UNITED STATES
NUCLEAR REGULATORY COMMISSION
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INFORMATION REPORT

For: The Commissioners

From: Robert J. Budnitz, Director
Office of Nuclear Regulatory Research

Thru: William J. Dircks, Acting Executive Director
for Operations

Subject: ACRS REPORT TO CONGRESS, "REVIEW AND EVALUATION OF THE NRC SAFETY RESEARCH PROGRAM FOR FISCAL YEAR 1981, NUREG-0657"

Purpose: To inform the Commission of RES comments and responses to the ACRS recommendations contained in the subject report.

Discussion: The Congress requires the Advisory Committee on Reactor Safeguards (ACRS) to submit annually a report to Congress on the NRC's reactor safety research program. This report is a detailed review of the Office of Nuclear Regulatory Research's safety research program and includes not only a technical review of programs, but also comments on the level of funding for each program. This is the third annual report of this type. This report series is different from the annual ACRS review of the RES budget, the first of which was sent to the NRC Commissioners in July 1979 (NUREG-0603).

The ACRS, on instructions from Congress, attempted to prioritize the entire safety research program in this report; however, assigment of priorities among decision units was found to be difficult. Nevertheless, the ACRS did assign priorities in several areas. Program areas of highest importance are indicated by suggested changes in the level of funding recommended by the ACRS. The RES Director and RES staff concur with the principal findings and recommendations in Chapter 1, and are in general agreement with ACRS's comments and priorities; however, there are some areas of disagreement.

Following is a discussion of the ACRS' major findings and recommended areas for new directions which are summarized in Chapters 1 and 2 of the report. Included in this discussion are RES's comments on important areas of disagreement. The enclosure to this information paper includes a lengthy discussion of all the ACRS recommendations and comments in NUREG-0657, including for completeness some items discussed in this covering paper.

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1. The ACRS emphasizes three research areas related to the accident at TMI-2 that they feel warrant greater emphasis than is currently planned by the NRC staff. These areas are: (i) studies of the courses of serious accidents; (ii) studies of molten core retention and steam explosions; and (iii) studies of plant operations and of systems behavior, particularly in various stages of post shutdown heat removal.

(i) RES agrees with the ACRS that these three areas warrant greater emphasis. We are increasing the effort on programs related to understanding the phenomena important to assessing the course of postulated core melt accident sequences, including interactions of molten core materials with plant structures and with residual water. To study the course of high risk and probable accident sequences, RES has initiated programs at INEL, LASL, and ORNL. These programs will analyze the multiple failures, including operator actions, leading to core damage in reactor accident sequences. This effort will provide analyses of the risk dominant accident sequences identified by the Interim Reliability Evaluation Program (IREP). In addition, the effort will address system interactions and man/machine intervention for accident mitigation and recovery. Recommended design improvements and suggested changes or additions to operating procedures, which will reduce the risk by mitigating or delaying the consequences, will be a major product of these studies.

Even with these important new programs, we in the RES staff continue to be concerned that the response is inadequate. We are now engaged in developing our FY82 budget plans, and in the course of this effort we are identifying some areas that may not be sufficiently addressed in our present FY80 program and FY81 plans. We will study these carefully, and we expect to meet with the ACRS soon to go over our present plans and get their reactions.

(ii) RES has recently developed a long range program plan for fuel melt research. This plan, which has matured significantly in the last couple of months (since the ACRS review), is, in our view, more responsive now to the ACRS concerns in NUREG-0657 than our earlier plans. Our present plan incorporates efforts on core melt accidents being performed within both the Water Reactor Safety Research and Advanced Reactor Safety Research areas, along with the improved reactor safety research work and probabilistic risk assessment work sponsored by the Probabilistic Analysis Staff (PAS). This long range fuel melt research program plan will provide the basis for reorientation of existing efforts and required budget decisions in FY 1981 and FY 1982. The fuel melt program is being coordinated with NRR

under the TMI Action Plan and will be reviewed with the ACRS in the next few months. More detailed discussions of our current plans in the steam explosion and core retention area are included in the enclosure in the response to ACRS Comment 6.3.4.

(iii) RES agrees that more emphasis should be given to the operational aspects of reactor safety, and we are developing a comprehensive program plan for research in this area which we expect to be able to discuss with the ACRS in the coming months. Part of the basis for developing a systematic research program in operational safety will be derived from studies now being performed under two new NRC programs. The Interim Reliability Evaluation Program is assessing which failures are most likely to lead to serious accidents. The newly formed NRC Office of Analysis and Evaluation of Operational Data is identifying and evaluating failures and operational problems as they occur. The consideration of which failures are important, along with what failures are occurring, will clarify priorities for a systematic research program in operational safety. We believe that these efforts respond adequately to the ACRS concern, although in some ways we are just as frustrated as the Committee about how long it seems to take to put together new programs.

The ACRS recommended that a new research program be established in systems behavior and interaction which includes an interdisciplinary approach to safety research. RES agrees that a program of this nature is needed and that we have not moved as fast as the ACRS believes is warranted. The planning and implementation of this new program will be a high priority matter for RES in the coming months.

2. Test results from the Two Loop Test Apparatus (TLTA) have been translated directly to prediction of the behavior of full-scale BWRs. The ACRS believes that this translation to full-scale BWRs is unfortunate and a misuse of the results. The Committee feels that the scaling behavior in TLTA has not been adequately analyzed and using it to predict the performance of full-scale systems can be quite misleading. The ACRS urges strongly that results from this facility not be injected into the licensing path. The ACRS objects, in particular, to the series of small break tests proposed to answer questions raised by the accident at TMI-2. According to the ACRS, the limits of applicability of TLTA test results to full scale-plants should be considered carefully in advance of any such tests, and the test results themselves should be

used for their contributions to code development and to the understanding of the essential physical phenomena. While TLTA has received some upgrade, the ACRS believes that an extensive, further upgrade is necessary and urges that this be pursued.

The RES response is as follows: RES has previously noted that TLTA is a scaled facility and, as such, has scaling compromises which limit its simulation capability. Since the TLTA has historically been used to investigate the blowdown phase of a BWR large LOCA, the facility has been constructed to minimize scaling deficiencies important to the blowdown phase. Certain facility features necessary for good blowdown phase simulation prevent adequate simulation of other parts of the small break LOCA transient and other non-LOCA transients.

The small break tests were originally designed as scoping tests which would have been used to identify scaling deficiencies for small break simulation. The approach has changed significantly. Calculational studies were used to identify anticipated simulation deficiencies and to aid in planning a test program which would minimize, if possible, some of the deficiencies. A number of scaling problems were identified and it was determined that while data useful for code assessment would be obtained, the tests would not simulate the expected BWR transient. These findings were transmitted to NRR prior to the start of the test series (letter from Sullivan to Rosztoczy dated 12/20/79, "TLTA Small Break Test"). We agree with the ACRS conclusion that the results of the TLTA small break tests should not be applied directly to BWR analysis.

We know that the current Two Loop Test Apparatus (TLTA) has facility limitations which prevent testing of the entire large LOCA transient. In addition, a possible need for multiple channels has been identified. A three channel upgraded version of the TLTA is currently being considered which would be capable of testing the entire LOCA transient. The existing TLTA also cannot adequately simulate small LOCA transients or non-LOCA transients due to scaling limitations. The proposed three channel facility would reduce some of the small break scaling problems (area/height ratio) due to its large volume. It appears that a facility can be constructed which can adequately simulate both large and small break LOCA's and other non-LOCA transients. Such a program modification is currently under consideration.

3. The ACRS recommended that the funding level for RES in FY 1981 be increased by $20.0 million in the following areas: (1) Advanced Reactors (Fast Breeders and Advanced Converters) by $13.0 million; (2) Risk Assessment by $2.4 million; and (3) Improved Reactor Safety by $5.5 million. The ACRS noted that it would be undesirable to reduce other program areas to meet these suggested funding levels; however, they strongly recommend the increases in Risk Assessment and Improved Reactor Safety, even if this requires a reduction in other portions of the overall safety research program.

   We agree that an increase of $20.0 million in FY 1981 would be desirable. However, we appreciate that there is an overall limit to the entire NRC budget, and that such additional resources are not likely to be available in FY 1981. Because of the possible need to realign priorities within NRC to accommodate the TMI Action Plan, and because of possible cuts below the President's FY 1981 Budget that are now under consideration in the Congress, we believe that it is premature to reprogram funds at this time. However, once the FY 1981 RES budget is clearly known, we will look carefully at the ACRS recommendation.

4. The ACRS has reiterated its strong support for a meaningful NRC research program related to the safety of advanced reactors. This strong support is contingent upon whether the FY 1981 legislation will or will not continue a program for the development of liquid metal fast breeder reactors. The ACRS indicates that if the decision is made to continue with LMFBR development in this country, then the NRC funding level should be at least $18 million in FY 1981. However, if the decision is made to delay indefinitely the development of LMFBRs, the proposed budget of $5 million is not sufficient to bring the current program to an orderly conclusion. RES agrees with these comments and recommendations.

5. The ACRS gave high priority to research programs in risk assessment and improved reactor safety and recommended a more rapid expansion of these program areas. The importance of using probabilistic risk assessment as a guide to decision-making and licensing requirements was noted. It was felt that the risk assessment program should be funded at $15 million in FY 1981 and improved reactor safety at $10 million in FY 1981. While it is felt by RES that these higher funding levels would be beneficial to both programs, the OMB proposed budget containing increases in these areas will still allow RES

to work at a reasonable rate on the higher priority items in both of these program areas. Whether reprogramming of additional funds into these areas will be desirable or feasible will be decided only after RES obtains its final FY80 appropriation through Congress.

6. The OMB funding level for LOFT of $43 million for FY 1981 is $6.3 million less than the ACRS had commented on favorably in July 1979 (NUREG-0603). The ACRS felt that the higher level would have been beneficial to the program, both from a technical perspective and in the overall cost effectiveness of the program. However, the lower level of funding on LOFT would be acceptable to the ACRS if the level of funding in risk assessment and improved reactor safety were brought up to the level recommended by them.

   The $6.3 million reduction in the LOFT budget for FY 1981 will reduce the number of tests from 9 to 6, and will also reduce INEL's capability to cope with more severe tests which might result in fuel damage. The reduced funding level will also cause a reduction in the scope of man/machine and fault-diagnostics work which was started in FY 1979. More importantly, we believe that the greater efficiency that the higher funding would allow for the LOFT facility is very cost-effective over the long run. Thus, our position at this time is that the additional funds should, if available, be used at LOFT.

7. The information on fuel behavior during reactivity insertion accidents (RIA) is still believed by the Office of Nuclear Reactor Regulation (NRR) to be inadequate. According to the ACRS, it is not clear that this experimental program has provided information of quantity and significance in proportion to its level of support. If these accidents are of sufficiently low risk (low probability and/or low energy insertions), such research is not necessary. The NRC Staff has not provided the ACRS with a convincing argument in favor of the need for the experiments on fuel behavior during RIA, or for most of the other experiments planned for PBF in FY 1980 and FY 1981.

   The ACRS believes that PBF probably can be used for experiments related to flow starvation and fuel melting accidents and urges an early and complete evaluation of the currently proposed PBF program. In the meantime, the ACRS believes that flexibility in reprogramming some PBF funds to other high priority work on steam explosions and core melt should be provided.

