor. A number of nations have instrumented fuel assemblies under test in this reactor. The NRC is a participant in this program.

D.3.3 PRIMARY SYSTEM INTEGRITY STUDIES

NRC Program

The NRC primary system integrity research is an experimental and analytical effort designed to upgrade the NRC basis for the design, fabrication, operation, and inspection criteria for the reactor primary system pressure boundary. Analysis procedures for evaluating the performance of the pressure vessel, piping, and associated components of the primary system pressure boundary of LWRs under normal, upset, and accident loadings are an important element of the program. A primary goal is to improve the definition of failure probabilities and failure modes, and to establish ways by which the failure probabilities can be reduced if this is considered necessary.

Special attention is given to the study of the primary system pressure boundary of light-water reactors because of the need to confine the nuclear core materials at all times, and thus the need to understand the types of failures in the primary system that might lead to breach of this confinement. The pressure vessel and piping constitute the second barrier to the release of radioactive material. The primary system pressure boundary of current reactors includes (1) a steel pressure vessel 6 to 12 inches thick, (2) steam-generator tubes, and (3) steel primary piping as much as 4 inches thick. The material properties of the primary system pressure boundary components have been studied extensively; however, improvements in this basic information are still sought to round out the basis for judgments affecting continuing reactor safety.

The NRC program is designed to provide information on the integrity of the primary system pressure boundary of light-water reactors. It consists of three experimental and analytical subelements: fracture mechanics, operational effects, and flaw detection.

The fracture mechanics work encompasses (1) vessel performance (hydraulic and pneumatic loading), (2) crack arrest (including static and dynamic studies and the use of irradiated specimens), and (3) response to postulated accident transients. The vessel response transient work encompasses thermal shock and steam-line-break accident conditions to assess the effects of abnormal pressures and shock following the injection of relatively cold emergency core cooling water after a LOCA.

The operational effects work encompasses studies on (1) crack growth, (2) irradiation embrittlement, and (3) steam-generator corrosion, intergranular stress-corrosion cracking, and sensitization.

These data are needed to quantify the loss in toughness of irradiated structural materials and the integrity of the steam-generator tubes. In view of the cracks and denting found in certain PWR steam-generation tubing, this work has immediate application to safe reactor operation.

The flaw detection work covers (1) improved ultrasonic characterization of flaws, (2) acoustic emission studies of flaw growth in piping and pressure vessels, and (3) studies of acoustic emission from flaws produced during welding.

In general, the primary system integrity subelements are geared to produce information to further develop and verify analysis procedures for crack propagation and arrest, steam-line break, thermal shock and pneumatic loading, cyclic crack growth, and irradiation embrittlement, all of which help establish the integrity of the primary system pressure boundary. Another objective is to develop additional basic criteria for testing procedures to ensure improved accuracy, value, reproducibility, and correlation of results. Ultimately, the results will be incorporated into improved industry code rules and standards for improved reactor safety designs and will help improve the basis for NRC decisions on operating reactors.

Reactor-Vendor Programs

Babcock & Wilcox has been studying leaks in its once-through steam generator. This program is rather extensive, including the use of various nondestructive examination techniques and analyses.

Combustion Engineering sponsors a reactor vessel material surveillance program and is studying the behavior of steam-generator tubes under operational and accident conditions. In its reactor vessel material surveillance program, Combustion Engineering is emphasizing the determination of radiation effects on pressure vessel materials. This work is primarily design oriented. Combustion Engineering is also structurally analyzing the behavior of steam-generator tubes under the loading conditions associated with a postulated LOCA. Another effort is the development of criteria to determine which steam-generator tubes should be plugged in order to overcome any difficulties from denting.

The General Electric research in this area deals primarily with eliminating cracks in the feedwater nozzle and sparger and in the control-rod drive. Design and material improvements have been selected and are being qualified by tests. Contingency repair development programs are under way.

