NUREG-0795

Comments on the NRC Safety Research Program Budget for Fiscal Year 1983

U.S. Nuclear Regulatory Commission

Advisory Committee on Reactor Safeguards

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Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

July 17, 1981

The Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

The Advisory Committee on Reactor Safeguards is pleased to transmit its comments on the Office of Nuclear Regulatory Research Budget for FY 1983.

Only that portion of the budget relating to Program Support has been considered. No attempt has been made to distinguish between Program Support Funds for research and for work related to standards development, since the latter are a relatively small proportion of the total.

The proposed funding levels considered are those recommended by the Executive Director for Operations for consideration by the Commission.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

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PART I

GENERAL COMMENTS

GENERAL COMMENTS

1. Introduction

Prior to our last report to the Congress on the Nuclear Regulatory Commission (NRC) Safety Research Program for FY 1982, NUREG-0751 (Ref. 1)*, we have taken the point of view that all of the safety research being proposed by the NRC Staff was generally useful, and have recommended areas which should be given priority or increases in funding. Hence, in our previous report to the Commission, NUREG-0699 (Ref. 2), although we recommended that the LOFT experimental program be terminated after FY 1982, we did not propose that LOFT funding be cut in FY 1982 to permit the recommended increases in Severe Accident Phenomena and Mitigation Research, and Systems and Reliability Analysis, or to cover research on Fast Breeder Reactors. However, in NUREG-0751, we took the point of view that budgetary constraints would exist, and recommended cuts in LOFT, in Loss-of-Coolant Accident (LOCA) and Transient Research, and in Waste Management to permit our recommendation for greater emphasis and funding to be given to Plant Operational Safety, Severe Accident Phenomena and Mitigation Research, and Systems and Reliability Analysis.

In this document we will again assume that budgetary constraints exist and that funding for the Advanced Reactors program also must be accommodated within a total funding level at or close to that proposed by the Executive Director for Operations (EDO).

2. Previous Recommendations

In NUREG-0699, we outlined several steps that we believed needed to be taken by the NRC if those safety research areas which are judged to have potentially the greater impact in protecting the public health and safety are to receive the necessary priority. These steps are listed below, together with some brief comments on their current status.

- The Commission will have to provide policy guidance on the major open safety issues.

  We believe that this basic task remains to be done.

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*References appear in Appendix A

- The NRC research user offices will have to reevaluate their approach to formulating requests for research and strive to consider these in some broad framework which takes into account the major issues confronting the agency.

  Since December 1980, the Office of Nuclear Reactor Regulation (NRR) has begun to exhibit significantly more cohesion and depth in its requests for research and in its comments on the safety research plan. However, we believe that NRR will have to devote further attention to developing priorities for its research requests and devote more effort to defining its longer-range research needs.

- The Office of Nuclear Regulatory Research (RES) will have to reevaluate its current and proposed programs in terms of risk-reduction potential and major regulatory needs.

  Although RES uses probabilistic risk analysis to assign priorities to unresolved safety issues and generic issues, to various alternatives proposed for the Degraded Core Cooling rulemaking, and to other major activities of NRR, it does not use such analysis to provide an important input into an evaluation of the efficacy of its own research program. We recommend that it do so. We also recommend that RES critically evaluate its research programs to see if its major experimental or theoretical programs are really capable of and likely to provide information of an importance commensurate with their expense.

- The NRC will have to judge whether some research, particularly that which involves large scale component testing or the application of existing methodology, should be the responsibility of industry rather than the NRC.

  There is a need for policy guidance on which research matters can and should be the responsibility of the industry.

- The NRC may have to reduce sharply some research which is merely confirmatory in nature where there is a good reason to believe that the current regulatory requirements provide adequate protection to the public.

  The NRC Staff appears to have made a good beginning in this direction.

In NUREG-0699, we identified several areas as requiring emphasis, including the following:

- plant operational behavior as a function of design and control;
- the impact of control systems and other nominally non-safety systems on safety;
- improved approaches to reduce the impact of design errors;
- reexamination of the general design criteria.

In NUREG-0751, we recommended that several areas, including the above, be given higher priority and increased funding in FY 1982 as follows:

"(a) The role of control systems in safety;

(b) Plant operational safety, including system behavior as a function of design;

(c) Reliability analysis of existing plants, including emphasis on the more detailed understanding which might prevent many operating occurrences which have potentially serious implications;

(d) Improved, more reliable shutdown heat removal systems, including dedicated and bunkered systems;

(e) Studies of degraded core and core melt accidents with emphasis on the conceptual design and evaluation of features to mitigate such accidents;

(f) Studies of the physical and chemical behavior of fission products in post-accident environments;

(g) The early development of an approach to supplement or replace the single failure criterion; and

(h) Other matters relevant to setting the principal design bases for plants to be constructed."

The NRC program for FY 1983 appears to be fairly responsive to our recommendations on items (c), (e), and (f) above. Programs are being initiated on items (a), (b), and (g), but require substantially greater funding. Items (d) and (h) seem not to be specifically identified by the NRC Staff as important safety research matters for FY 1983. Although the Task Action Plan A-45 on Shutdown Decay Heat Removal Requirements (Ref. 3) deals with much of the content of item (d), its initial emphasis is on existing plants. We believe RES should participate significantly in this work, and in addition, should carry on a more general investigation, the results of which can contribute to the design of future plants.

3. Design-Related Research

We wish to call attention to an aspect of safety research that has been weak or deficient in the past NRC program, namely, design-related safety research. In order for the NRC to develop improved safety criteria, standards, and guides, it is important that the necessary knowledge of design possibilities, capabilities, tradeoffs, etc., be available. This is the case whether the NRC is issuing design criteria or performance criteria. It is the case for the rulemakings on Degraded Core Cooling and Minimum Engineered Safety Features.

The NRC Staff has funded some reviews and introductory evaluations of design alternatives for improved shutdown heat removal systems, for containment approaches to mitigate degraded core accidents, and for design features to reduce the potential for successful sabotage. These studies represent a potentially useful first step. But their ultimate usefulness depends on the completion of a next step in which enough knowledge and information is developed to permit the promulgation of an appropriate NRC rule. A modest effort along these lines has been proposed for the Degraded Core Cooling and Minimum Engineered Safety Features rulemakings. However, the effort proposed is far too small to provide the needed information on a timely basis, and it does not include a program to develop design requirements with regard to sabotage or the specific objective of developing NRC requirements for future light-water reactors (LWRs), particularly standard plants.

We believe that a sufficient emphasis and funding should be given to such design-related safety research to provide the needed information on a timely basis.

4. Budget Recommendations

We make the following budget recommendations for the NRC safety research program in FY 1983, arranged by Decision Units and summarized in Table 1.

LOCA and Transient Research

The proposed budget level is adequate. However, $2 million should be reallocated within this Decision Unit from the program on Code Assessment and Application to the program on Code Improvement and Maintenance.

LOFT

Only the funding level necessary to place the reactor in a standby status and to cover project closeout costs and a completion of engineering and analysis tasks associated with the FY 1982 experimental program should be allocated for LOFT in FY 1983. This has been roughly estimated by the NRC Staff to be about $14.5 million.

Accident Evaluation and Mitigation

The proposed total funding of $50.4 million in FY 1983 appears to be adequate. However, it should be reallocated internally to provide significantly greater effort on the program on Accident Mitigation, enough to provide the information needed for decisionmaking in the Degraded Core Cooling rulemaking by the end of 1983. This should be accomplished by a reevaluation of the merits of the currently proposed experimental programs.

Advanced Reactors

A funding level of $20 million should be allocated for Liquid Metal Fast Breeder Reactor safety research and $2.5 million for Gas-Cooled Reactor safety research.