RES has attempted to be responsive to the ACRS recommendations for fuel behavior tests in past years. The Committee commented in their 1977 Report to Congress (NUREG-0392) that the current programs on fuel behavior were considered responsive to the safety concerns noted by NRR and the ACRS. In December 1978 (NUREG-0496), the Committee generally agreed with the PBF test program but requested the staff to reassess the priority assigned to RIA tests. In response to this request, the staff did, in fact, reassess the program and greatly reduced the number of RIA tests planned and changed the tests to emphasize higher levels of burnup. In July 1979 (NUREG-0603), the Committee requested that NRR reevaluate the regulatory requirements for RIA tests and suggested that the program be reoriented to emphasize the study of processes leading to medium and severe core damage. In response to this request, the program was reevaluated by RES, NRR, SD, and the PBF Review Group. The conclusion of this review was that, because NRC's regulations require that RIA's be evaluated as part of the licensing review of each plant, the planned RIA tests are needed. NRR is preparing a User's Need letter for these tests, and we will discuss the entire PBF program and priorities with the ACRS subcommittees in the coming months.

RES agrees that the PBF program should be reoriented to emphasize the study of processes leading to medium and severe core damage, and we took action last summer to develop a program plan for severe fuel damage tests. Because of the lead time involved in planning the tests, designing the test trains, and fabricating the hardware and test fuel, these experiments will not be implemented until 1982. It is very important that the fuel damage tests be well planned and thoroughly reviewed by NRC and the technical community. We expect to discuss these tests and seek the Committee's guidance in the coming months.

With regard to the recommendation to provide flexibility to reprogram some PBF funds to other high priority work on steam explosions and core melt, we will investigate the options available. Reprogramming has an impact on the future PBF program that must be carefully evaluated, because planning and fabrication of hardware for the new flow starvation and fuel damage tests will require funds over and above those budgeted for the current test program. The only way to make significant funds available for reprogramming would be to stop testing in PBF until the fuel damage tests are ready, and at this time we do not believe the safety tests planned in PBF over the next two years should be eliminated or deferred.

8. The ACRS commented that the confirmatory programs related to core behavior following a large LOCA should have low priority; that the joint U.S.-Canadian research program at the NRC reactor in Canada should be terminated in FY 1983 as planned; and that before committing to the multi-national research program at the ESSOR reactor complex in Ispra, Italy, the NRC Staff should be convinced that there are not higher priority NRC research needs.

RES disagrees and notes that the NRR audit curves that are now used to predict fuel rod ballooning and rupture and resultant threat to post-accident core coolability after a loss-of-coolant accident (LOCA) are primarily based on out-of-pile separate effects tests which employ electrical heaters that have never yet been shown to simulate accurately the in-reactor ballooning of full length low enrichment commercial reactor fuel rods. Only a few data points based on TREAT and FR-2 tests using short test fuel rods are included in the data base.

The NRU tests are the only in-pile fuel bundle tests which use prototypic, full-length, commercial-enrichment fuel rods to establish the validity of all the out-of-pile test data which currently purport to represent the nature and extent of clad ballooning during the heat-up period of a LOCA. Review group and technical community consensus is that the NRU tests are needed to verify the out-of-pile simulation of in-pile behavior by clusters of electrically heated fuel rod simulators.

The current NRU LOCA program is on schedule and the plans have been to terminate this program in FY 1983. However, we cannot at this time commit to complete our relationship with the Canadian NRU reactor in 1983 until we have assessed the need for possible further tests on long large bundles or even long single rods of highly irradiated commercial enrichment fuel under accident conditions of continuing interest to the NRC.

Regarding priorities within the Fuel Behavior decision unit, we find that support for ESSOR has a relatively high priority. The fuel bundle test program (called Super Sara) proposed for the ESSOR reactor was reviewed by RES and the European Community groups following the Three Mile Island accident and was reoriented to delete some LOCA tests and to add tests which will produce more severe fuel damage by extending core heatup to the temperature of cladding melting. This reorientation of the ESSOR program is in response to the RES and ACRS desire to study the

course of serious accidents. The ESSOR facility will provide, by virtue of its nuclear heat source, length (2 meters), and bundle size (32-36 rods), the more realistic set of thermal hydraulic and cladding oxidation conditions that are needed to understand better the rate and distribution of core damage and the possibilities of stopping the accident before meltdown becomes inevitable. The PBF, while lacking this capability, will complement this approach by focusing on fuel rod melting, geometry changes, and resulting core coolability as mentioned in our earlier paragraphs. The Super Sara program has now been approved by the European Communities as part of their 1980-83 program, and the total cost of the project, including past costs for loop construction, is estimated at $139 million in current dollars. NRC's commitment is to provide three test trains and manpower support in analysis, instrumentation needs and post-irradiation examination. These costs are estimated to be about $1.5-2.0 million per year (in current dollars) for 6 years. Thus because of the high leverage that we have in obtaining important data on severely damaged fuel for a relatively modest cost to NRC, we believe the ESSOR program has a relatively high priority within our Fuel Behavior program.

9. Throughout the report, the ACRS has commented on "new directions in research" which have been or should be incorporated into the proposed research program. Most of these new directions have come to light or have been emphasized as a result of the accident at TMI-2. The ACRS felt that most of these new programs could be accommodated by a reallocation of funds, except in the area of risk assessment and improved reactor safety, for which the ACRS recommended additional funding.

   We agree that the new directions in research recommended in the ACRS report should be incorporated into the research program, and we have attempted to do so. Some additional increase in Risk Assessment and Improved Reactor Safety may also be possible. As mentioned just above, we will study this further during detailed planning of the FY 1981 research program, in the context of overall NRC needs after adoption of the TMI Action Plan.

10. The ACRS noted that because of the rapid growth rate of the Waste Management research program, they are concerned about the ability of the present staff to manage this program area effectively. The ACRS recommended that the NRC be authorized and funded to expand the technical staff

for this program by at least five people above the fiftaen proposed for FY 1981.

We agree that the additional number of technical staff recommended by the ACRS would help to manage the waste management research program effectively. However, as with program support funds we recognize that the overall NRC allotment is limited. We received an OMB-approved staffing level of 11 for FY 1980 and 15 for FY 1981. In filling our FY 1980 staffing allotment, we have added a geochemist during the past few months and will be acquiring a geophysicist shortly. We hope to obtain approval to hire a geological engineer, a systems analyst, and a materials scientist in the near future.

11. The ACRS highlighted the problems associated with the limited nature of "confirmatory research" and the restriction placed on RES of being limited in initiating new work except as requested by a User Office. They also noted that the distinction between "research to improve reactor safety" and "confirmatory research" is no longer useful and recommended that Congress review the legislative charter of the NRC research program and eliminate this distinction.

The problem associated with the restriction on the ability of RES to start new programs except in response to a user request is being addressed in the revised procedures (SECY-79-635) for User Request and Endorsement. The revised procedures will allow RES to initiate programs itself, up to 10 percent of the RES budget, limited to $1 million per program per year. These revised procedures have a trial period of 1 year in FY 1981. More importantly, there has been an attitude change in the User Offices in the last year that will allow RES much more flexibility through better communication and trust.

We fully endorse the ACRS's recommendation to Congress that the distinction between "research to improve reactor safety" and "confirmatory research" is no longer needed. We in RES believe that this is a major issue, and we are gathering our own thoughts together now about what the agency should do to improve our policy in this regard. In order for RES to be more responsive to the agency's mission, it is important that we have the latitude and flexibility to do whatever work will contribute meaningfully to that mission without compromising the credibility or independence of the NRC.

Summary: It is the opinion of RES that the annual ACRS review of our research program is extremely valuable. Indeed, it is the only thorough, broad-based review that the RES program gets. We believe that our interaction with the ACRS is of the utmost benefit and that their reports should be given important consideration. We have attempted to do so ourselves, recognizing that there still remain some areas where our judgments do not entirely agree with those of the Committee. In the enclosure to this Commission Paper, a detailed RES comment is given on what we believe to be the major ACRS statements and recommendations contained in the body of NUREG-0657.

Coordination: There has been no coordination of the paper within other offices. Early drafts of this paper and its enclosure were distributed to the ACRS and the Commissioner assistants on April 4, for use by an ACRS subcommittee on April 9.

Robert J. Budnitz

Robert J. Budnitz, Director
Office of Nuclear Regulatory Research

Enclosure: RES Comments and
Responses to ACRS Recommendations
(NUREG-0657)

ENCLOSURE

RES Comments and Responses to ACRS Recommendations (NUREG-0657)

Note: RES comments begin with Chapter 3. Chapters 1 and 2 are a summary which are commented on in the Commission Information Paper

Chapter 3. SYSTEMS ENGINEERING

3.3.1 Semiscale (page 3-2)

ACRS Statement

The Semiscale program will include studies of transients induced by small breaks, and a survey of transients initiated on the secondary side. The transients initiated on the secondary side are the most common sources of challenges to the ECCS. The ACRS recognizes the value of these programs in Semiscale.

Semiscale is an integrated test facility, but not in the sense that Semiscale data can be translated directly to full-size PWRs. If so translated, the Semiscale data can be misleading, and for this reason the ACRS urges that Semiscale be separated from the licensing path. The important contributions of Semiscale are of two kinds: first, Semiscale tests contribute to the general understanding of the pertinent physical phenomena; and, second, these tests make an important contribution to reactor code development.

RES Comment

We are pleased with the ACRS recognition of the value of the current Semiscale plans.

We agree that data from Semiscale and from other similar experiments should be separated from the licensing path. We also consider that the principal contributions of the Semiscale program are improving code development by a combination of pretest predictions and post-test evaluations of the integrated system response to a range of postulated accidents. These assessed codes are then to be used to perform accident analysis in large commercial PWRs.

3.3.2 Blowdown and Reflood Heat Transfer

ACRS Statement (page 3-2)

A significant facility in this program is the Two Loop Test Apparatus (TLTA) which is an integrated test facility that is presumed to do for boiling water reactors (BWRs) what Semiscale does for PWRs. Test results from TLTA have been translated directly to prediction of the behavior of full-scale BWRs. The ACRS believes that this translation to full-scale BWRs is unfortunate and a misuse of the results. The scaling behavior in TLTA has not been adequately analyzed and using it to predict the performance of full-scale systems can be quite misleading. The ACRS urges strongly that results from this facility not be injected into the licensing path. The ACRS objects, in particular, to the series of small break tests proposed to answer questions raised by the accident at TMI-2. The limits of applicability of TLTA test results to full scale-plants should be considered carefully in advance of any such tests, and the test results themselves should be used as recommended above for Semiscale test results -- for their contributions to code development and to the understanding of the essential physical phenomena.

While TLTA has received some upgrade, the ACRS believes that an extensive, further upgrade is necessary and urges that this be pursued.

Another program in this category is the spray test facility at Lynn, Massachusetts, which is a 30-degree sector of the spray installation in a BWR. Steam-water interaction effects will be studied in this facility and the results will be of importance.

RES Comment

TLTA Scaling Limitations

The Two Loop Test Apparatus (TLTA) is a scaled facility and, as such, has scaling compromises which limit its simulation capability. Since the TLTA has historically been used to investigate the blowdown phase of a BWR large LOCA, the facility has been constructed to minimize scaling deficiencies important to the blowdown phase. Certain facility features necessary for good blowdown phase simulation prevent adequate simulation of other parts of the small break LOCA transient and other non-LOCA transients.