Working with EPRI support, Westinghouse is studying methods of annealing the radiation damage out of a reactor pressure vessel. In addition to studying the behavior of its steam-generator tubes and working on new, improved tube designs, Westinghouse is sponsoring research on acoustic emission techniques in order to locate pipe cracks. Both in-reactor and loop tests have been performed to obtain extensive data on flow-induced vibrations. Westinghouse currently has an analysis code that couples hydraulic and structural behavior.

EPRI Program

EPRI is funding research (such as that at Westinghouse) on the annealing out of radiation damage and on crack growth detection and prevention. It is also funding work on corrosion, nondestructive examination techniques, fracture mechanics, and residual fatigue life analysis (including probabilistic fracture mechanics, welding, fittings, and thermal and stress analyses and other related analytical model development).

Foreign Programs

The Federal Republic of Germany researchers have developed a remotely controlled ultrasonic testing system that will permit a complete nondestructive repeated proof test on the entire reactor vessel. Studies on the fracture safety of primary coolant system components have been carried out with the objective of establishing the safety relevance of cracks in the heat-affected zone of welds and in the base metal in connection with embrittled materials. An extensive series of nonnuclear structural tests is planned for the components of the decommissioned 100-MW Heiss-Dampf Reaktor (HDR) experimental system.

France has run structural tests on vessels, drums, and models. It is emphasizing the elimination of irradiation embrittlement. Statistical studies of the failure probability of reactor pressure vessels are also in progress.

At Fessenheim, France is performing acoustic emission studies on the primary system. Steam-generator tube integrity is being studied by means of Foucault current probes. The effects of pipe whip during a LOCA are being studied in the AQUITAINE program.

Japan is emphasizing studies on stress-corrosion cracking, pipe whip, and steam-generator performance. The goal is to improve structural integrity and system reliability.

D.3.4 COMPUTER CODE DEVELOPMENT

NRC Program

The NRC computer code development program is an analytical effort designed to provide better digital computer codes for use in computing the behavior of full-scale reactor systems under postulated accident conditions. These codes are a key part of the safety assessment of nuclear power plants. Completed NRC codes are made publicly available through the Code Center at the DOE Argonne National Laboratory.

Most of the present code development work is aimed at assessing the consequences of a LOCA and the behavior of the emergency core cooling system in pressurized water reactors and boiling water reactors. However, the development of many of these codes is proceeding in a flexible, modular fashion so that, in the near future, they will easily be made applicable to other postulated system transients such as anticipated transients without scram (ATWS), reactivity-initiated accidents (RIA), etc. Computer code development is a very important part of the NRC reactor safety research program because the computational techniques embodied in the computer codes provide the principal means of assessing accidents in nuclear power plants. These computational methods provide the means of assessing the consequences of a postulated LOCA, including the performance of the engineered safety features in preventing postulated fission product releases and the response of the reactor system to other postulated accidents. Scaling of models and experimental data is a key part of the code development program element.

The NRC computer code development program element encompasses analytical subelements designed to model accidents, especially the LOCA, in light-water reactors. The principal subelements are system codes, component codes, and code verification.

The NRC system codes model the entire postulated accident and/or the major parts of a nuclear power plant. The component codes model in greater detail the behavior of the various components of a reactor system. Code verification is the process by which the developed computer codes are independently validated against test data to determine their analysis capability.

Concurrently, NRC is sponsoring a comparison of its best estimate computer models with the more conservative licensing evaluation models using statistical error techniques. Under the NRC licensing program evaluation models and some best estimate models are also compared through special experimental prediction exercises.

To date, NRC-funded researchers have extended the existing systems code (RELAP) to include improvements in the modeling of PWR reflood and have released to the public the first version of an advanced systems code, TRAC. This new code uses more sophisticated numerical and modeling techniques.

The NRC now has available the first version of a new best estimate containment code that is useful in assessing the performance of the third barrier--the containment system--in preventing the release of radioactive material.

To date, all of the modeling efforts have shown adequate agreement with test data, giving further confidence that reactor safety is conservatively bounded.

Industry Programs

Babcock & Wilcox has developed a mathematical technique (using response-surface methodology) to establish existing safety margins through the application of statistical analysis to actual, operationally determined, plant behavior.