Reactor and Facility Engineering

We recommend an increase of $2 million to the proposed funding of this Decision Unit to a level of $40.2 million; these funds are to initiate a more comprehensive safety research program on reactor aging effects and to permit augmentation of the work in probabilistic seismic safety, including study of a BWR.

Facility Operations and Safeguards

We recommend an increase of $3 million to the proposed total budget of this Decision Unit to a level of $19.8 million partly to enable a more appropriate emphasis on control systems, and on plant operational behavior. Additional effort is also needed on the contribution of human error to risk. This should include not only operators and operating practices, but also maintenance personnel and maintenance practices. Studies on design steps to prevent sabotage should be accomplished within this Decision Unit, with appropriate input from the Division of Risk Analysis.

Waste Management

The proposed funding level is appropriate.

Siting and Environment

The proposed funding level is appropriate.

Systems and Reliability Analysis

We recommend an increase of $4 million in the funding of this Decision Unit to a level of $25.7 million; these additional funds should be used specifically for research on the matters recommended in Chapter 9 of this report.

5. Specific Comments and Recommendations

Specific comments and recommendations regarding the scope, nature, and funding levels of the various Decision Units of the research program are presented in Part II of this report.

TABLE 1

OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAM SUPPORT BUDGET FOR FY 1983
(DOLLARS IN MILLIONS)

	PROPOSED	ACRS RECOMMENDATIONS
1. LOCA AND TRANSIENT RESEARCH	31.0	31.0
2. LOFT	40.6	14.5
3. ACCIDENT EVALUATION AND MITIGATION	50.4	50.4
4. ADVANCED REACTORS	2.5 (21.0)*	22.5
5. REACTOR AND FACILITY ENGINEERING	38.2	40.2
6. FACILITY OPERATIONS AND SAFEGUARDS	16.8	19.8
7. WASTE MANAGEMENT	19.6	19.6
8. SITING AND ENVIRONMENTAL RESEARCH	14.7	14.7
9. SYSTEMS AND RELIABILITY ANALYSIS	21.7	25.7
TOTAL PROGRAM SUPPORT	235.5	238.4

*Set aside by EDO for Commission action.

PART II

SPECIFIC COMMENTS

1. LOCA AND TRANSIENT RESEARCH

1.1 Introduction

This Decision Unit includes several programs which are directed primarily toward improved understanding of reactor behavior in loss-of-coolant accidents (LOCAs). In previous reports, we have remarked on the extensive reorientation of these programs which gave increased attention to small-break problems. We continue to support this change of emphasis, and in addition suggest that the research program be broadened to consider wider classes of transients other than those associated only with LOCAs. We shall reiterate this point in our discussion in the next chapter on LOFT.

Included in this Decision Unit is a program of improvement and assessment of codes which has as its objective an analytic description and understanding of light-water-reactor (LWR) transients. The last group of programs in this Decision Unit is directed toward the understanding of core and fuel behavior under conditions in which the core is inadequately cooled. Comments on all of these programs follow.

1.2 Semiscale

Deficiencies in size and configuration of Semiscale are compensated for, at least in part, by the economy of test operation and the relatively rapid turn around test time. The NRC Staff has undertaken a serious study of the limitations of Semiscale and of the scaling questions which arise in translating observations in Semiscale to full scale. We commend this effort and urge its continuation.

We support the program in Semiscale (Mod-2A) and also recommend another version of Semiscale (Mod-5) directed toward the Babcock and Wilcox type pressurized water reactor (PWR). This recommendation for the Mod-5 version of Semiscale was made in NUREG-0699 (Ref. 2)*, and we hope that this program can be implemented in the near future.

1.3 Separate Effects Experiments and Model Development

One of the programs in this item is the Two Loop Test Apparatus (TLTA) or the planned TLTA upgrade known as Full Integral Simulation Test (FIST). Such a facility is the analogue of Semiscale for

*References appear in Appendix A.

boiling water reactors (BWRs). We have repeatedly recommended an improved program of this kind for BWRs. Some questions also remain regarding the possible significance of core bypass and loss of emergency core cooling (ECC) spray water in a medium to small break LOCA. Tests to evaluate the extent of this problem should be undertaken.

Other programs in this area are FLECHT-SEASET at Westinghouse and the Thermal Hydraulic Test Facility (THTF) at Oak Ridge National Laboratory (ORNL). Useful data have been obtained in both programs. The data from FLECHT-SEASET are of good accuracy but are limited to low pressure; THTF, on the other hand, does operate at high pressure. The THTF program is scheduled to end in FY 1982.

As remarked in NUREG-0699, we believe that the present effort on code assessment is inadequate. For an improved program, additional experiments on separate effects are needed. Many of these experiments would not require large facilities since the experimentation should be directed toward establishing essential physical and engineering bases for the codes. The Office of Nuclear Regulatory Research (RES) has increased its efforts in this direction, and we encourage further increases in these efforts.

The model development program consists for the most part of relatively small projects in various research laboratories. We endorse this kind of program as being useful and productive. RES should appreciate that this program provides a helpful interaction with an important part of the engineering and scientific community and should therefore seek to broaden this contact by finding from time to time new contractors in this community. The NRC still needs to improve its procedures for awarding these contracts.

1.4 3-D Program

This international program involves Japan, the Federal Republic of Germany (FRG), and the United States and was begun when LWR safety research was preoccupied with the problem of large-break LOCAs in PWRs. Since the program, as an international obligation, cannot easily be cancelled, at the least it is in urgent need of redirection. The primary need for this redirection comes from the expected decision by the FRG to proceed with the construction of the Upper Plenum Test Facility (UPTF). The construction and test program for this facility can be expected to last over a decade and the cumulative cost to the United States can be expected to be between \$50 million and \$100 million. The mission of this program was determined some time ago to provide information relating to special questions regarding large-break LOCAs. We believe that the information which will be obtained in UPTF is not of primary importance for PWR safety, and that this information could be obtained at less cost and more quickly in other ways.

It must be emphasized that there are other parts of this international program which have been very useful and have been connected with less costly facilities. Among these should be mentioned the PKL facility in FRG and the Cylindrical Core Test Facility (CCTF) and the Slab Core Test Facility (SCTF) both in Japan. Further, there are plans in Japan to construct a Large Scale Test Facility (LSTF) which promises to be an outstanding facility for the study of medium- and small-break LOCAs in PWRs. The NRC should plan to participate in the program in this facility together with the program in an associated facility, the Two-Phase Flow Test Facility.

In the area of international programs, we recommend that the NRC take steps to get access to the experimental data obtained in Japan from their test program for BWRs. In particular, it would be useful to have access both to the data and to the facilities in which work has been done on LOCAs and ECC system performance for BWRs. We understand that RES is pursuing this possibility and wish to encourage them in this effort.

1.5 Code Improvement and Maintenance

The tasks of developing best-estimate codes for PWR and BWR systems are not complete. TRAC, an advanced code, requires and should receive further work. This work includes both development and assessment. The RELAP-5 Code also should be supported; this code is receiving general recognition as being useful.

This program of Code Improvement and Maintenance can be of basic significance for reactor safety and should be given continued support.

The availability of very fast running codes for analysis of system behavior during transients for BWRs and PWRs should be evaluated. If codes that have the necessary compromises on detail and on accuracy of modeling of system behavior and physics are not available to enable the running of transients very much faster than real time, such code development should be undertaken.

1.6 Code Assessment and Application

It has already been remarked here that the code assessment program is inadequate, particularly in the case of TRAC. We recommend, however, that both TRAC and RELAP-5 assessment be continued at a reduced level if necessary to permit the recommended code improvement.

1.7 Fuel Behavior Under Operational Transients

This program includes pellet-clad interaction (PCI) investigations and fuel rod code maintenance and improvements. These programs are being deemphasized in agreement with our previous recommendations. Some PCI investigations, however, are still underway. We believe that NRC should avail themselves of the considerable PCI work being performed by industry and ensure that work is not duplicated. We have reservations about the validity and thus the value of the PROFIT Code as a guide.