The small break tests were originally designed as scoping tests which would have been used to identify scaling deficiencies for small break simulation. The approach has changed significantly. Calculational studies were used to identify anticipated simulation deficiencies and to aid in planning a test program which would minimize, if possible, some of the deficiencies. A number of scaling problems were identified and it was determined that while data useful for code assessment would be obtained, the tests would not simulate the expected BWR transient. These findings were transmitted to NRR prior to the start of the test series (letter from Sullivan to Rosztoczy dated 12/20/79, "TLTA Small Break Test"). We agree with the ACRS conclusion that the results of the TLTA small break tests should not be applied directly to BWR analysis.

Proposed TLTA Upgrade

The current Two Loop Test Apparatus (TLTA) has facility limitations which prevent testing of the entire large LOCA transient. In addition, a possible need for multiple channels has been identified. A three channel upgraded version of the TLTA is currently being considered which would be capable of testing the entire LOCA transient. The existing TLTA also can not adequately simulate small LOCA transients or non-LOCA transients due to scaling limitations. The proposed three channel facility would reduce some of the small break scaling problems (area/height ratio) due to its large volume. It appears that a facility can be constructed which can adequately simulate both large and small break LOCA's and other non-LOCA transients. Such a program modification is currently under consideration.

CCFL Refill/Reflood Program

This program investigates several aspects of the BWR large LOCA including spray distribution, refill and reflood phases, and certain other phenomena. The program also contains a significant analysis effort and will provide input to the effort to develop a BWR version of our advanced code, TRAC.

This program is now well underway and essentially on schedule. Program task plans are in final stages of development, the spray distribution task testing is about half completed, and testing that will provide input to the refill facility design is underway. GE and INEL code developers are meeting monthly, and INEL has released a preliminary BWR version of TRAC.

3.3.3 3-D Flow Distribution

ACRS Statement (page 3-2)

This large and continuing item is being modified to relate more effectively to present perceptions of some of the most significant problems in reactor safety. The ACRS concurs in its continuation, since the results will be useful and a strong commitment has been in place for several years to participate in this international (FRG-Japanese-U.S.) study of reactor safety features.

RES Comment

The 3D Program has been modified as a result of TMI to include natural circulation tests as well as large-break LOCA and blocked bundle core tests. Although the low pressure design of the 2D/3D facilities limits the applicability of small break tests that can be performed, they are being incorporated into the program to the extent possible. The test logic will be discussed with the ACRS at a future information meeting. Core uncovery tests and natural circulation tests are now in the process of being performed on the PKL facility in Germany. Data are being provided to all the participants. Major commitments to 3D facility construction have been made by Germany and Japan. The NRC is designing and developing advanced two phase instrumentation for measurements in these facilities.

3.3.4 Model Development Program

ACRS Statement (page 3-3)

This program consists of small projects in various university laboratories. The ACRS encourages this kind of program as being useful and productive; at the same time, the program provides an interaction with an important part of the engineering and scientific community.

RES Comment

RES agrees that contributions by university laboratories in basic areas of heat transfer and fluid flow form an important part in the evaluation of improved models of abnormal reactor behavior.

3.3.5 Operational Safety

ACRS Statement (page 3-3)

The past and current research program in Operational Safety was initiated on an ad hoc basis as a result of operating experience or particular regulatory requirements. The ACRS believes that the program to date has been useful. However, in NUREG-0603, the ACRS recommended that the NRC develop a systematic research program on the safety implications of procedures for operation, maintenance, testing and surveillance. In addition, the ACRS recommended that an NRC safety research program on systems behavior should be developed. The ACRS believes that priority should be given to the initiation of a broad research program on operational aspects of reactor safety.

RES Comment

We agree that more emphasis should be given to the operational aspects of reactor safety, and this will be done over the coming months.

The basis for developing a systematic research program in operational safety will be derived from studies now being performed under two new NRC programs. The Interim Reliability Evaluation Program will be assessing which failures are most likely to lead to serious accidents. The newly formed NRC Office of Analysis and Evaluation of Operational Data will be identifying and evaluating failures and operational problems as they occur. The consideration of which failures are important, along with what failures are occurring, will clarify priorities for a systematic research program in operational safety.

For example, one such area now planned for FY 1982 will focus on the electrical equipment that must operate during and following accidents. A survey of specific equipment used in operating plants will be conducted to establish a list of generic designs for this equipment. Each of these generic designs will be evaluated to identify:

a. Design deficiencies,
b. Material compatibility problems,
c. Potential fabrication problems,
d. Potential problems caused by ambient and accident environments,
e. Potential problems caused by interfaces,
f. Installation problems,
g. Maintainability problems, and
h. Wear and aging problems.

For a selected group of most vulnerable generic designs, the following additional assessment will be conducted:

a. An audit will be conducted of the manufacturers' quality control and the quality control during installation and use; and

b. Tests will be conducted simulating accident conditions simulating the effects of wear aging, inherent design deficiencies and interface conditions.

The enhanced operator capability program has been significantly expanded. The elements of this program are listed below.

a. Plant Status Monitoring - The information needed by the operator to establish unambiguously the status of the plant is being systematically analyzed to assist in the development of plant status monitoring requirements. This includes instrumentation to follow the course of an accident and to identify the status of engineered safety features. These efforts supplement activities by the regulatory staff to develop and implement positions related to status monitoring (e.g., Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident;" Regulatory Guide 1.47, "Bypassed

and Inoperable Status Indication for Nuclear Power Plant Safety Systems;" definition of plant safety state vector; and capabilities of onsite and offsite technical support centers). These studies are being performed by Idaho National Engineering Laboratory.

b. Improved Instrumentation - Research is underway to test the feasibility of several new concepts for measuring safety-related physical parameters. Appropriate instrumentation is being designed, laboratory tested, and finally field tested in nuclear power plants to ensure workability. Emphasis is being placed on possibility for retrofit, reliability, and durability. Instrumentation needs identified include water level in the core, gas bubble in steam generators, low flow rates during natural circulation, flow through the relief valve, and radiation monitoring. These studies are being performed primarily at Oak Ridge National Laboratory and Idaho National Engineering Laboratory. We will also study whether void fraction can be measured directly in a reliable way.

c. Operator-Process Communication - Current practices in using lights, alarms, other annunciators and cathode ray tube displays in control rooms will be reviewed to assess how well they facilitate operator-machine interaction. Recommendations to improve operator-machine interaction in control rooms will be developed, and supporting laboratory or field experiments will be carried out.

d. Disturbance Analysis Systems - The validity of methodologies used in computerized diagnostic systems is being identified and evaluated. The findings will help the regulatory staff to determine the need for and nature of requirements for such systems. The goals are to recommend functional requirements for computerized systems capable of diagnosing the cause of a disturbance and to confirm the adequacy of technical approaches used by the industry in developing and demonstrating such systems. The effectiveness of prototype systems installed in operating power plants will be assessed. In addition, the LOFT project is upgrading its capabilities to use computer and advanced graphics to monitor the status of the reactor and diagnose disturbances. The system will be helpful in testing the feasibility and effectiveness of proposed improvements in the operator-machine interface. These studies are being performed primarily by Oak Ridge National Laboratory and Idaho National Engineering Laboratory.

Other ongoing areas of operational safety research are continuing and being strengthened. A new facility for qualification testing of electrical components has been completed at Sandia Laboratories. The fire test methodology program has intended to cover replication tests of actual cable installations. A program for demonstrating continuous on-line monitoring of a large PWR has been implemented and is expected to be in place at the Sequoyah Plant this spring. We also have recently undertaken to follow and evaluate the EPRI safety and relief valve testing program.

We believe that these new programs, some of which have only been developed since the ACRS review of our program several months ago, are adequately responsive to the ACRS. We recognize the importance of this area, and believe that this area could likely grow significantly over the coming years. We look forward to continuing interaction with the ACRS as the program develops.

3.3.6 Natural Circulation Capability of PWR Systems

ACRS Statement (page 3-3)

Heat removal by natural circulation is a critically important safety consideration during some shutdown transients. During loss of all AC power transients in some PWRs, it is the only means of transferring fission product decay heat from the reactor core to the heat removal system, short of coolant boiling in the core. Transition from natural circulation to boiling may be necessary during such transients. An experimental program is needed to establish a better understanding of this process. It might utilize a combination of facilities such as nuclear power stations operated at low power levels, LOFT, separate-effects facilities (U.S. and foreign), and some visualization-type, bench scale experiments. A list of variables to be investigated should be established and an experimental program should be planned for this purpose. This work should have high priority.

RES Comment

A series of tests has been scoped for the LOFT and the Semiscale facilities aimed at studying both the single phase and the two phase natural circulation in steady state and transients. Other domestic programs involve the FLECHT SEASET integral tests, TLTA, and tests in the Sequoyah Plant (single phase natural circulation only in Sequoyah). The PKL test facility in Germany is currently conducting small break tests (500 psia initial pressure), including core uncovery and two phase natural circulation. These results are being reported to the NRC under the 2D/3D cooperative program.

The LOFT small break series has begun testing which, when complete, will include:

- onset of single phase, then two-phase natural circulation
- loss of natural circulation
- system behavior with the steam generator tubes super-heated steam filled
- reflux boiling in the steam generator
- condensation in the steam generator with both liquid fall back and carryover
- return to natural circulation
- plant recovery from these conditions at both high and low PCS pressure, to a cold shutdown condition.

LOFT has also planned a series of operational transients and ATWS tests which will permit comparison with data from operating plants.

Finally these areas must be studied for the intermediate break size and tests of this type are now planned for LOFT.

Also basic studies concerning flow stability and heat transfer in U-tube steam generators during two phase natural circulation are being carried out at MIT.

3.4 Recommendations

ACRS Recommendation (page 3-3)

The research in this area (e.g., Systems Engineering) should be funded at the level requested. At least some of the studies included in the program on Operational Safety will contribute (or can be adapted to contribute) to the objectives called out in Chapter 2; and, as detailed plans are developed for the work to be undertaken, these should be directed as far as possible to contribute further to those objectives.

RES Comment

We agree.

Chapter 4. LOFT

4.4 ACRS Recommendations (page 4-1)
The support level of $43.0 million for LOFT is $6.3 million less than the ACRS commented on favorably in NUREG-0603. The ACRS believes that the LOFT program could use the higher figure of $49.3 million effectively in FY 1981; and also, in view of the large unavoidable expenses required for the upkeep and operation of this facility, that the higher level would be more cost-effective. However, the ACRS can accept the proposed support level of $43.0 million for LOFT on the basis that the $6.3 million reduction is restored to the total reactor safety research program and is used to support greatly accelerated programs in research to improve reactor safety and to initiate, or substantially augment, the new directions in research recommended in NUREG-0603 and discussed in Chapter 2 of this report.

RES Comment
RES agrees that the LOFT program could use the higher figure of $49.3 million effectively in FY 1981. We estimate that the additional $6.3 million would increase the testing rate by 50 percent, from six tests to nine tests in FY 1981. However, we can tighten our belts to live with the President's budget of $43 million and still conduct a good test program in LOFT that supports NRC's regulatory needs.