Model development at Combustion Engineering includes thermal-hydraulic analyses, reactor kinetics, steam-line-break analyses, LOCA models, reflood heat transfer, and the utilization of nuclear steam supply system operating data for the validation of physics design methods. The thermal-hydraulic analyses revolve around experiments and a C-E code (TORC) based on the NRC COBRA-IIIC computer code. The reactor kinetics work is done with the C-E HERMITE code. The TORC and HERMITE codes have been merged to obtain a three-dimensional thermal-hydraulic code with reactivity feedback effects. Analytical studies on the postulated secondary (steam-line) break are under way. The C-E LOCA "best estimate" model is an attempt to put some realism (as opposed to the conservatism of licensing analysis) into LOCA analysis. The reflood heat transfer modeling is based on the C-E code THERM, which is verified by comparison with test results. The use of operating data for code analysis is primarily aimed at optimizing the reloading of cores.

General Electric has not developed best estimate models for the entire LOCA transient; however, best estimate models have been developed to describe phenomena observed in the experimental studies on blowdown/ECCS interaction, which were described in Section D.3.1.

Westinghouse has developed an analytical model to describe the behavior of its upper head injection system, which is an alternate ECC concept. In addition to the upper head injection modeling, Westinghouse has developed a new drift flux model, has analyzed the NRC-sponsored fourth LOFT test (to check out a Westinghouse code), has analyzed the Japanese ROSA II upper head injection tests, and is modeling the response to a postulated LOCA in a plant with a loop out of service. A model for describing interactions and steam-line breaks has also been developed.

Working under EPRI sponsorship, Westinghouse has developed a statistical method for performing safety analyses of nuclear plants.

EPRI Program

EPRI is sponsoring the development of system codes to analyze various accidents such as a LOCA, ATWS, and RIA. It has also sponsored several data compilations relating to the nuclear design of reactors. Analytical studies of various experiments, including those sponsored by NRC, will lead to improved and validated reactor models. EPRI has recently set up a code center to disseminate its codes.

Foreign Programs

The Federal Republic of Germany, France, and Japan all have active code development programs. In addition, the Nordic group (Denmark, Finland, Norway, and Sweden) has been developing sophisticated LOCA codes.

D.3.5 REACTOR OPERATIONAL SAFETY RESEARCH

NRC Program

The NRC reactor operational safety research program is an experimental and analytical program designed (1) to support other NRC offices in the development and confirmation of regulatory standards and guides and (2) to provide research information on specific reactor operational safety matters.

Aside from the design-basis LOCA that is used by the NRC licensing staff to evaluate the engineered safety features of a nuclear power plant, there are a number of operational safety areas of interest. Many of these derive from the day-to-day evaluation of plant operating behavior and are often the subject of specific regulatory guides and standards.

Included in this NRC program element are fire protection research, component qualification testing evaluation, noise diagnostics, human engineering, and general reactor operational safety studies and support.

To date this program element has verified the NRC electrical cable tray separation criteria used to prevent the spread of electrically initiated fires. In addition, in assessing qualification testing methods, researchers have found no synergistic effects for test components exposed to the simultaneous environments of a postulated LOCA rather than to a sequential set of environments. A product of the noise diagnostics subelement has been the verification of a manufacturing fix in BWRs to prevent channel box wear. Human engineering activities are under way in support of NRC inspection and enforcement activities.

EPRI Program

While each of the U.S. reactor vendors has its own operational safety program, most of the material in the open literature has been published by EPRI.

EPRI has sponsored measurements on the radiation levels at nuclear power plants and the sources. Surveys have been made of the nuclear power industry's needs for information on radiation protection, radiation transport, and shielding. An evaluation of scale-model methods for operability qualification of seismic category pumps and valves as well as an assessment of industry valve problems have been funded by EPRI.

In addition to sponsoring a study of remote multiplexing for power plant applications and a summary of nuclear power plant operating experience, EPRI has funded a review of control-room design from a human factors standpoint. Studies on failure analysis and failure prevention have been undertaken.