1.8 Recommendations

Aside from the 3-D international program which needs special consideration, the LOCA and Transient Research is budgeted at an economic and efficient level and we recommend that this budget level be approved. We recommend that the budget level for the Code Improvement and Maintenance program be increased by about \$2 million by reallocation within this Decision Unit; these funds should be taken from the program on Code Assessment and Application.

The 3-D Program is an international commitment to which the NRC must adhere. There are many aspects of this program which contribute significantly to reactor safety in a cost effective way. As our discussion above indicates, the only item of concern is the UPTF program in the FRG. It appears that the FRG will proceed with the construction of this facility and the particular impact on United States expenditures will come from the long time span involved before completion. We recommend that the NRC continue participation, but make a serious effort to reduce the expenditures associated with UPTF.

2. LOFT

2.1 Introduction

The LOFT facility has contributed to the understanding of the phenomena encountered in large-break LOCAs and has helped to demonstrate, within reasonable limits, the effectiveness of PWR emergency core cooling systems. More recently, it has been useful in studies of the phenomena which may appear during transients and small-break LOCAs. We reiterate, however, the view expressed in NUREG-0751 and NUREG-0699, that programs in LOFT do not make a cost effective contribution to reactor safety.

2.2 The LOFT Test Program

In NUREG-0751 and NUREG-0699, we recommended that the LOFT test program be terminated by the end of FY 1982. It was our strongly held view that there was an urgent need to transfer the funds thereby made available to other safety research programs which would contribute much more effectively to reactor safety. This is still our view.

2.3 Recommendations

We recommend that a funding level of $14.5 million be provided for LOFT in FY 1983 to place the reactor in a standby status and to cover project closeout costs and a completion of engineering and analysis tasks associated with the FY 1982 experimental program.

3. ACCIDENT EVALUATION AND MITIGATION

3.1 Introduction

Most of the activities under this Decision Unit are meant to provide information that will be needed in the rulemakings that will deal with Siting, Minimum Engineered Safety Features, Hydrogen Control, and Degraded Core Cooling.

Because much of the content of these rules must deal with territory as yet only meagerly explored, a considerable amount of experimental and analytical work will be needed for formulation of these rules. Although the general areas of investigation presently can be defined, it is necessary to raise and to answer a number of questions early in the process in order to focus the work properly.

Illustrative of the types of decisions needed are answers to the following:

- Is the principal emphasis to be put on preventing core melt, or on dealing with the melted core after melting has occurred?
- Should a final system design have to deal with 100% of the hydrogen that can be generated in any conceivable accident? If not, what is the appropriate fraction?
- Is the degraded core problem to be dealt with primarily on a probabilistic or on a deterministic basis?
- Is evacuation of the surrounding population to be seriously considered as an accident mitigation system? If so, how much credit is to be taken for evacuation?

Answers to these, and similar questions that need to be raised, may be modified as the research develops, but they need to be formulated and some initial decisions are required to focus the research program.

In NUREG-0699 and NUREG-0751, we recommended that a high-level task force be established with the responsibility for recommending the research needs and for estimating the resources required to support these rulemaking proceedings. A Degraded Core Cooling Steering Group was established and after some deliberation, recently completed a report. The report does not respond to this need. Guidance is still lacking. We reiterate that decisions must be based on clear identification of information needs and that programs must be designed to respond to these needs. The Commissioners should assist by defining the safety philosophy and objectives to guide the work.

3.2 Behavior of Damaged Fuel

As a bounding estimate, risk assessment studies have assumed that undercooling a core leads to melting the entire core. This very conservative assumption has lead people to ignore the substantial difference between overheating and melting the oxide core. The behavior, and margins to collapse, of an overheated core is an essential yet largely unstudied question where answers are needed for decisions on accident evaluation and mitigation. The program is still in the formative stage but has made good progress.

The examination of the Three Mile Island, Unit 2 (TMI-2) core is highly desirable and efforts should be continued to accelerate the cleanup and core examination at TMI-2.

The Severe Fuel Damage (SFD) test series in the Power Burst Facility (PBF) is intended to provide information on the behavior of overheated cores. It is potentially a much more valuable use for the PBF than the operational transient work now terminating and warrants priority. These experiments will be probing into a very complex area and special attention will be needed if they are to achieve their potential.

We believe that two new programs in this area should be deleted for now and the funds thus made available be shifted to higher priority projects; these are:

(a) The Deformed Core Coolability (DECCA) program. This is expensive, duplicative and in our opinion unnecessary. Better information is available from other sources.

(b) LWR Debris Coolability program. This work is of questionable relevance to damaged core behavior, is an expensive way to do heat transfer work, and tends to duplicate work being done on fuel melt behavior. We would defer it until a clear need and relevance are shown.

3.3 Fuel Melt Behavior

Research in this program aims at a better understanding of the behavior of molten fuel as it drops into the reactor cavity, melts through the vessel, drops on to the basemat below, and interacts with the concrete basemat. This investigation is of great importance. The experimental work is expensive, is difficult to carry out, and even more difficult to interpret. What the NRC Staff described to us was primarily the work planned for FY 1982. Specific plans for FY 1983 appear to be primarily an extension of what will be done in FY 1982. This is appropriate if the FY 1982 work is relevant. However, in this area especially, policy guidance is needed on how much is to be attempted in the way of a detailed understanding of the processes being considered.

The Code Development work associated with these studies is extensive. Here, especially, there should be considerable attention given to the question of how much detail is desirable or feasible. Otherwise, it is possible for the experimental program to become a vehicle for code development while the code becomes primarily a tool for describing the experiment.

This Decision Unit includes also an evaluation of the MARCH Code. Our discussions of the MARCH Code with the NRC Staff and others lead us to urge all who use the Code and its results to recognize its limitations. For example, the behavior of a melted core as described by the MARCH Code is largely a matter of speculation. We recommend further work on the MARCH Code to make it easier to use and to improve its capability.

This program includes also continuing work on hydrogen generation. In light of the importance of the hydrogen problem, the work seems generally appropriate. However, that part devoted to hydrogen generation by corrosion of zinc and ferrous alloys by coolant does not deserve a high priority.

3.4 Fission Product Release and Transport

This work aims at an improved description of the radiological source term for severe accidents. The NRC Staff recently completed and published NUREG-0772 (Ref. 4) which provides technical bases for estimating fission product behavior during LWR accidents. This document points out what is known and identifies what is not known about fission product behavior. Plans are to undertake further research to achieve better definition. We believe this work and associated work on aerosol definition deserve high priority.

3.5 Accident Mitigation

This program concentrates on systems which may be useful in mitigating severe accidents. Current activities include consideration of filtered vented containment systems, core retention systems, and general studies of severe accident sequences. We consider this work to be of paramount importance because a thinking through of the many possibilities and a detailed consideration of some of them is a necessary precursor for the work of the whole Decision Unit. We believe additional emphasis should be given to this program.

We are concerned that information required for rulemaking proceedings will not be available unless special attention is given to early definition of the needed research associated with this program. We are indicating our evaluation of the importance of the work in this area by recommending an increase in the proposed budget. We are not sure that this is enough. We recommend that a thorough planning process be carried out by representatives of appropriate units and that programs and needed funds be identified. If the funds currently proposed for this program are inadequate, we recommend that additional funds be made available by transfer from other work within this Decision Unit.

3.6 Recommendations

We endorse the requested funding level for this Decision Unit. However, we recommend that the funding level for the Accident Mitigation program be increased by $3 million by reallocation within this Decision Unit; these funds should be taken from the program on Behavior of Damaged Fuel.