Chapter 5. CODE DEVELOPMENT

ACRS COMMENTS AND RECOMMENDATIONS:

5.3 Comments

ACRS Comment (page 5-1)

The principal computer code of choice, TRAC, suffers so far from incomplete knowledge of some of the necessary physical parameters. It should be pointed out that a fairly complete description of the possible physical situations in a reactor transient is required for the microscopic description used in TRAC. The microscopic description leads to long running times and thereby limits a rapid survey of the many possible transients. While an effort is under way to develop a fast running version of TRAC, the ACRS believes the RELAP5 computer code, which is already somewhat faster than TRAC, also should be developed to provide a second fast running code.

5.4 Recommendations

ACRS Recommendation (page 5-1)

The ACRS recommends the continued development of RELAP-5 as another general code of potential value. The ACRS recommends also that a strong program be initiated for the development of methodology and techniques that would facilitate the implementation of more sophisticated reactor simulators, not necessarily limited to real-time analysis. This would enable a more detailed understanding of the course of events in complex transients that include multiple failure and operator intervention. The ACRS believes that the proposed budget is adequate to include these developments without additional funds.

RES Comment

Development of both TRAC-PF1 (the first fast running version of TRAC) and RELAP-5 is currently proceeding under NRC/RES sponsorship. TRAC-PF1 is being developed under the guidance of the RES Analysis Development Branch staff and is subjected to the review of the Advanced Code Review Group. In addition, TRAC-PF1 will be subjected to a formalized code assessment process involving a number of national laboratories. The RELAP-5 code is currently aimed at providing analytic support to the INEL test facilities. When RELAP-5 capabilities have sufficiently matured to warrant its release to the public, the code will be subjected to the formalized review and assessment process used for all NRC developed codes.

NRC is considering the development of an engineering simulator that would allow investigation of accidents involving multiple equipment failures, operator interactions and even core damage, in real time or better. Two approaches are currently under evaluation to achieve this goal. One approach starts with the current vintage training simulator, seeking systematic improvements. The other approach involves the bottom-up design of the analytical procedure for simulating the reactor coolant system amenable to the existing special purpose computer hardware, such as an all digital simulator or an analog (hybrid) computer tied to an assembly of digital micro-processors. The balance of plant would be simulated utilizing the technology employed in current training simulators, except for introducing flexibility to address different LWR designs. Any of these would be a costly and lengthy development effort. We expect to lay out the options for a Commission decision in the next few months.

Chapter 6. FUEL BEHAVIOR

6.3.1 Clad and Fuel; Fuel Codes

ACRS Comment (page 6-1)

This work is of substantial aid in providing an NRC capability in fuel behavior analysis, and should continue at current levels. However, a greater breadth of input into the physical modeling would be desirable. Work on modeling of severe overheating, which occurred at TMI-2, is encouraged.

RES Comment:

We appreciate the Committee's support in this area and will, accordingly, continue our efforts to provide the NRC with up-to-date, fully assessed, fuel codes. As to increasing the "breadth of input into the physical modeling," we agree and will look for ways to expand beyond what is currently being implemented and planned. Currently, three national laboratories have direct input into the model development area (INEL, ANL, and Battelle Northwest) whereas data input for assessment comes from sources all over the world (Japan, FRG, Norway, England, etc.). Moreover, program reviews are held periodically in which the model development effort is presented and discussed. Attendees of these meetings include personnel from the national laboratories, ACRS staff, NRR staff, RES staff, reactor vendors, the utilities, EPRI, and private contractors. We hope and expect that this response to the ACRS comment is adequate.

We fully agree with the Committee's comment on modeling of severely overheated fuel. Work in this area resulted in the in-house developed TMIBOIL code which was used extensively by the Rogovin TMI Special Inquiry Group in assessing TMI-2 core damage. Moreover, modifications are planned for FRAP-T in FY 81 to expand its capability to handle slow transients which require much larger time steps and a moving water/steam interface. Finally, we will investigate the conversion of existing foreign and domestic severe damage codes to handle those transients which result in a loss of fuel element geometry.

6.3.2 In-Pile Testing at Power Burst Facility (PBF)

ACRS Comment (page 6-1)

PBF represents about 60 percent of the total fuel behavior research budget. The information on fuel behavior during reactivity insertion accidents (RIA) is still believed by the Office of Nuclear Reactor Regulation (NRR) to be inadequate. It is not clear that this experimental program has provided information of quantity and significance in proportion to its level of support. If these accidents are of sufficiently low risk (low probability and/or low energy insertions), such research is not necessary. The NRC Staff has not provided the ACRS with a convincing argument in favor of the need for the experiments on fuel behavior during RIA, or for most of the other experiments planned for PBF in FY 1980 and FY 1981.

The ACRS believe that PBF probably can be used for experiments related to flow starvation and fuel melting accidents and urges an early and complete evaluation of the currently proposed PBF program. In the meantime, the ACRS believes that flexibility in reprogramming some PBF funds to other high priority work on steam explosions and core melt should be provided.

RES Comment

RES has attempted to be responsive to the ACRS recommendations for fuel behavior tests in past years. The Committee commented in their 1977 Report to Congress (NUREG-0392) that the current programs on fuel behavior were considered responsive to the safety concerns noted by NRR and the ACRS. In December 1978 (NUREG-0496), the Committee generally agreed with the PBF test program but requested the staff to reassess the priority assigned to RIA tests. In response to this request, the staff did, in fact, reassess the program and greatly reduced the number of RIA tests planned and changed the tests to emphasize higher levels of burnup. In July 1979 (NUREG-0603), the Committee requested that NRR reevaluate the regulatory requirements for RIA tests and suggested that the program be reoriented to emphasize the study of processes leading to medium and severe core damage. In response to this request, the program was reevaluated by RES, NRR, SD, and the PBF Review Group. The conclusion of this review was that, because NRC's regulations require that RIA's be evaluated as part of the licensing review of each plant, the planned RIA tests are needed. NRR is preparing a User's Need letter for these tests, and we will discuss the entire PBF program and priorities with the ACRS subcommittees in the coming months.

RES agrees that the PBF program should be reoriented to emphasize the study of processes leading to medium and severe core damage, and we took action last summer to develop a program plan for severe fuel damage tests. Because of the lead time involved in planning the tests, designing the test trains, and fabricating the hardware and test fuel, these experiments cannot be implemented before 1982. It is very important that the fuel damage tests be well planned and thoroughly reviewed by NRC and the technical community. We expect to discuss these tests and seek the Committee's guidance in the coming months.

With regard to the recommendation to provide flexibility to reprogram some PBF funds to other high priority work on steam explosions and core melt, we will investigate the options available. Reprogramming has an impact on the future PBF program that must be carefully evaluated, because planning and fabrication of hardware for the new flow starvation and fuel damage tests will require funds over and above those budgeted for the current test program. The only way to make significant funds available for reprogramming would be to stop testing in PBF until the fuel damage tests are ready, and at this time we do not believe the safety tests planned in PBF over the next two years should be eliminated or deferred.

6.3.3 Other In-Pile Testing

ACRS Comment (page 6-2)

These are confirmatory programs related to core behavior following a large LOCA. The priority is probably low. The joint U.S.-Canadian research program at the NRU reactor in Canada should be terminated in FY 1983 as planned. Before committing to the multi-national research program at the ESSOR reactor complex in Ispra, Italy, the NRC Staff should be convinced that there are not higher priority NRC research needs.

RES Comment

The NRR audit curves that are now used to predict fuel rod ballooning and rupture and resultant threat to post-accident core coolability after a

loss-of-coolant accident (LOCA) are primarily based on out-of-pile separate effects tests which employ electrical heaters that have never yet been shown to simulate accurately the in-reactor ballooning of full length low enrichment commercial reactor fuel rods. Only a few data points based on TREAT and FR-2 tests using short test fuel rods are included in the data base.

The NRU tests are the only in-pile fuel bundle tests which will use prototypic, full-length, commercial-enrichment fuel rods to establish the validity of all the out-of-pile test data which currently purport to represent the nature and extent of clad ballooning during the heat-up period of a LOCA. Review group and technical community consensus is that the NRU tests are needed to verify the out-of-pile simulation of in-pile behavior by cluster of electrically heated fuel rod simulators.

The current NRU LOCA program is on schedule and the plans have been to terminate this program in FY 1983. However, we cannot at this time commit to complete our relationship with the Canadian NRU reactor in 1983 until we have assessed the need for possible further tests on long large bundles or even long single rods of highly irradiated commercial enrichment fuel under accident conditions of continuing interest to the NRC.

Regarding priorities within the Fuel Behavior decision unit, we find that support for ESSOR has a relatively high priority. The fuel bundle test program (called Super Sara) proposed for the ESSOR reactor was reviewed by RES and the European Community groups following the Three Mile Island accident and was reoriented to delete some LOCA tests and to add tests which will produce more severe fuel damage by extending core heatup to the temperature of cladding melting. This reorientation of the ESSOR program is in response to the RES and ACRS desire to study the course of serious accidents. The ESSOR facility will provide, by virtue of its nuclear heat source, length (2 meters), and bundle size (32-36 rods), the more realistic set of thermal hydraulic and cladding oxidation conditions that are needed to understand better the rate and distribution of core damage and the possibilities of stopping the accident before meltdown becomes inevitable. The PBF, while lacking this capability, will complement this approach by focusing on fuel rod melting, geometry changes, and resulting core coolability as mentioned in our earlier paragraphs. The Super Sara program has now been approved by the European Communities as part of their 1980-83 program, and the total cost of the project, including past costs for loop construction, is estimated at $139 million in current dollars. NRC's commitment is to provide three test trains and manpower support in analysis, instrumentation needs and post-irradiation examination. These costs are estimated to be about $1.5-2.0 million per year (in current dollars) for 6 years. Thus, because of the high leverage that we have in obtaining important data on severely damaged fuel for a relatively modest cost to NRC, we believe the ESSOR program has a relatively high priority within our Fuel Behavior Program.

6.3.4 Fuel Melt

ACRS Comment (page 6-2)

This work currently includes steam explosions and interactions of molten core material with concrete. In NUREG-0496, the ACRS recommended that work on phenomena important to the course of postulated core melt accidents should continue to receive high priority. In sections 1.2.4 and 1.2.6 of NUREG-0603 (See Appendix B), the ACRS recommended an augmented research program on steam explosions and a conceptual study to examine the practicality of molten core retention within containment. The ACRS recommends that the existing program be reoriented and strengthened accordingly and, furthermore, that it be closely coordinated with work being done on the cause of severe accidents.

RES Comments

A. Steam Explosion Research

The current NRC steam explosion research program (conducted at Sandia Laboratories) is substantially funded ($500K FY 1980). We had previously budgeted $765K for the 24-month period FY 81-82. These funds would be utilized for: (1) conducting tests in the fully-instrumented intermediate-scale test facility (approximately 50% of the budget); (2) small scale phenomenological model development experiments (approximately 20%); and (3) test analysis, model development, and containment failure analysis (approximately 30%).

Recent budget plans for FY 1981 and FY 1982 direct (through reprogramming) an additional $485K per year to the steam explosion program to conduct large (>100kg melt mass) tests (to verify the effect of melt mass on steam explosion energy conversion ratio and to aid in extrapolating the results obtained from the intermediate scale tests to prototypic LWR melt sizes) and to secure the services of additional personnel to participate in testing, test analysis, and model development. This represents an increase of approximately 125% over previously planned funding.