Foreign Programs

The Federal Republic of Germany, France, and Japan are all actively involved in programs designed to improve reactor operational safety. Special emphasis has been placed on controlling radiation doses and radioactive releases.

D.3.6 SITE SAFETY RESEARCH

NRC Program

The NRC site safety research program was set up to provide information to assist in confirmation that nuclear power plant sites have been properly characterized with regard to the effects of earthquakes, tornadoes, floods, and other natural phenomena. Another objective is to provide evaluations of engineering design methods and practices used to mitigate the effects of natural phenomena and to quantify the levels of conservatism provided. A third objective is to assess alternate concepts related to nuclear facility siting (such as floating nuclear plants and underground siting) in order to provide technical information for review of future facilities that may employ the concepts.

NRC-sponsored research under way in site safety has evolved in response to information needs within five general categories: (1) regional information on severe environmental phenomena; (2) understanding of seismic, hydrologic, and meteorologic events; (3) methodology for geotechnical, hydrological, and meteorological site evaluation; (4) assessment of engineering design methods and practices; and (5) evaluation of alternate site related concepts.

Current NRC research emphasis is placed on regional seismological evaluations in the eastern United States; measurement, characterization, and geographic distribution of tornadoes; and assessing methodology for site evaluation. This program is closely coordinated with related programs in other Federal agencies (U.S. Geological Survey, National Oceanic and Atmospheric Administration, National Science Foundation, Department of Energy) and other organizations (New York State, Electric Power Research Institute).

Under this NRC program element microearthquake detection networks are now installed in areas with a historical occurrence of large earthquakes: Charleston, South Carolina; New Madrid, Missouri; and Anna, Ohio. The Northeastern U.S. Seismic Network is operating for similar reasons but covers a larger area. The ultimate goal of these and related studies is to provide an improved basis for the assignment of earthquake intensity to different siting regions for design purposes.

Data are being collected and collated on tornadoes, tsunamis, atmospheric turbulence, and earthquake motions to develop and validate improved analysis models of nuclear plant behavior under extreme environmental conditions.

EPRI Program

Most of the publicly available industry literature on site safety has been developed under EPRI programs. EPRI funded an assessment of the seismic design of nuclear power plants as well as the development of wind field and trajectory models for tornado-propelled objects. Full-scale tornado-missile impact tests have been carried out at Sandia Laboratories under EPRI sponsorship. Plume diffusion studies have also been performed.

Foreign Programs

The Federal Republic of Germany plans some seismic related research involving the use of explosives to simulate earthquakes near a nuclear test facility.

Japan has an extensive research program in this area, including a 15 x 15 meter shaker table capable of handling 1000-ton components. This research is aimed at determining the response of reactor components and systems to seismic disturbances and at establishing appropriate safety margins.

D.3.7 RISK ASSESSMENT

NRC Program

The risk assessment methodologies developed in the Reactor Safety Study[1] are of significant potential use in many areas of nuclear safety research. In response to the needs of the nuclear regulatory process, NRC is performing work on a number of aspects of risk assessment research.

The basic avenues being followed and planned in NRC risk assessment research are:

- Continued development of risk assessment and probabilistic analysis methodology, the examination of LWRs whose designs differ from those analyzed in the Reactor Safety Study, and the performance of additional accident risk assessments for appropriate portions of the nuclear fuel cycle.

- Performance of in-house research and management of external research to meet the needs of other NRC offices (such as in the analysis of generic issues) and to develop useful tools for licensing reviews and for use by other offices of NRC (e.g., Standards Development, Inspection and Enforcement, and State Programs).

- Training of personnel in NRC and in government laboratories to enhance NRC capabilities in this area and to implement increased utilization of probabilistic analysis techniques in the licensing and regulatory processes.

- Development of a plan for studies to determine acceptable levels of risk for nuclear facilities.

These are discussed in detail below.