4. ADVANCED REACTORS

4.1 Introduction

This is a new Decision Unit, devoted solely to research on two types of advanced reactors: Liquid Metal Fast Breeder Reactors (LMFBRs), and Gas-Cooled Reactors (GCRs).

4.2 LMFBR Research

We have been informed by RES and NRR personnel that they expect NRC to be requested to reinitiate licensing of the Clinch River Breeder Reactor (CRBR) in the near future. Further, they believe that the Department of Energy (DOE) is moving clearly in the direction of a long-range LMFBR program and that development of a plant larger than CRBR will follow the latter in a few years. In view of the apparent imminence of CRBR licensing, RES has developed a phased plan for expansion of the LMFBR safety research program during FY 1982 which will bring the rate of expenditure to a level of approximately $20 million per year by the start of FY 1983. This plan involves maintaining generic work initially at a level of about $8 million per year, with the expansion of the program to be based upon detailed studies by RES and NRR jointly to identify NRR needs for licensing CRBR. Some of these needs are expected to involve acceleration of current work; however, we believe that the bulk of the expansion should involve new work such as core melt accident evaluation using risk analysis and other developing techniques, systems analyses, operational behavior, control system dynamics, relief valve behavior, and environmental effects on equipment.

We recommend a budget level of about $20 million for FY 1983 to carry out initial work under a plan to be developed jointly by NRR and RES defining safety research needs for CRBR licensing and follow-on licensing programs. We believe that at least half of the $20 million expenditure should be devoted to work in new areas like those identified above.

We believe also that much more attention should be devoted to long-range planning in LMFBR research than has been the case in the past. For example, there are no established and agreed-upon safety-related design criteria within NRC for LMFBRs and, without such criteria, licensing reviews are more ad hoc and it is difficult to define needed research programs, their costs, and their schedules. Further, current research work is not dealing with all of the topics we believe important; it may not be addressing the most urgent ones, and the level of expenditure may be quite unrealistic. Therefore, we believe that NRC should move promptly to initiate design-criteria studies for demonstration-size LMFBRs, to define safety research needs for such units, and to define an appropriate safety research program. Such a program should incorporate the views of NRR, RES, DOE, nuclear steam system suppliers, the utility industry, and independent experts. The program should be coordinated with ongoing safety research being performed by DOE as well as by foreign groups with which the NRC has cooperative agreements, and should include topics such as the following:

- What are the safety issues on which research is needed?
- How is each issue to be resolved?
- Who should do this work? NRC? DOE? Industry? Other?
- When is the result needed and why?
- What is the estimated cost to resolve the issue and is it practical to finance the effort?

Insofar as practical considerations and timing will permit, the results of this effort should be applied also to CRBR.

4.3 GCR Research

The current safety research program on GCRs is devoted largely to support of safety evaluations related to the Fort St. Vrain high-temperature gas-cooled reactor (HTGR). However, this program will near completion in FY 1982, and the proposed program for FY 1983 is based primarily on the anticipation that further development of the HTGR may lead to a license application within the next several years. On this basis, the proposed program and funding level is adequate, and we propose no changes for FY 1983.

4.4 Recommendations

We recommend that the research programs in this Decision Unit be funded at a level of $22.5 million; $20 million for LMFBRs and $2.5 million for GCRs.

5. REACTOR AND FACILITY ENGINEERING

5.1 Introduction

This is a new Decision Unit that has resulted from restructuring the research programs to reflect the consolidation of RES and the Office of Standards Development functions. This Decision Unit deals with programs pertinent to plant and facility design engineering and component qualification. Specific comments on these programs follows:

5.2 Mechanical and Structural Engineering

5.2.1 Mechanical Engineering

The Mechanical Engineering Safety Research Program was organized in FY 1981 to develop a better technological basis for safety regulation as the aftermath of recent experience with less-than-satisfactory performance in safety-related equipment of a mechanical engineering nature. The intent is to strengthen regulatory understanding of engineering practices and develop probabilistic approaches to the treatment of some engineering questions.

We support the general usefulness of this program. The level of funding appears reasonable for the objectives but the portion invested in computational procedures seems excessive. Experience with piping systems, especially seismic restraint devices, indicates that improvements in equipment by better design, maintenance, and inspection are needed if the credited safety capabilities are to be realized. The requirements for equipment qualification are not well defined and considerable work is necessary to establish a basis for regulations. Methods of checking computational procedures are needed in some areas. The capacity and capability of safety and relief valves under the anticipated working conditions of gas, liquid and two-phase flow operation are particularly important as shown by recent discussions concerning ATWS regulatory requirements and experiences at TMI-2 and Crystal River.

The combination of dynamic loads effort appears to be concentrated on complex computer analysis techniques that may not be usable because of insufficient data base and the need to use gross assumptions of structural behavior in the analytical techniques. This effort would be better applied in determining how inherent capabilities of materials can be used to compensate for errors and to off-set overly conservative assumptions in combined load analysis, e.g., factors that limit the size of double-ended pipe breaks, ductile response of piping, rate effects of fluid release, and structural damping in seismic events.

We endorse the effort to participate in and capitalize on foreign research programs being pursued concurrently in these areas. The interchange of information with experts in other countries having a different perspective will broaden the understanding of mechanical engineering applications and reduce the likelihood that important safety issues will be overlooked.

The RES Staff should examine some elements of the program to see whether its perception of need is consistent with that of the anticipated users of the information. Notably, the project on Effects of Nonsafety Systems which is planned to start in later fiscal years, relates to Task Action Plan A-17 (Ref. 3) whose results would have been reported before the proposed program is initiated. At the same time, much of mechanical component qualification requirements will have been instituted in NRC regulations long before the proposed research is completed. This comment should not be interpreted as suggesting a lack of need for the proposed work, but rather a need to coordinate the research schedule with the application schedule.

The RES Staff should evaluate also whether it is appropriate for NRC or industry to finance these proposed work areas. We support the value of some duplication of industry research to assure the availability of independent results and of consulting expertise. Work planned for the Safety/Relief Valve Test Program fits this criterion. We question the advisability of work done apart from the industry activity or as a substitute for work rightfully the responsibility of the industry. Snubber qualification, for example, might more appropriately be sponsored by the industrial suppliers and the industry users.

5.2.2 Structural Engineering

We would assign relatively high priority to the programs on Load Combinations for Design of Structures and to the International Cooperation Program. The latter shows promise of providing knowledge from tests or from actual experience that will be valuable in assessing the accuracy and effectiveness of seismic design procedures.

We do not support the program on Effectiveness of Quality Assurance and Inspection Procedures with its present scope. We believe that an analysis of construction errors or deficiencies to evaluate their relative contributions to risk would serve to focus this program on those areas where improved QA or inspection would be of greatest benefit.

We do not support that portion of the program on Dynamic Testing and Damage Assessment that deals with the assessment of post-earthquake or post-accident structural damage. It does not seem likely that the extent or type of damage that will actually occur can be forecast without an exceptionally comprehensive and expensive program, the cost of which cannot be justified by its relatively small potential contribution to risk reduction. That portion of the program concerned with the evaluation of analytical results by comparison with the results from experiments has merit but should involve also comparisons with results from experience with actual structures. Comparisons of analytical results produced by one computer code with those produced by another has little merit; this is being done in the Soil-Structure Interaction Program and has been proposed for others. It is time to seek data from experiments or experience with which to compare the results from several of the complex computer codes now being used to analyze and design structures for seismic and other loadings.