We believe that the existing steam explosion program, augmented by the indicated additional funding, is responsive to the recommendations of the ACRS.

B. Molten Core Retention

RES has formulated and NRR has endorsed a proposed research program in this area in response to an NRR User's Need Request. The proposed research program is primarily aimed at establishing a data base to support the technical evaluation of the adequacy of the core retention device proposed for use in the current Floating Nuclear Plant design. However, the proposed research program will also provide information necessary to evaluate the generic applicability of similar passive core retention devices in land based plants. As part of this program, the conceptual study recommended by the ACRS will be undertaken and given high priority. Initial efforts will draw from, and expand upon, the core retention evaluation performed during the Zion/Indian Point core melt accident studies. This activity will be initiated in the near future.

Detailed planning of this work is underway at Sandia and the preliminary program plan will be available in FY 1980. Initial funding of $500K for the overall program is budgeted in the FY 1980 TMI Supplement. Funding in FY 1981 will require reprogramming of RES funds. Reprogramming decisions will be made following review of the integrated fuel melt program plan, noted previously in Section 1.1.1, and reassessment of research priorities.

Chapter 7. PRIMARY SYSTEM INTEGRITY

7.2 General

ACRS Recommendation (page 7-1)

A program should be started to provide a firm basis for establishing and judging adequate systems for coolant chemistry control. Several of the recent incidents of pipe and nozzle cracking occurred at locations not subject to routine in-service inspection. This suggests a need for a systematic review of current thinking about system behavior and degradation and an evaluation of the possible need for redirecting research in order to anticipate or prevent similar cracking incidents in the future.

RES Response

RES has undertaken a study in FY 1980 to define a program of research in environmental and coolant chemistry effects on cracking and stress corrosion in BWR and PWR piping systems. This newly defined program will be implemented in FY 1981 with funding adequate for a good start on this coolant chemistry control effort. However, we do not plan to study plant design or evaluate specific systems used for water chemistry control but rather we will study the parameters for water purity to alleviate corrosion and cracking problems. We will pursue this issue further with the ACRS.

7.3.2 Operating Effects

ACRS Comment (page 7-1)

This program consists of two areas: Irradiation Effects and Dosimetry, a valuable well organized program; and Steam Generators, a program about which the ACRS has reservations. The main effort of this latter program involves a detailed, destructive examination of one of the steam generators removed from the Surry Power Station. A careful study must be made to determine if a positive contribution can be made by the steam generator study before performing work in addition to that needed to determine the correlation between nondestructive examination indications and tube integrity.

RES Response

The entire research program to be performed on the replaced Surry steam generator has been planned in response to a request for such work from NRR (E. G. Case to S. Levine, "Request for Confirmatory Research Study of a Replaced Steam Generator for Various Causes and Forms of Degradation). In turn, the complete plan for research (which includes the nondestructive examination correlation work as a first step, concurrently with some destructive testing and some corrosion-related work), has been developed by RES and agreed to by the NRR staff on August 10, 1979, as being fully in accordance with their needs (RR-NRR-78-15, June 27, 1978). We believe that this program will make a positive contribution to understanding margins in degraded S-G tubing and in assessing eddy current inspection techniques. There are 25 plants operating and 27 plants under construction that have the same steam generator design as the Surry steam generator. Thus, the results of this study will be applicable to the steam generators in some 52 plants.

With regard to the ACRS comment that a careful study be made before performing work beyond nondestructive examination studies, we believe that the program

plan we have developed with NRR addresses this recommendation. As the program progresses, we expect to review it with NRR and the ACRS, and if it is agreed that modifications are needed, we will make them.

7.3.3 Non-Destructive Examination

ACRS Recommendation (page 7-2)

This is an expanding program on an important topic. The coherence, as well as coordination with regulatory needs, leaves something to be desired. The program should be funded, but the ACRS urges that the NRR and RES managements improve the coordination of the programs on Primary System Integrity with respect to regulatory needs. Also, emphasis should be placed on how program developments will influence design and practice in plants.

RES Response

We agree that there must be close coordination between the RES and NRR staffs to ensure that our research programs are responsive to the regulatory needs of the agency. The RES staff continues to hold Review Group meetings in the nondestructive examination subject area to which NRR, SD, and I&E personnel are invited. All NDE programs have been endorsed within the last year. The staff has just completed two DOE projects this fiscal year responding to specific I&E requests.

All of the nondestructive examination programs and all of the branch programs in other areas are carefully directed to the final application of ultimate use in the field in operating plants, or to direct input for changes in the ASME Code or Regulatory Guides. Specifically, the new nondestructive examination developments are directed to be compatible with current practices so that no new design or operating requirements will be needed for the immediate application of the new techniques.

7.3.4 Corrosion and Cracking

ACRS Recommendation (page 7-2)

The new program on cracking in BWR piping should be broadened to consider the corrosion-accelerated problems found in PWR pressure boundaries. The criteria for water chemistry limits, plant design, and operating procedures required to approach more trouble-free operating conditions should be addressed.

RES Response

The RES program in nondestructive examination emphasizes not only in-service inspection research, but also development of global on-line continuous monitoring for anticipating cracking and for following the development of cracking (See response to 7.2). Use of these on-line systems (when they are fully developed) will mean that cracking can be predicted better and its growth monitored more accurately so that corrective action can be taken prior to leakage or growth to critical condition, for any location in the plant. We have recognized the necessity for a fresh review of our research priorities; hence, we are planning a risk assessment review of our work and plans with the Probabilistic Analysis Staff in order to develop an improved basis for priorities within the research.

Chapter 8. SEISMIC, ENGINEERING AND SITE SAFETY

8.3.1 Seismology, Geology, and SSMRP

ACRS Comment (page 8-1)

The research program on seismology and geology and the SSMRP are among the high priority NRC programs. It is important that the SSMRP program be structured to provide input as early as is feasible into the broad safety policy considerations concerning the seismic design bases of nuclear power plants. This should include a timely preliminary evaluation of the seismic contribution to the probability of serious accidents and the principal contributors to uncertainty in such probability estimates.

RES Response

We agree. The Seismic Safety Margins Research Program (SSMRP) is a long-term research program that will be completed in 4 or 5 years. However, the program is structured to provide intermediate results to the licensing staff that will influence the broad safety policy considerations concerning the seismic design bases of nuclear power plants. These results will be transmitted to the staff in technical reports published throughout the life of the program. There will be approximately 22 technical reports published in FY 1980. Topics to be reported in FY 1980 include:

1. Review of structural building response analysis with special emphasis on damping and nonlinearity - February 1980

2. Assessment and expansion of the strong ground motion data base - April 1980.

3. Subsystem Response Review Reports assessing the state-of-the-art of subsystem response determination, accuracy and uncertainties - April 1980.

4. Soil Structure Interaction (SSI) Review Reports assessing the state-of-the-art of SSI analysis methodology, accuracy, uncertainties, and itemizing benchmark problems - June 1980.

5. Response of nuclear power plant structures to three input components - June 1980.

6. Effect of structural damping on nuclear power plant structures - June 1980

7. Best-Estimate-Evaluation Method (BE-EM) permit, an evaluation and comparison of best estimate responses with the responses obtained using the Standard Review Plan (SRP) requirements - August 1980.

8. Report on methods used to gather fragility related information and data summary - September 1980.

9. Report on expert opinion received on component fragility - September 1980.

10. Simulation of time histories parameterized by magnitude, epicentral distance, and site conditions - October 1980.

11. Sensitivity study investigating the effects of uncertainties on subsystem response - October 1980.

12. Operating Seismic Safety Analysis Code (SEISIM) - January 1981.

The SSMRP will initially use a typical PWR (Zion I nuclear power plant) as a model to conduct sensitivity studies and quantify the uncertainties associated with the various parameters of the seismic methodology chain. Later phases of the program will consider other PWR and BWR designs. It is expected that in FY 1982, the SSMRP will provide a preliminary evaluation of the probability of failure of structures, systems and components and the probability of radioactive releases over a range of earthquake levels. The principal contributors to uncertainty in such probability estimates will also be identified. The SSMRP is confined to probabilities of failure and radioactive release under earthquake and possible simultaneous accident conditions. However, the results and models developed in the SSMRP could be used to compare the risks associated with the seismic hazard to other possible accident sequences.

8.3.2 Hydrology

ACRS Comment (page 8-1)

The program on hydrology should be kept under continuing evaluation to see if the current low level effort is adequate to support possible informational needs arising from the consideration in future siting policy of liquid pathways effects from serious accidents.

RES Response

We agree. The hydrology program undergoes evaluation to ensure that the present funding level is adequate to support the stated consideration. This comment apparently relates to the new siting policy that would require consideration of groundwater paths in the liquid pathway using Class-9 accident source terms.

The state-of-the-art in radiohydrology is adequate to model rather accurately the groundwater flow and sorption of radionuclides by soils, both of which are highly site-dependent and variable. While it is currently possible to predict timing, quantity, and concentrations of releases to surface water bodies from given sources (sump water), some research is needed to formulate criteria for use in siting. Additional research requirements probably will come from the Siting Policy Task Group.

8.3.3 Structural and Mechanical Engineering

ACRS Comment (page 8-1)

The programs in structural and mechanical engineering are relatively new. Effort should be devoted to the formulation by FY 1981 of a broad research program responsive to the NRC needs arising from operating reactors and from reactors to be constructed. The research program should include efforts devoted to p ovide the NRC with an improved capability for design audit and to evaluate the significance of off-design conditions during potential accidents or other severe loading conditions such as a large earthquake, as well as an improved basis of experimental verification of seismic design.

RES Response

Research program plans have been developed by the Structural and Mechanical Engineering Research Branches and are currently undergoing a thorough review by the NRC user offices. In addition, these coordinated research plans will be reviewed with appropriate ACRS Subcommittees. The following table shows the current research programs that address the ACRS area of concern.

STRUCTURAL AND MECHANICAL ENGINEERING RESEARCH

AREAS OF RESEARCH NEED	RESEARCH PROGRAM
Improved Capability for Design Audit	Benchmarking of Computer Codes Engineering Characterization of Seismic Motion Load Combination for Design of Structures
Evaluation of Significance of Off-Design Conditions During Potential Accident or Other Severe Loads	Effects of Hydrogen Explosion Flood Effects SSMRP Safety Margins of Containment Safety Margins of Other Category I Structures
Improved Basis of Experimental Verification of Seismic Design	Safety Margins of Containment Safety Margins of Other Category I Structures Measurement to Predict Structural Performance Adequacy of Codes and Standards Dynamic Testing of Nuclear Power Plant Structures and Components

Chapter 9. ADVANCED REACTORS

All ACRS comments and RES replies in this chapter are predicated on the possibility of program continuation into FY 81 and beyond. It is recognized that the President's budget calls for a phase out of the HTGR program at the end of FY 80 and of the LMFBR program at the end of FY 81, in which case the technical discussion here is of no practical applicability.