Methodology Development. In order to improve the quality of risk assessments, a significant amount of NRC-sponsored work is now in progress in the area of methodology development. This work involves:

- Checking and improving the Reactor Safety Study consequence model in regard to meteorology, the effect of rain, and predictions of health effects, as well as to make sensitivity studies to determine important parameters.

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[1]Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400, U.S. Nuclear Regulatory Commission, Washington, D.C., October 1975.

- Improving modeling capabilities in regard to seismic effects, fire effects, human errors, and common-mode failures.

- Establishing a system for the collection and analysis of reactor plant component failures in response to the needs of NRC Offices.

- Application of the Reactor Safety Study methodology and insights in conjunction with further methodological development on other portions of the nuclear fuel cycle.

Methodology Applications. The methodology utilized in the Reactor Safety Study (RSS) has found use in a number of pertinent areas. For example, a sizable effort is under way to examine reactors whose safety features are significantly different in design from the two reactors examined in the RSS, in order to extend the applicability of engineering insights gained in the RSS and to explore their effects on predicted risks. This effort will aid in the utilization of probabilistic techniques in licensing processes and in the performance of future risk assessments.

The performance of in-house analyses and research to assist other NRC offices has been and continues to be an extensive effort, resulting from requests made by the Advisory Committee on Reactor Safeguards and the growing recognition in the various NRC program offices of the utility of probabilistic techniques in regulatory processes. Examples of such analyses are the assessed impacts of computerized reactor protection systems as well as the following:

- Seismically induced fires
- Turbine missiles
- DC battery failures
- Reactor vessel overpressurization incidents

There has been continued development and utilization of the computer code FRANTIC, which permits analyses of reactor systems to optimize their reliability as a function of testing requirements. This code is being used by the NRC Office of Nuclear Reactor Regulation to establish testing requirements for technical specifications that have a better foundation than those previously established on a more judgmental basis.

In addition to studies on safety improvements achievable by alternate containment designs, work is under way to provide a quantitative assessment of the risks from Class 3-8 accidents (i.e., accidents equivalent to or smaller than the design-basis accident in severity). Also in progress are efforts to examine ways in which probabilistic techniques can be used to aid inspection and enforcement processes, and to provide a technical basis for guidance to states on emergency plans.

A program has continued for the training of NRC personnel in the techniques and applications of the Reactor Safety Study methodology. Five 2-week courses have been previously offered; such courses will continue in the future. Plans are also being developed to train NRC personnel in depth to help develop probabilistic capabilities in other offices. This is an important effort because the proper application of probabilistic techniques can do much to aid efficiency and to stabilize the licensing process.

There is some existing opinion that it is necessary to define criteria for an acceptable level of risk for nuclear power plants. However, the quantitative determination of acceptable levels of risk on a broad, socially acceptable basis for any endeavor is a formidable task. Although the Reactor Safety Study made a first step in quantitative risk assessment, the quantification of benefits and the comparison of risk and benefits in commensurate terms appear to be extraordinarily difficult and will require many years of research. It has been determined that such analyses would be a useful, long-term program; as such, a program is now in the process of formation.

EPRI Program

The use of statistical techniques by the U.S. reactor industry was described in the preceding sections. In addition to their work, EPRI is sponsoring studies on statistics and risk assessment. As noted earlier, EPRI has sponsored a probabilistic fracture mechanics program. Several probabilistic safety analysis reports have come out of the EPRI program, including a critique of the NRC Reactor Safety Study. Sensitivity assessments in reactor safety analysis have been made. At Westinghouse, EPRI sponsored a methodology development for the statistical evaluation of reactor safety analysis. An EPRI-sponsored effort related to this topic is work on modeling and estimating system availability.

Foreign Programs

The Federal Republic of Germany is sponsoring a risk analysis study similar to the NRC Reactor Safety Study.

France places great emphasis on probabilistic techniques and has set up a reliability system covering about 1000 components or systems. Seismic mapping, aircraft crash simulations, and the effects of industrial dangers are being studied.

Japan is collecting operational data and is developing a method to conduct a probabilistic safety assessment of nuclear power plants.

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