We understand that the Commissioners have disapproved the proposed contract for Benchmarking of Computer Codes Used in Structural Design as being too ambitious, among other reasons. We understand also that NRR has indicated no strong desire for an in-house code to be used to validate those used by the industry. We have no quarrel with these decisions, but we do believe that there is a need for the licensing staff to have greater confidence in the ability of these codes to yield correct results and lead to satisfactory designs. Correctness in this case means not only correctness of the algorithms and correctness of the mathematical models but correctness of the result in comparison with the best available physical evidence of how structures actually respond to loads. We believe that this can be done by comparing the codes with "standard problems" based to the extent possible on tests or measurements on real or model structures. These comparisons should be made not by the NRC but by the architect engineers, vendors, licensees, or applicants, as appropriate. NRC research should include determining whether such validation is possible and what its limits might be, selecting standard problems, and developing of instructions and requirements for validation and criteria for evaluating the results. In other words, NRC research should be aimed at providing the NRC Staff or its consultants with the ability to ask the right questions and to evaluate the answers. This approach should lead to a significant reduction in the scope and cost of the program. Further reduction could be made by reducing the scope to include initially only one of the three task areas originally proposed; i.e., Seismic Analyses, Containments, or Category I Structures. And finally, since it may not be possible to transmit all of the expertise gained from this project to the NRC Staff, although this should be done to the greatest extent practicable, it may be desirable to select a contractor that would be available to serve as a consultant to the Staff in connection with subsequent uses of the code validation procedures.

The current consideration of accidents leading to degraded or melted cores requires the ability to predict the conditions of pressure, temperature, etc. for which the fission products will be released from the containment in amounts and at rates that will produce a threat to the public. The program on Safety Margins for Containments addresses this problem but in a manner we consider wrong and unnecessarily expensive. This program seems to be intended to provide the NRC Staff with the means to predict containment behavior, including presumably both "failure" and leak rate, for all of the large number of different types of containments that have been or are being built. We believe that the responsibility for computing and justifying containment behavior and capacity should be placed on the applicant or licensee. The NRC, of course, must have the knowledge, background, and expertise to direct and evaluate the computations or demonstrations required, and the research program should be designed to provide those abilities. The NRC Staff should be able to define, on the basis of their consequences, those failure modes and limit states that must be calculated and probably should translate those limits into structural phenomena. The NRC Staff should have the knowledge to request calculations appropriate to the various types of containments and to evaluate the results for each specific type. The NRC Staff need not and should not attempt to develop methods of analysis appropriate to each of the many types of containments, but it may be desirable to do so for a very limited number of cases in order to develop the expertise needed to direct and review the licensee or applicant program. Similarly, the NRC Staff need not be responsible solely for tests to verify the various analyses, but might do some testing to develop questions and to develop experience needed to evaluate the predictive capability of the analyses. The NRC research in this area should have the active collaboration of personnel from the Division of Risk Analysis (DRA), since a probabilistic consideration of leakage rates and failure modes is important.

Those research programs not discussed here are considered of average priority and of generally acceptable scope.

5.2.3 Seismic Safety Margins Research Program (SSMRP)

It is clear that we need to know whether earthquakes are significant contributors to the risks from accidents in nuclear power plants. It is our understanding that one objective of the SSMRP is to develop a methodology to answer this question.

It is not yet clear how the methodology developed in the SSMRP can or will be used to evaluate seismic risks. For one thing, it appears now that many of the data needed to describe the seismic input, structural response, component fragility, etc., are not only quite site-specific and plant-specific but are subject to considerable uncertainties. The importance of this cannot be evaluated until the planned sensitivity studies are completed, and an evaluation is made of other contributions to uncertainty or possible error.

A second problem relates to the likely usefulness of the current SSMRP methodology. It appears to be very costly to implement and the accuracy of the results may be difficult to assess, aside from the large effects of uncertainties.

There is need for a practical method of assessing the seismic contribution to risk in the probabilistic risk analyses currently underway and those which are likely to be performed at an accelerated rate in the future. However, it is not clear that the current SSMRP will provide the NRC Staff with this capability.

We believe that a research program on probabilistic seismic safety is important. However, we are not satisfied that the current SSMRP is properly structured to meet the needs. We recommend the establishment of a task force that includes representation from NRR and the DRA to develop an appropriate program and scope of research in that area. The work of this task force might result in redirection or reorientation of the SSMRP or in recommendations for different or additional research efforts on probabilistic seismic safety.

The current SSMRP effort dealing with specific reactors has been focused on a PWR. In order to find out whether there are any special seismic safety questions for BWRs, we recommend that the SSMRP undertake a study of such a plant while the task force undertakes its review.

We recommend that additional funds be provided to support the BWR study and the task force study.

5.3 Primary System Integrity

5.3.1 Fracture Mechanics

This is a good long-range program that is providing a sound basis for decisions on the integrity of pressure vessels and piping. The question of thermal shock in pressurized systems represents an important uncertainty for the integrity of the primary system, especially for older pressure vessels. Over the last several years, this program has been developing the techniques and information needed to make informed regulatory decisions in this, as well as other areas.

In the piping area, RES should continue to work with NRR to define programs which will provide an acceptable basis for reducing the number of constraints or supports on piping systems, especially the primary loop, while maintaining adequate safety margins for all plant operating conditions.

5.3.2 Operating Effects on Materials

The major contributor to uncertainties in assuring the integrity of the primary and secondary boundaries are the effects of operating environment, radiation, and water chemistry. This program addresses these issues in a sound, coherent manner.

One of its larger components is the examination of the Surry steam generator by the Battelle Northwest Laboratory. This provides a unique facility for needed NRC work on the verification of nondestructive examination (NDE) techniques. However, it could also be very useful to industry for activities such as training of NDE operators, chemical cleaning studies, etc. We suggest that the NRC limit their exclusive control of the facility to a short period, say two more years, and look into then transferring the facility to private management so that a wider range of work of value to the industry can be performed with the steam generator.

5.3.3 Nondestructive Examination

Periodic inspection of reactor components is regularly carried out to assure that no dangerous flaws are present in the primary coolant system pressure boundary. NRC must be capable of judging how reliable these techniques are and must be able to develop criteria for the acceptability of new techniques. The current programs address these questions at an adequate level.

5.4 Electrical Equipment Qualification

Research in this area includes the qualification testing for LOCA and main steamline break environments, the development of accelerated aging methods, the determination of representative LOCA source terms, and fire protection evaluations.

The proposed program for FY 1983 contains funding for the performance of fire replication tests. In NUREG-0699 and NUREG-0751, we stated that the NRC funding of these tests cannot be justified by the information that will be obtained. We have no new information to change this view. We do not support this part of the program.

Selected TMI-2 instrumentation and electrical equipment will be removed, examined, and tested to obtain a better understanding of the performance of equipment which had been qualified under existing standards. This program will be carried out in cooperation with DOE and industry. We support this work, but believe its value is diminished significantly unless it can be done in the near future.

5.5 Fuel Cycle Facility Safety

This program includes research to provide the licensing offices the competence to evaluate source terms during normal operation of fuel cycle facilities, to develop models for realistic analyses of accidents within such facilities, and to assess the effects of dry storage of spent fuel. Specific attention is directed to analyses of accident scenarios leading to the generation, transport, and release of aerosols from fuel cycle facilities. We agree with the importance of this effort and we support the plans for it.

5.6 Effluent Control and Chemical Systems

This program includes research on improving the accuracy for evaluating the performance of effluent control systems in LWRs and fuel cycle facilities. Included are studies of hydrogen control and mitigation systems. In order to achieve these objectives, a major effort is being directed to obtaining more accurate radionuclide source term data. We endorse these efforts and we are encouraged to note the newly planned associated studies for developing QA/QC procedures for effluent monitoring systems. Inasmuch, however, as studies of Licensee Event Reports (LERs) from operating nuclear power plants have shown a relatively high frequency of failures in the equipment installed to monitor the performance of effluent control systems, we recommend that a portion of the funding for this program be directed to research on the evaluation and correction of these failures. Data show this to be a continuing problem with no signs of improved performance in recent years.