9.3.1 Analysis

ACRS Comment (page 9-2)

The ACRS does not believe that the proposed FY 1981 allocation provide enough emphasis on other than core disruptive accidents. Attention is directed again to the recommendation quoted in part above. It is believed that both accident delineation and accident prevention should receive greater attention than now seems indicated. In addition, the accident delineation work that is proposed seems to put too much emphasis on the Clinch River Breeder Reactor. However, the SIMMER computer code and the other analytical activities are viewed as important and valuable, and need to be continued at a level adequate to sustain them.

RES Response

We agree that more emphasis should be placed on investigating accidents other than core disruptive accidents. Our ability to address these accidents depends to a large degree on the funding level of the ARSR program.

About 40% of the analysis budget is directed at the continuing resolution of key licensing issues highlighted in the FFTF and CRBR reviews: treatment of core-melt and other whole core accidents. In view of the significant progress made in the past years, as well as the importance of current work, it would be unwise to decrease this activity significantly. The shift in the SSC activity from development to application, the focusing of work at the University of Arizona and Sandia on systematic accident delineation, and new management and staffing of the accident delineation work at Sandia should yield positive results. We agree with the ACRS that it is regrettable that the only detailed design available for review is the CRBR; we are working with DOE to keep abreast of CDS efforts, but it will be about 2 years before specific examinations can begin. We will increase our effort to obtain design and safety information on foreign breeder reactors in the coming year.

9.3.2 Aerosol Release and Transport

ACRS Comment (page 9-2)

This is a combination of analyses and experiments aimed at an important problem area. This work seems well planned and is producing results.

RES Response

We agree. The results from this program are now being used for planning the extension of aerosol technology to core melt accident mitigation, e.g., filtered, vented containment concepts. The high relevance of this LMFBR work to issues in LWR safety should also be noted.

9.3.3 Materials Interaction

ACRS Recommendation (page 9-2)

This item includes funds for loop design and fabrication and for a series of fuel tests. It is clear that fuel research needs to be done. While the NRC needs to do work on problems crucial to licensing concerns, a more determined effort should be made to have the fuel developers assume a large part of the investigative burden. In addition, more effort is needed to obtain a more precise formulation of the questions to be asked and how the answers are to be obtained with these facilities.

RES Response

We agree with the thrust of the recommendation. A major portion of the research needs is being borne by the developers (DOE) in programs in the TREAT, ETR, FFTF reactors. The total DOE supported programs are a factor of 10 larger than the NRC program. We expect that when FFTF begins full power operation and fuel testing begins, the ACRS recommendations will be substantially met. Another program of research in this area to which the NRC is a party is the French CABRI program. This program is on the order of $15 million/yr.

As recommended by the ACRS, we are pursuing a sensitivity analysis at LASL with assistance from ANL, BNL, and Sandia to define more precisely those fuel safety tests crucial to fast reactor safety.

9.3.4 System Integrity

ACRS Recommendation (page 9-3)

The proposed program involves testing of the CONTAIN computer code and carrying out a set of experiments associated with molten core retention, core debris coolability, and container cell liner response to accident loads. Some of the work on molten core retention is also useful in connection with licensing concerns of the Floating Nuclear Plant and in consideration of problems associated with severely damaged cores in water reactors generally. The work associated with this item seems appropriate to future needs in the development and licensing of fast breeder reactors. However, the ACRS believes, as recommended in NUREG-0496, that specific attention should be given to the study of alternate containment systems and to conceptual studies of systems for retaining a molten core in containment.

RES Response:

In FY 1981 RES will initiate conceptual studies of alternate containment systems and systems for molten core retention, if substantial restoration of the Fast Breeder Reactor FY 1981 budget occurs. The purpose would be to factor such systems into accident delineation work. Initial efforts would draw from and expand upon the designs proposed for the FFTF filtered, vented containment modifications, and the CRBRP core retention device, as well as the work on the Floating Nuclear Plant core ladle and the Zion/Indian core melt accident studies.

Chapter 10. REACTOR ENVIRONMENTAL EFFECTS

We appreciate the Committee's endorsement of our research programs related to mathematical models for the transport of radionuclides via liquid and atmospheric pathways and our programs related to occupational radiation assessment and protection. We have noted the Committee's observations on other aspects of our program and provide the following responses.

10.2 General

ACRS Comment (page 10-1)

The Committee again noted the need to conduct research in the following areas:

(a) Research to determine the basic factors that govern radionuclide buildup in reactor coolant systems, including the influence of operating practices on such buildup.

(b) Research to develop methods for evaluating the effectiveness of measures for removing radionuclides from the primary coolant circuits of operating reactors.

RES Response

In the past, there has been a great deal of research carried out to identify the sources, characteristics, and locations of the radionuclides deposited in reactor cooling systems, and on the performance (effectiveness) of coolant clean-up systems using techniques such as evaporation, ion exchange, and filtration. RES has conducted research which documents the operational efficiencies of effluent control systems and sponsored a workshop on the same subject. An on-going program at INEL is documenting decontamination factors for reactor coolant clean-up systems under a variety of operating conditions.

With respect to occupational exposures, RES has recently developed a model (NUREG/CR-0446) which is being used by both the licensing staff and industry to determine the cost effectiveness of various methods for radiation protection of the worker. Although there is a substantial body of data which permits evaluations of licensees' plans to assure occupational exposures are kept at ALARA levels, RES is continuing to conduct and monitor research which can influence occupational exposures and related ALARA determinations.

The on-going RES program on fuel failures due to pellet-cladding interaction (PCI) and radionuclide releases derived from PBF experiments is expected to provide useful information on fission products as contributors to the buildup of radiation levels in working areas. In addition, RES is developing and testing methods for nondestructive, passive failure detection systems such as acoustic emission and internal friction measurement which may eliminate the need for direct in-service inspection of the vessel and piping by plant personnel thereby reducing occupational exposures. The proposed studies of the Surry steam generator, referenced in other sections of this report, will develop significant information on both corrosion and remote inspection techniques.

In related efforts, research has also been planned for FY 1981 to assess the effectiveness of various techniques for crud removal. EPRI is conducting studies of techniques such as chemical and power/flow rate manipulation to dislodge deposited radioactivity. RES also plans to include studies to measure the effectiveness of strong and weak decontamination solutions on crud release, and to establish the compatibility of such solutions with coolant clean-up and waste solidification systems. RES is also conducting a thorough review of the subject of crud formation and buildup to identify any gaps in the data base which would preclude achieving an ALARA objective for occupational exposure.

10.3 Comments

10.3.1 Priority Items Within the Program as Planned

ACRS Comment (page 10-2)

Development of Mathematical Models for the Atmospheric Transport of Airborne Radionuclides. Although models for the transport of airborne radionuclides over short distances are reasonably adequate, there continues to be a need for an improved capability to assess the behavior of airborne releases at moderate (5 to 15 kilometers) and greater distances (16 to 80 kilometers) from nuclear power plants. This is especially important relative to emergency planning, where models are needed to provide projections on a real-time basis.

RES Response

RES has developed a program utilizing field experiments to be conducted in shoreline environments, and in cooperation with FEMA, at Indian Point to obtain concentration data up to 80 km from the plant. A concurrent program has been inaugurated to compare atmospheric dispersion models using data already available with the objective of determining the most appropriate model for use at any particular site. Two other projects compare model to model results arising from given test conditions and provide for continued "upgrading" of the CRAC code used in the Reactor Safety Study. Also research to determine the feasibility and possible design of an off-site environmental monitoring network and telemetry system is planned to begin under the FY 1980 supplemental budget.

10.2(c) Research on Emergency Planning

and

10.3.2 Priority Programs Not Within the Program As Planned

ACRS Comments (pages 10-1 and 10-2)

The Committee noted further that additional research was needed on the aspects of emergency planning.

Accident Source Terms. Better definition of accident source terms is needed. Emphasis should be place on requirements for instrument systems to provide the definitive type of data necessary for realtime projections of the nature and consequences of a release.

RES Response

As a part of our operational safety research program we have initiated qualification testing of post accident monitoring equipment. This program is being carried out in support of Regulatory Guide 1.97. It is planned to

conduct LOCA qualification tests on generic instrumentation required for normal and post accident monitoring to determine both the design adequacy of the generic designs and the suitability of current test methodology. A new LOCA test facility recently has been built at Sandia Laboratories and testing will begin this fiscal year. Testing of radiation instrumentation required for post accident monitoring is planned later in the program, currently projected for FY 1982.

The previously mentioned fission product release work from fuel elements will provide improved data on the quantities of fission products which might be released as a result of Class 9 accidents. The topic of accident source terms has also been identified in NRC's Three Mile Island Task Action Plan (TAP) as an area where increased research is necessary. While the precise details of the research have not yet been defined, program plans are being formulated in the following areas.

(a) In-pile studies of fuel behavior in the Power Burst Facility;
(b) Hydrogen studies to provide a better understanding of radiolysis, metal-water reactions, embrittlements;
(c) Post accident coolant chemistry and related pellet clad-coolant interactions;
(d) Interaction of core melt with structures, soils;
(e) Aerosol production and transport.

Source term calculations have been made as part of the operational safety research effort to define the magnitude, rate and particle type released as a function of time using the leakage assumptions in Regulatory Guide 1.89 and WASH-1400. This program is an on-going effort and new calculations of source term will be made when revised source term release fractions are determined in the above mentioned fuel release studies. Also BCL is performing an uncertainty analysis on the accident source term for core melt accident sequences.

ACRS Comment (page 10-2)
Interdictive Measures. Studies of the full range of interdictive measures with emphasis on their suitability for given sites and means for their improvement are needed. Accompanying this research should be a reevaluation of Protective Action Guides and the initiation of research to develop a better scientific basis for their establishment.

RES Response
A study by Sandia Labs and PAS (NUREG/CR-1433) has been completed on a cost/benefit analysis of using Potassium Iodide as an emergency protective measure for the general public. In addition, work has been initiated to determine the potential benefit of taking ad-hoc respiratory protection measures during an accident such as breathing through a handkerchief or towel. PAS is also sponsoring an effort to develop a set of realistic accident scenarios to be used in testing state emergency plans.

Additional research is planned, as a part of the FY 1980 supplemental budget, to evaluate potential emergency response and planning at Federal, State, and

local levels. These will include consideration of interdictive measures. Also technical assistance projects and staff efforts in other NRC offices are being applied to EPA/FRC review of Protective Action Guides.

Recovery and Reentry Phase Following Accident. Further investigations of improved measures that might be implemented in the recovery and reentry phase following an accident must be made. This program should include evaluations of designs and procedures to facilitate the decontamination and recovery of major nuclear power plant systems. It should also include research on procedures to aid decisions by medical and other authorities concerning the affected offsite population; methods for decontaminating and reclaiming offsite land, buildings and equipment; and the establishment of dose limits or guides for population groups desiring to return to areas that have been evacuated.

RES Response

Research to accumulate physical data on the nature and extent of core damage and fission product deposition and plateout at TMI is being planned and pursued jointly by NRC, DOE, EPRI and GPU. The information provided by these studies concerning fission product behavior and accident source terms is expected to be valuable for planning emergency response, recovery and reentry procedures and for planning additional research.

The NRC is reviewing the EPA-FRC Protective Action Guides (PAG) and Part 140 of Title 10 Code of Federal Regulations pertaining to extraordinary nuclear events. Decontamination methodology studies have been directed to nuclear systems and have not included off-site land, buildings or equipment. These subjects will be considered in future planning to determine what research is needed to improve NRC emergency response, decisionmaking and guidance.