5.7 Decommissioning

This program supports research designed to establish and validate the criteria by which the licensing offices will determine whether nuclear power plants and other facilities have been satisfactorily decommissioned and their sites can be released for unrestricted public use. We reiterate our recommendation in NUREG-0657 (Ref. 5) that features of design that will facilitate later decommissioning of nuclear facilities should receive priority within this unit. The current program does not reflect this emphasis. In addition, we suggest that there is a need for a greater effort, on the part of the licensing staff, to incorporate into their rules and regulations the benefits of the research that has been conducted in this subject area in the past.

5.8 Recommendations

We recommend that the FY 1983 funding for this Decision Unit be increased by $2 million, from $38.2 million to $40.2 million, to help fund a more comprehensive safety research program on reactor aging effects and to permit augmentation of the work in probabilistic seismic safety, including study of a BWR.

6. FACILITY OPERATIONS AND SAFEGUARDS

6.1 Introduction

This Decision Unit represents a new grouping of programs as a result of the reorganization within RES; it currently includes programs on: Human Engineering and Man-Machine Interface, Plant Instrumentation and Control, Occupational Protection, Emergency Preparedness, and Safeguards.

This Decision Unit encompasses programs related to many phases of facility operation and control which we believe require more immediate attention. Particular new emphasis is recommended for the following:

- Plant operational behavior as a function of design and control.
- The impact of control systems and other nonsafety systems on safety.
- The need for improved radiological protection.
- Design features to prevent sabotage.
- Ability to respond to sabotage attempts.

Specific comments on these programs follow.

6.2 Human Engineering and Man-Machine Interface

The currently outlined program for research in the area of human factors includes studies of human error rate, reviewing control room design from a human factors perspective and enhancing operator selection, training, and performance. The research in the man-machine interface area will include information control and display design and evaluation; human reliability and human performance standards; procedures development and analysis; and computer-based aids to human performance.

The goal of the research in this area is to improve NRC's basic understanding of the impact of humans on reactor safety and of factors that affect the performance of the man-machine system. We have previously identified this program as a high priority area and suggested areas for additional research in NUREG-0751. The ultimate objective of this research is to reduce the human contribution to risk to an acceptably low level. The current program seems to be a reasonable attempt to address the concerns regarding human factors which resulted from the TMI-2 accident.

The formation of a program plan by the Human Factors Society for RES appears to be an acceptable method of assuring that those areas of research, deemed to have the most risk reduction potential, are addressed first. We continue to believe that this should be an area of expanding research over the next few years and believe that the funding levels proposed represent the minimum amount appropriate for this area. It is still our belief that additional areas of research as outlined in NUREG-0751 should be pursued and that commensurate funding be made available.

6.3 Plant Instrumentation and Control

A program on improved plant instrumentation was begun in FY 1980 in response to various recommendations that improved information on plant conditions during abnormal occurrences was needed by operators. Work is being conducted on instrumentation systems for the detection of inadequate core cooling. Current emphasis is on that part of the system that indicates liquid level in the pressure vessel. Assessment will be made of instrumentation and of qualification programs proposed in response to Regulatory Guide 1.97, Revision 2 (Ref. 6) requirements. Work on noise diagnostics is scheduled to be carried into FY 1983. We believe that this work is directed toward a useful objective. Work is also being planned to examine the problems associated with electromagnetic pulse and lightning protection.

Evaluations will be performed to develop an improved understanding of and solutions for problems encountered with operating safety-related instrumentation and electrical equipment. Existing operating experience will be used to identify problems associated with equipment of this type, and to develop regulatory recommendations. We believe that this work can reduce the risk associated with plant operation and continue to endorse an expansion of this effort.

In addition, there are plans to evaluate the safety implications of control systems. We endorse this activity and recommend that its funding level be augmented and that it include active participation by members of the DRA in order to accomplish an appropriate interaction between the deterministic and probabilistic approaches.

6.4 Occupational Protection

Recent data and projections for the future show a continuing increase in the collective occupational doses associated with the operation of commercial nuclear power plants. Whereas a few years ago, the generally accepted value for a single power plant was about 500 person-rem per year, the latest tabulation published (Ref. 7) by the NRC Staff showed that the average collective dose per operating unit increased by 19% between 1976 and 1979 and now approximates 600 person-rem per year. The collective dose per megawatt-year increased by 30%. Projections are that some plants will have total collective doses of as much as 5,000 person-rem for 1982, and that average values will continue to increase dramatically over the next few years.

According to the Long Range Research Plan (Ref. 8) there is a need for development of a better understanding of the "corrosion, erosion, transport, and deposition phenomena within the primary coolant system and the effects on these of changes in design, materials of construction, quality control, housekeeping practices, and operations." The Plan states also that "there are few methods or data for analyzing the performance and reliability of LWR decontamination systems or for evaluating the net contribution (or reduction) they might make to occupational exposure."

The continued increase in collective occupational doses at operating nuclear power plants makes research in these areas of progressively greater interest. We believe that it should be given priority.

In terms of assessing current occupational exposures, we are encouraged by the research activities planned for improving the QA/QC aspects of personnel monitoring systems. This includes attention to performance testing of survey instruments and personnel monitoring systems as well as bioassay procedures. We endorse these efforts.

In terms of nuclear power plant personnel, consideration also needs to be given to studies to determine the optimum staffing from the standpoint of maintenance, operation, and minimization of individual and collective doses.

In view of the need for greater emphasis on the control of collective occupational doses and on the associated newly proposed QA/QC programs, we recommend that funding for this program be increased.

6.5 Emergency Preparedness

Included in this program area are evaluation and testing of emergency instrumentation for assessing nuclear power plant radiation levels and releases under accident conditions; evaluation of public warning systems; study of human factors affecting the response of the plant staff and the general public during an emergency; the effect of site selection on emergency preparedness; and the efficacy of countermeasures, recovery, and mitigation actions to be applied in the event of an emergency.

Although research in this program has been considerably expanded in recent years, several areas remain that need to be addressed in a more vigorous manner. These include the assessment of groundwater contamination in the event of a serious breach of containment (currently proposed at an FY 1983 funding level of $0.1 million), and the development of methods for the restoration of lands contaminated in the event of accidental radionuclide releases. Both of these subjects have implications relative to the siting of nuclear power plants.

Although we accept the proposed reduction in the requested level of NRC funding for this program for FY 1983, we do so only with the understanding that much of the research previously conducted by NRC in this subject area will now be handled by the Federal Emergency Management Agency (FEMA). We urge the NRC to support FEMA in its request for funds to assume these activities.

6.6 Safeguards

This program is composed of three segments: (1) Materials Control and Accounting, (2) Physical Security, and (3) Threat and Strategy.

6.6.1 Materials Control and Accounting

Funding for this program will be drastically reduced. Emphasis will continue to be placed on improved material management capabilities including increased attention to determining the amount of material held up in processing equipment.

6.6.2 Physical Security

The emphasis of the effort for physical security will be devoted to applying techniques already developed for use in the licensing and regulatory process. Emphasis will be placed on special studies which will help to provide answers to safeguard problems and to resolve safeguard issues.

6.6.3 Threat and Strategy

The work in this area is to develop appropriate responses to threats or appropriate actions in the event of successful sabotage or theft. Human factors aspects of the interactions between the adversary and safeguards systems will be addressed.

We recommend that the work in this Decision Unit include those studies necessary for the NRC to be able to develop requirements with regard to design to protect against sabotage by an insider. This program should, if possible, be completed by the end of FY 1983.

6.7 Recommendations

The funding level for this Decision Unit should be increased by $3.0 million, from $16.8 million to $19.8 million, with emphasis as outlined above.

7. WASTE MANAGEMENT

7.1 Introduction

This Decision Unit includes research on the safety problems of handling and ultimate disposal of high and low level radioactive wastes and uraninum mill tailings. The safe disposal of all of these types of wastes has been and continues to represent a major public concern in the application of nuclear energy for large-scale power generation.

7.2 High Level Waste

The objectives of this program are to identify failure mechanisms that affect long-term waste isolation capability; to identify technical requirements that may be needed to mitigate the consequences of accidental or unplanned movement of radionuclides; and to define the uncertainties or confidence levels in the data, analytical methods, and predictions for each of these areas of concern.

One of the major controversies in determining criteria for assuring the retention of deposited radioactive wastes over a period of thousands of years is the lack of definitive data to verify projected release and migration rates. We are encouraged by plans of the NRC Staff to gain insight into these problems through studies of the migration of naturally occurring radionuclides in various soil structures. Full advantage should be taken of this approach.

Other studies deserving emphasis include the development of techniques for monitoring conditions within a repository in which waste containers have been placed, evaluations of the potential impacts of short-term climatic changes on waste isolation factors such as the direction and rate of groundwater flow, and an assessment of the requirements for the disposal of transuranic wastes (these need not necessarily require the same approach as proposed for application to high level wastes). We also urge that increased attention be directed to research to provide basic technical data, in addition to research of an applied or confirmatory nature.

In the past we have supported the high priority given to this program by the NRC Staff. We believe that research work on high level waste handling and disposal should be vigorously pursued so that the necessary technical information is made available on a timely basis for decisions regarding licensing and regulatory activities.

7.3 Low Level Waste

In NUREG-0657, NUREG-0699, and NUREG-0751, we emphasized the need for sufficient research work to expedite the licensing and regulation of the handling and disposal of low level radioactive wastes. We reiterate that position for FY 1983. The existing situation mandates the selection of new disposal sites within the near future. Research related to the development of criteria for judging acceptability of such sites should be expedited. Acquisition of the technical data to support development of a proposed rule on Low Level Waste Disposal is extremely important. This should be given priority over efforts directed to solving problems at existing low level waste disposal sites. Some research effort should also be directed to exploration of the possible disposal of low level wastes in sites at an intermediate depth below the earth's surface, as contrasted to shallow land burial.

A problem area not apparently being addressed in the research program currently proposed is the treatment and disposal of wastes resulting from the decontamination of systems within operating reactors and of facilities, such as TMI-2, that become contaminated as a result of an accident. Because of the possible presence of chelating agents, such wastes may require special handling and disposal procedures. We recommend that these problems be addressed.

The volumes of low level radioactive wastes currently being generated at commercial nuclear power plants are far larger than they need to be. This is because such wastes contain large amounts of nonradioactive wastes as well as void spaces. Although one approach to solving this problem is to locate additional waste disposal sites, we believe that more attention needs to be addressed to solving this problem at its source.

7.4 Uranium Recovery

The disposal of uranium mill tailings which result from uranium recovery and concentration operations has long been a public concern. We agree that the work to develop criteria for dealing satisfactorily with the large number of existing uranium mill tailings and to provide early guidance for the licensing and regulation of new mills warrants the amount of funding requested.

In reviewing specific research projects in this subject area, we were disappointed in the lack of studies directed toward solving the basic problem as contrasted to those designed merely to ameliorate releases from existing tailing piles.

7.5 Recommendations

The proposed budget for research on Waste Management for FY 1983 is less than that proposed for FY 1982, with a major reduction being made in the area of High Level Waste. We believe that with proper rearrangement and consolidation, coupled with some shifting of funds among subelements, the proposed reduction can be accommodated.

8. SITING AND ENVIRONMENT

8.1 Introduction

Research in this program area includes studies of the Earth Sciences (i.e., seismology, geology, meteorology, and hydrology), Siting, Health Effects, and Environmental Impacts.

8.2 Earth Sciences

This program is devoted to research in the areas of seismology, geology, meteorology, and hydrology. We believe that these studies are of considerable importance to the development of a new rule on Siting (10 CFR 100), to the improved seismic design basis for future nuclear facilities, to an improved assessment of the seismic safety of existing facilities, to improved techniques for the evaluation of the potential effects of flooding on nuclear power plants, to the improvement of the models used in the CRAC Code, and to several other important problems facing the NRC. A major portion of this work is devoted to developing a better understanding of the seismic and geologic behavior of several important regions of the United States. We continue to support this work and believe it is responsive to our previous recommendations.

8.3 Siting

This program pertains to research on improving methods for evaluating alternate sites for nuclear facilities, on obtaining additional information on the relationships of population density, land-use patterns, and siting alternatives for incorporation into the nuclear facility site-selection process, and on providing validated technical bases for standards and siting criteria. This work will include studies to provide input for developing criteria for assessing single versus multi-unit sites, for evaluating changes in land-value and -use patterns and population density, and for providing the wide ranging types of information necessary for support of the rulemaking on Siting.

As would be expected, work in several other Decision Units will provide data applicable to nuclear facility siting problems. This includes improvement in the CRAC Code, advances in the development and application of risk assessment techniques, the development of minimum requirements for engineered safety features, and the establishment of safety goals.

Because of the fundamental importance of siting in assuring adequate protection of the health and safety of the public, we believe this program fully justifies the increased funding proposed for FY 1983.

8.4 Health Effects

The objectives of this program are to improve the understanding of the relationship between exposure to radiation and the magnitude of the biological effects produced, to provide information on the metabolism of inhaled and ingested compounds containing radionuclides not previously investigated, to improve the methodology for predicting deaths and illnesses as a result of radiation exposures, and to evaluate the effectiveness of protective actions and of instruments to detect and measure radiation. Some of this work will help support the current effort to evaluate, for possible incorporation into NRC regulations, the recommendations of the International Commission on Radiological Protection (ICRP) as expressed in its Publications 26, 29, and 30 (Refs. 9-11). The proposed research on the biological effects of neutrons is needed to evaluate the significance of neutron exposures at nuclear power plants.

Although, in the past, studies have been directed to the development of techniques for increasing the removal of radionuclides ingested or inhaled by radiation workers, we note that such efforts are no longer included in this program. We recommend that consideration be given to correcting this deficiency.

In NUREG-0751, we recommended that the work in this program be reviewed, evaluated, and coordinated with the National Institutes of Health Research Committees, the National Council on Radiation Protection and Measurements (NCRP), the National Academy of Sciences, and other federal agencies (e.g., DOE and EPA). We are encouraged to see progress in this area, especially the close cooperation being developed with the NCRP.

8.5 Environmental Impacts

This program includes laboratory and field studies to provide a better understanding of the movement of radioactive releases through aquatic environments, including their transport in rivers and in coastal zones, and their ultimate passage through ecosystems and food chains to people. Associated studies include the development of mathematical models to simulate such transport, including the dispersion and diffusion of radionuclide effluents and the associated effects of sediment deposition and resuspension. Also included is research to provide more quantitative methods for predicting the socioeconomic impacts of such releases, both under normal conditions and accidents. Such work is required to provide the NRC with the necessary tools for assessing the environmental impacts of nuclear power plants as required by NEPA and to evaluate and incorporate into NRC regulations newer recommendations as published by the ICRP and the NCRP.

8.6 Recommendations

Overall, we believe that the proposed level of funding for this Decision Unit is appropriate.

9. SYSTEMS AND RELIABILITY ANALYSIS

9.1 Introduction

This Decision Unit now has a somewhat changed scope from that included in the Decision Unit, Systems and Reliability Analysis (SARA), as discussed in NUREG-0751. Human error data analysis is now formally included in the new Decision Unit, Facility Operations and Safeguards, rather than in SARA, and the program on Transportation and Materials Risk has been moved into SARA.