The authority for granting reentry to an area that has been evacuated is vested in the State government. There are in EPA-FRC PAG No. 7, strontium 89-90 and cesium 137 limits in soil which are applicable to long term residence in contaminated soil areas. The limits are such that the resulting exposure to adult bone marrow or whole body dose not exceed 0.5 rad in the first year and, similarly, the dose to bone marrow or whole body to an infant shall not exceed 0.2 rad in the first year. Additionally, the EPA has published proposed EPA-FRC Guidance on Dose Limits for Persons Exposed to Transuranic Elements in the General Environment which is applicable to releases to the environment from accidents. The NRC emergency planning guide takes into account the EPA-FRC Protective Action Guides regarding limiting direct exposure to the plume, and shielding afforded by dwellings, and is coordinating with the DHEW-Food and Drug Administration all matters related to safe food supply. DOE-NRC research is studying soil-plant uptake of radionuclides and the effects of various soil treatment and agricultural practices on the rate of transfer of radionuclides from contaminated soils to food crops which will be applicable to regulatory guidance on use of contaminated agricultural lands.

10.3.3 Items within the Research Program that are Considered of Low Priority

ACRS Comment (page 10-3)

The ACRS does not believe that there is an urgent need for emphasis on research to improve the models for describing low level airborne or liquid radionuclide releases from nuclear power plants under routine conditions. This is especially true relative to refinements in the calculations that support 10 CFR 50, Appendix I.

RES Response

The research being conducted to provide data and models to support NRC's assessment of licensees' compliance with Appendix I requirements has been undertaken in response to continuing requests from the licensing staff. In conducting this research, we have noted that compliance with Appendix I can only be assessed through the use of models. Thus, the accuracy of the input data to predictive models and the models themselves are essential elements in the licensing review.

Several programs are being conducted to support Appendix I. An in-plant measurement program is being conducted to validate the quantities and characteristics of PWR effluents in support of a revision of Reg. Guide 1.1. RES is also developing a liquid pathway model which will include sorption on sediments in predicting radionuclide transport. This work is necessary to upgrade Reg. Guide 1.113 and NUREG-0140, which omit consideration of radionuclide transport via sediments in demonstrating compliance with Appendix I. On-going modelling and validation of atmospheric transport processes out to 50 miles (as required by Appendix I) and locally, with a variety of topographic conditions, are also being continued.

In evaluating these programs, it should be noted that the data being developed also support regulatory assessments other than those related exclusively to Appendix I. The in-plant measurement program is providing information on crud behavior and on the performance efficiencies of gaseous and liquid effluent treatment systems under a variety of operating conditions. Such data can be applied to required occupational exposure assessments and to establishing decontamination requirements. The atmospheric and liquid pathway modelling efforts also support assessments of accident impacts, emergency planning and interdictive measures. For these reasons we believe that not only are these research programs needed to support the licensing staff in public hearings, but they are also developing high quality data applicable to a variety of other regulatory objectives.

Chapter 11. FUEL CYCLE

11.3 Comments

ACRS Comment (page 11-1)

Decommissioning. The ACRS believes that the research on decommissioning of fuel cycle facilities is important and should be funded above the level planned by the NRC. More emphasis should be given to research on the problems of decommissioning or long-term care of shallow land burial sites.

RES Response

Until recently, RES was unable to initiate research in this area because user office endorsement had not been obtained. In early FY 1980, however, a user need for fuel cycle facility decommissioning research was received from NMSS, and we are proceeding with immediate implementation. A total of $300K is available in FY 1981 for research on fuel fabrication and chemical plant decommissioning. These studies are planned to extend for more than three years and will include on-site measurements. Although program costs are projected to be somewhat greater in FY 1982 and 83, the FY 1981 funds appear to be adequate for program startup. The decommissioning of low level waste burial sites is discussed in Chapter 12.

ACRS Comment (page 11-1)

Safety System Performance. The ACRS believes that more work is needed to assure adequate performance of safety systems when they are called upon. For example, more information is needed on conditions that adversely affect air filter system capability and on testing methods to confirm that satisfactory performance capability exists.

RES Response

Research was initiated in FY 1979 to evaluate realistically facility equipment responses during accidents which could lead to significant off-site releases of radionuclides. One of the areas of investigation covers possible failure modes of ventilation and filtration systems. DOE has research in progress to evaluate the effects of explosions in fuel cycle facilities.

ACRS Comment (page 11-1)

Effluent Control. The ACRS believes that the research effort on effluent control should be augmented. Increased emphasis is needed on the problem of radioactive gaseous wastes with respect to their removal, confinement, and long-term storage or disposal.

RES Reponse

HEPA filters aged in actual fuel cycle facilities are being tested as part of the program identified above under effluent control research. However, these tests will not in themselves provide definitive data on the effects of the complete range of possible degrading environments. An initial effort to conduct such tests and evaluate testing methods to confirm filter system performance may be initiated in FY 1981 as part of the effluent control project if funds are available. In FY 1981, our support will be only $300K instead of the $750K that could have been available if the orginally planned FY 1981 budget had not been cut because of overall fiscal constraints. One of

the projects deleted was intended to evaluate the effectiveness of collection and onsite storage of gaseous radionuclides. If funds should become available, we will initiate research in this area in FY 1981. Work is also being done by RES on the risks of the removal, confinement, storage, and disposal of gaseous wastes.

11.4 Recommendation

ACRS Recommendation (page 11-1)

The ACRS recommends that more emphasis be given to research areas cited in Section 11.3; this can be accommodated by decreased effort on transportation research.

RES Response

RES does not believe it would be prudent, at this time, to reduce its efforts in transportation research. Over the last several years there has been increasing concern on the part of the public and the Congress regarding the safety of transporting radioactive materials. These concerns prompted the Commission to initiate an evaluation of its regulation from a risk perspective. Although this evaluation (NUREG-0170) indicated that under existing NRC regulations the transportation of radioactive materials presents a very small risk to the public health and safety, the NMSS staff concluded, and the Commission concurred, in the need to develop additional information relative to protection of the public against high consequence accidents without regard to their predicted frequency of occurrence. Accordingly, RES was requested to develop, if possible, an optimum combination of package test requirements and acceptance standards which would reflect a cost effective balance between current packaging technology, severe accident environments and the risk to the public health and safety. The central research requirement is to conduct physical testing of shipping packages; however, the complex relationships among the various parameters makes it advantageous to conduct the program as an integrated research activity rather than as a mixture of research, technical assistance and staff actions. An important element in the acceptability of the revised regulations will be the ability of the NRC to demonstrate the technical basis for their development. We share with NMSS the belief that this program is essential to support future regulatory decisions in the area of transportation.

Chapter 12. WASTE MANAGEMENT

12.2 General

ACRS Comment (page 12-1)

In its 1978 report (NUREG-0496) to the Congress, the ACRS criticized the NRC for the poor formulation and management of research work on waste management problems and for the inadequate rate of progress. Similar criticism was expressed by the ACRS to the NRC in the July 1979 report (NUREG-0603). In NUREG-0603 the ACRS added that upgrading was needed in the NRC research staff capability.

The ACRS believes that the NRC has taken positive steps to improve this situation. Staff capability has improved; however, more is needed. Commendable progress is also being made toward improved assessment and selection of research; further attention, however, is needed on this matter, especially the ordering of priorities.

RES Response

We recognize the need to improve our capability to assign priorities to our research. RES and NMSS staff are currently working together to identify and assign priorities to technical issues associated with the implementation of draft rule 10 CFR 60, and to identify related research requirements. The assignment of priorities to research is on the basis of the importance of each of the technical issues to the regulatory process and the completion date required for the research results to be useful. As we currently formulate our five-year plan for research, priorities are being reviewed from a broader perspective, taking into account NRC's long range research needs.

12.3 Comments

12.3.1 High Level Waste (HLW)

ACRS Comment (page 12-1)

The ACRS believes that work in this area, including that related to the ultimate disposal of spent fuel, has high priority and that adequate funding is necessary for its timely completion. It is therefore urgent that the NRC develop detailed criteria and procedures needed to evaluate: (1) the suitability of a HLW package for ultimate disposal, and (2) the licensability of a geologic site as a repository for HLW packages. Some of the unique technical problems with respect to the latter task may require further augmentation of staff capability, for example, added expertise in geological sciences and engineering.

RES Response

Research is being conducted to support the formulation of detailed criteria and procedures needed to evaluate the licensability of a geologic site as a repository for HLW packages. Current research projects at the University of Arizona involve formulation of basic understanding governing groundwater, rock mass sealing, and evaluation of geophysical methods for site characterization.

A major research effort to design and establish field test facilities for large-scale in-situ experiments in rock mechanics, hydrology, and geochemistry, is presently being considered.

Research requirements have also been identified as a result of the joint RES/NMSS/OSD study of information needs for licensing under rule 10 CFR 60. These research requirements have been taken into account in the RES five-year plan. Additional research projects based on this review will be initiated in the near future, as RES resources permit.

With respect to staff augmentation, we intend to hire a geological engineer and, if possible, a mining engineer and a systems analyst.

The products from the risk assessment methodology development projects for High Level Waste and Spent Fuel include licensing tools and procedures which are being used by the NRC Interoffice Waste Management Modeling Group to assist in the development of detailed criteria and procedures needed to evaluate the licensability of a geologic site as a repository for HLW packages. The sensitivity analysis methods developed in these projects are giving insight into the importance of the leachability of the waste form. Therefore, this work is expected to shed light on the importance to risk of the retention capabilities of the waste and its packaging material. Expertise in geological sciences is now included in the qualification descriptions used in the recruitment of risk assessment personnel for work in the area of fuel cycle risk applications.

The suitability of a HLW package for ultimate disposal can be evaluated by considering: (1) the long-term stability of the waste package under repository conditions, and (2) design and engineered features of HLW packages with respect to safe handling during operation. The technical data needed to formulate the evaluation criteria on long-term stability of the waste package will be developed through material science and technology research. A limited material research program is currently underway; additional work will be initiated during the coming year to test the durability of various waste forms and waste package components and to identify thermodynamic, kinetic, and other physical/chemical processes and mechanisms by which the waste package material may degrade. Models will be developed to predict the long-term stability ($\sim$ 1000 years) of the waste package, based on the degradation mechanisms identified. The separate effects of each attribute such as temperature, pressure, solution chemistry, and radiation will be quantified with respect to overall material characteristics and provide a data base for quality control and assurance standards. This work will take several years, through FY 1987.

The state-of-technology for safe handling of radioactive packages such as spent fuels and low-level wastes will be applicable at least in part to HLW package design. However, the existing technology will be critically reviewed with respect to adaptability to HLW package handling such as underground operability. Consequences of various handling accidents will be evaluated for the establishment of package design criteria. Work will begin in FY 1982 and provide results in the 1984-1985 time frame.