In NUREG-0751, we recommended a change in emphasis and a considerably expanded research program in this Decision Unit for FY 1982, as follows:

"We recommend that this Decision Unit be allocated at least $23.9 million for FY 1982, an increase of $9.0 million over the proposed budget, and that this additional funding be allocated approximately as follows:

(a) A large increase in emphasis and resources for the task on alternate decay heat removal systems, including consideration of sabotage, enabling enough effort to provide a basis for regulatory decisionmaking no later than the end of FY 1982 ($2.0 million).

(b) A very considerable acceleration in the development of information needed to estimate the likely effect on risk of various potential design changes intended to mitigate accidents leading to severe core damage or core melt in LWRs ($1.25 million).

(c) The development of accident precursor screening techniques and their extensive application to the existing operating plants ($1.0 million).

(d) The early development of a focused, cohesive program to provide the information needed to determine the appropriate regulatory approach to control systems and to information needs of the reactor operator ($1.0 million).

(e) Critical review and evaluation of probabilisitic analyses and risk studies performed by licensees and construction permit holders ($1.0 million).

(f) An examination of possible weaknesses in the current application of the single failure criterion, and the early development of an improved approach ($0.5 million).

(g) The development of a basis for an improved approach to minimizing significant design errors ($0.5 million).

(h) A systematic approach to possible design steps to reduce the potential for serious accidents which might be caused by sabotage by an insider ($0.5 million).

(i) A program to better define property damage from accidents involving large releases of radioactive materials, including the effect on societal resources ($0.5 million).

(j) A critical evaluation of the merits of LWR regulatory requirements in other countries which differ significantly from those of the NRC ($0.5 million).

(k) The early development of quality assurance (QA) criteria for probabilistic analyses to be used in the regulatory process ($0.25 million).

We recommend that the matters listed above be given priority, even if it means reducing the funding for other programs, ongoing and proposed, in this Decision Unit."

The NRC Staff has described programs which indicate that an increased emphasis will be given in SARA or in other Decision Units to many of the items listed above, particularly items (b), (c), (e), (f), (g), and (k). However, on most of these items, the level of support in FY 1983 is much less than that recommended by us in FY 1982. The FY 1983 program provides a negligible or far from adequate effort on items (d), (h), and (j) and for a reasonable amount of effort on item (i). With regard to item (a), a substantial effort on decay heat removal systems is contemplated as part of the Task Action Plan A-45; however, this effort is aimed primarily at existing plants. We believe that the DRA should participate significantly in this work and in addition should carry on a more general investigation, the results of which can contribute to setting design requirements for future plants.

NRR has stated its plans to build up its capability to apply probabilistic methodology to licensing matters, a step we strongly support. However, we do not believe that the buildup of this capability in NRR will be swift enough or large enough by FY 1983 that there will not remain a continued need for DRA to apply probabilistic risk assessment (PRA) methodology during this time period, especially in a peer review mode. Furthermore, our recommendations for expanded research programs in the SARA Decision Unit generally do not fall in the category of licensing applications of PRA methodology, but rather involve the use of PRA methodology to develop information which can form a part of the basis for generic regulatory decisionmaking. Hence we believe that funding greater than that recommended by the EDO should be provided for SARA in FY 1983.

We recommend also that positive steps be taken by RES management to ensure that the needed close collaboration between DRA and the other research divisions occurs.

9.2 Risk Methods and Data Evaluation

We support the overall level of funding proposed for this program but recommend some reorientation of the work. We believe that the effort which relates to common cause and external events should be substantially increased. We recommend also that efforts be instituted on methods to enable inclusion in PRAs of the effects of design errors, sabotage and externally and internally produced flooding. We recommend that the DRA should be asked to review and critique the SSMRP program which is funded under Decision Unit 5, "Reactor and Facility Engineering." We also recommend that work be initiated on improved methods for analyzing reliability requirements for control systems and on additional screening techniques for accident precursors.

9.3 Reactor Risk and Reliability Analysis

We endorse and recommend much increased support for the proposed research in support of a Degraded Core Cooling Rule and on a much broadened interpretation of minimum engineered safety features. We also support the proposed research on regulatory analysis, on IREP/NREP, on accident sequence analysis, and on consequence analysis.

As discussed in Section 9.1, we believe that there are several items that are missing from the program or are much too weak. We are recommending additional funding for this program specifically to deal with those items and to provide expanded support for the major rulemakings referred to above.

One of the key items in the codes used for assessment of accident consequences is the model used to estimate the impacts on the neighboring population. Currently, the mainstay of the analyses used for this purpose is the CRAC Code. Although efforts are being made to improve this Code, we are concerned about the pace of progress. Since this Code is important to emergency preparedness, probabilistic risk assessment, and the support of the rulemaking on Siting, we believe that efforts for its improvement should be expedited. At the present time, the Code is deficient in terms of the range, depth, and flexibility of countermeasures that can be considered, and in terms of the assessment of the health effects of various exposure rates and levels. There is also a need to develop an ancillary code for evaluating the impacts of releases via the liquid pathway. We recommend that this work be given the needed support.

9.4 Transportation and Materials Risk

This program includes research on risks in the fuel cycle, in transportation of irradiated fuel, and from nuclear materials. We believe that, while work in this area is useful, it is generally of lower priority than the proposed research in the program on "Reactor Risk and Reliability Analysis" (Section 9.3), as well as many of the recommendations presented in Section 9.1 of this report.

9.5 Recommendations

We recommend that the FY 1983 funding for the SARA Decision Unit be increased by \$4 million, from \$21.7 million to \$25.7 million, and that these additional resources be used to implement our recommendations.

APPENDIXES

APPENDIX A

REFERENCES

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*Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and/or the National Technical Information Service, Springfield, VA 22161.

APPENDIX B

GLOSSARY

ATWS	Anticipated Transients Without Scram
BWR	Boiling Water Reactor
CCTF	Cylindrical Core Test Facility
CRAC	Calculation of Reactor Accident Consequences
CRBR	Clinch River Breeder Reactor
DECCA	Deformed Core Coolability
DOE	Department of Energy
DRA	Division of Risk Analysis, RES
ECC	Emergency Core Cooling
EDO	Executive Director for Operations
EPA	Environmental Protection Agency
FEMA	Federal Emergency Management Agency
FIST	Full Integral Simulation Test
FRG	Federal Republic of Germany
FY	Fiscal Year
GCR	Gas-Cooled Reactor
HTGR	High Temperature Gas Cooled Reactor
ICRP	International Commission on Radiological Protection
IREP	Interim Reliability Evaluation Program
LER	Licensee Event Report
LMFBR	Liquid Metal Fast Breeder Reactor

LOCA	Loss-of-Coolant Accident
LOFT	Loss of Fluid Test
LSTF	Large Scale Test Facility
LWR	Light-Water Reactor
MARCH	Meltdown Accident Response Characteristics Code
NCRP	National Council on Radiation Protection
NDE	Nondestructive Examination
NEPA	National Environmental Policy Act
NRC	Nuclear Regulatory Commission
NREP	National Reliability Evaluation Program
NRR	Office of Nuclear Reactor Regulation
ORNL	Oak Ridge National Laboratory
PBF	Power Burst Facility
PCI	Pellet-Clad Interaction
PKL	Test Facility In Germany designed to model plant systems behavior during Loss-of-Coolant Accidents and Transients
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
QA/QC	Quality Assurance/Quality Control
RELAP-5	Advanced System Code used to model Loss-of-Coolant Accidents
RES	Office of Nuclear Regulatory Research
SARA	Systems and Reliability Analysis
SCTF	Slab Core Test Facility

SFD	Severe Fuel Damage
SSMRP	Seismic Safety Margins Research Program
THTF	Thermal Hydraulic Test Facility
TLTA	Two Loop Test Apparatus
TMI-2	Three Mile Island, Unit 2
TRAC	Transient Reactor Analysis Code
UPTF	Upper Plenum Test Facility

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