12.3.2 Low Level Waste (LLW)

ACRS Comment (page 12-2)

The ACRS considers this a high priority item. Although the planned FY 1981 funding by NRC for LLW research once might have been considered to be adequate, certain recent events suggest this may no longer be true. Experience has shown the need for better bases to enhance the guidance of LLW management, such as in recovery operations after a reactor accident. The ACRS believes also that more research is needed to provide better rationale and wider technical bases for the development of detailed site selection criteria and better procedures to assess the suitability of sites and site practices in the licensing and operation of commercial shallow land burial facilities for LLW. This information is urgently needed in order that additional acceptable sites can be expeditiously selected, evaluated, and licensed. Sites that are acceptable for this purpose can provide the needed flexibility to accommodate large waste volumes and shipping route alternatives.

RES Response

Field tests to assess the suitability of selection criteria are underway at two existing eastern LLW sites. The studies include assessments of erosion rates, characterization of degraded wastes in trenches, hydrological measurements in both the saturated and unsaturated zones, comparison of distribution coefficients for different chemical species of radionuclides, including comparison of formation from field and laboratory tests, and the effects of vegetation on radionuclide migration. The data obtained from the field work will be used to test existing and proposed predictive models and to aid in identifying important parameters of sites and their environs relative to ensuring control of radionuclide migration and maintenance of the integrity of the facility. We plan to extend the studies to other sites, including western, arid sites. In addition, we will assess geophysical instruments and methods for site characterization and radiometric instruments for waste package and site monitoring. We are currently funding an evaluation study of pulsed radar methods for subsurface profiling of shallow land burial sites; it will be completed in FY 1981.

Current research is focused on factors that can improve the safety of shallow land burial through better trench cap design, grouting, compaction and maintenance to prevent creation of voids that can lead to collapse and infiltration of water. Ongoing tests of the effectiveness of improved trench cap designs will include the use of non-radioactive, non-toxic tracers of water movement. Improved long term isolation can be expected. Decommissioning (closure) is also specifically being studied.

ACRS Comment (page 12-2)

Specific areas of research needed relative to LLW management include:

(a) Techniques for reducing the volume of low-level waste as well as the exploration of alternatives to shallow land burial as a method for disposal.

(b) Field monitoring equipment to evaluate the acceptability of LLW packages as received at a disposal site, including their content of free standing liquids.

(c) Methods for negating the influence of chelating agents, commonly used in radionuclide decontamination operations, in terms of later migration of radio nuclides within and from a disposal site.

(d) Criteria for permitting public access to facilities formerly used in nuclear work, and development of associated field monitoring equipment to assure compliance with the criteria.

RES Response

(a) Elimination of unnecessary waste generation and a reduction in the volume of the waste once it is generated can help in ameliorating waste storage and disposal capacity shortages.

The decision to use specific volume reduction techniques is a complex one and requires an evaluation of many variables keyed to the specific wastes which the licensees expect to generate. This decision involves consideration of not only the obvious benefits, but also the adverse consequences. Because many of the volume reduction techniques are under development, operating experience in full-scale applications is limited. Volume reduction of some high activity wastes may yield a product which has a higher radioactivity concentration than is acceptable for disposal by shallow land burial. Utilization of volume reduction systems may also increase the requirements for maintenance activities and protection against occupational exposures. Most systems have restrictions concerning the type of waste handled, requiring segregation of some wastes which could reduce the performance of the system. The additional handling involved can increase occupational exposures. The potential for increase in airborne releases from volume reduction facilities must also be considered.

Numerous volume reduction studies have been initiated by DOE and industry. NRC is presently sponsoring a study to define and assess the processes available for reducing waste volume and to provide economic evaluations of several of these. The study is scheduled for completion in May 1980. Waste forms resulting from volume reduction methods, such as incineration, will be tested and evaluated under an ongoing NRC research program, as samples become available from DOE and commercial volume reduction development programs. The ongoing assessment of geologic alternatives to shallow land burial will identify by FY 1982 geologic alternatives judged to be feasible and cost effective. Beginning in FY 1983, field tests will be started to provide specific information for each disposal method to provide a basis for a comparison of the alternatives. This research will include considerations of site suitability, facility design and operation, monitoring, and facility closure for each of the alternative methods. During FY 1984-1985, there will be a transition from emphasis on research for shallow land burial to facilities in deeper geologic media. During that period, a feasibility study for engineered storage will also be initiated.

(b) A non-destructive method for measuring free-standing water in drums has been developed by Westinghouse, Hanford, under a DOE contract. We are presently considering a study that includes evaluation of the effectiveness and reliability of radiation monitoring methods.

(c) The draft regulation for disposal of low-level radioactive waste, 10 CFR Part 61, includes a provision that would require treatment of wastes containing chelating or organic material to eliminate the potential for interaction of radionuclides to form stable complexes that could increase the mobility of the radionuclides. NMSS is making a preliminary assessment of methods for achieving this requirement through a technical assistance contract. One of the objectives of this contract is to identify NRC's research requirements. These requirements will be taken into account in our RES program plans as they become identified.

(d) RES is now evaluating the adequacy of technical information required for permitting public access to decommissioned sites. In the meantime, we will continue to provide the Office of Standards Development with needed technical support in this area.

12.4 Recommendations

ACRS Recommendation (pages 12-2 and 12-3)

In view of the growth rate of NRC's research in this area (from $4.5 million in FY 1979 to $8.6 million proposed for FY 1980 to $13.6 million requested in FY 1981), the ACRS is still concerned about the adequacy or manpower and expertise available in RES and in the Office of Nuclear Material Safety and Safeguards (NMSS) to manage the program effectively and yet stay abreast of important developments that occur outside the NRC, for example, in DOE and in foreign countries. The ACRS believes that additional staff capabilities and technical expertise are urgently needed in this area and, therefore, recommends that the NRC be authorized and funded to expand the technical staff for this program by at least five people above the fifteen now proposed.

RES Response

We agree with and endorse these views of the ACRS. However, we also understand that the NRC as a whole has only limited resources, and the decision as to how they should be deployed best to meet the overall needs of the agency is one that can only be made by the Commission, with whose decision we concur.

Chapter 13. SAFEGUARDS

13.4 Recommendations

ACRS Recommendation (pages 13-2 and 13-3)

All of the projects proposed are needed to meet safeguards requirements. In many instances more rapid progress would be desirable; whereas slower progress would scarcely seem to be acceptable. The ACRS considers the proposed level of funding to be marginal, but adequate.

It is recommended that possible conflicts between desirable safeguards requirements and essential operational safety requirements be identified and resolved, and that human response in the context of proposed mechanical and procedural safeguards provisions should be studied. Work in these areas probably could be accommodated within the proposed budget by some curtailment of that presently planned for other projects under Alternative Strategies (Section 13).

RES Response

We agree that current spending is at about the minimum acceptable level. Our reasons for this judgment are related to the NRC user support philosophy which has heavily conditioned the NRC program. Research in evaluative methods, which has been the heart of our program, has a high degree of user acceptance and is being tested by the NMSS staff to determine the utility of these research products to support the implementation of upgrade rules, particularly 73.55. The staff has also incorporated research results as a means of addressing the inspection and enforcement problems of NFS, Erwin.

We also agree that more rapid progress would be possible if funding levels were higher, and we have user support requests to back this judgment. Slower progress connotes only a prolongation of the subjective judgment approach to evaluation of licensee response to upgrade rules in the areas of physical protection, contingency planning, and guard training.

The staff currently has no objective means by which the safeguards research efforts can be directly compared with general safety research or environmental research. Thus, assignment of priorities to our Safeguards program is by its nature subjective. However, the Committee's recommendations to curtail projects under "Alternative Strategies" may not be feasible. Alternative strategies traditionally cover safeguards research not directly associated with evaluation methods. The projects under this heading have always been strongly user-supported because they have been very responsive to urgent needs of the user offices. RES concurs with the NMSS position and recommends that no funds be deleted from this program in FY 1981.

Chapter 14. RISK ASSESSMENT

We appreciate the strong endorsement of our risk assessment program by the ACRS.

14.2 General

ACRS Comment (page 14-1)

The ACRS strongly supports the planned growth in this research program. Furthermore, in its recommendations for new directions in research (Chapter 2), the ACRS identified several additional areas which can logically be located in this program. These matters include studies of the courses of serious accidents, inadequacies in the single failure criterion, and the effects of the considerations of serious accidents on siting criteria. The ACRS therefore believes that this research program should receive some of the funding which can be reallocated from the proposed reduction in the support of LOFT.

RES Response

The present research program is not pursuing this general area just as the ACRS recommends. The ACRS recommended a series of analytical studies for the course of serious accidents, i.e., those beyond the design basis. RES has segmented this work into a number of areas, all related to the course of serious accidents. These include the IREP, mechanistic studies of plant accident sequences such as total system blackout, the work on filtered vented containments, and the development of instrumentation to help diagnose and follow the course of such serious accidents.

The IREP (Interim Reliability Evaluation Program) has been discussed with the ACRS. It is directed toward discovering the principal causes and sequences that lead to serious accidents. It should greatly increase our understanding of the courses but not of the phenomenology of the serious accidents.

Separate mechanistic studies have begun to analyze a few serious accident scenarios such as total station blackout, to discover the likely course of events and, in the case of blackout, the point of no return for damage to the core and other sytems. The work on filtered vented containments is directed toward that risk significant end of the scenario, the containment reaction. This work is analyzing core melt accidents from an intact geometry using MARCH-CORRAL.

The development of serious accident instrumentation is right now directed toward those parameters in serious scenarios which are needed for diagnosis of the accident situation and to monitor recovery from it.

Consequences of serious accidents will be studied for existing reactor sites to rank the sites and provide recommendations and input for siting criteria. Planning is complete, and initial efforts have begun in this area.

14.3 Comments

ACRS Comment (a) (page 14-1)

The Interim Reliability Evaluation Program has considerable potential for an important contribution to the improvement of the safety of existing reactors

and should receive high priority. The nuclear industry should initiate and place emphasis on its own concurrent program in order to evaluate more quickly reactor design aspects which can and should be improved in a timely fashion.

RES Response

The IREP was initiated to apply existing risk assessment techniques to each of the commercial light water reactor plants in the U.S. The objectives are to identify particularly high-risk accident sequences at individual plants and to develop a cadre of experienced practitioners of these safety analysis techniques. This program has been given the highest priority within the PAS. Funds for early initiation of this program during FY 1980 have been obtained by postponing planned new initiatives having a lower priority, and by reducing planned funding for selected projects where a slower rate of accomplishment could be tolerated. It will be necessary to have frequent contacts with the nuclear industry during the conduct of IREP, and we are exploring the possibility of requesting the owner of each utility to participate in the program by assigning an operations engineer to the plant IREP team. We agree strongly with the statement of the ACRS quoted above that industry should use probabilistic analysis and risk assessment techniques to assist in the evaluation of reactor designs and of operational history. PAS is also working on the development of standard procedures for IREP studies that will lend themselves to implementation by licensees as well as by NRC staff analysts and NRC contractor. There have already been a number of contracts with owners of reactors and their contractors to cooperate with industry efforts in risk assessment.

ACRS Comment (b) (page 14-1)

Priority should be given to an evaluation of flood models and to a more realistic examination of potential on-site and off-site effects of a large release of radioactive material, including possible decontamination measures.

RES Response

Within the methodology development effort of PAS/RES, we have been developing methods to evaluate the frequency of occurrence of severe floods. In addition, PAS/RES has been attempting to develop more realistic methods for examining the potential effects of a larger release of radioactive material. We believe that the priorities we have assigned are consistent with the ACRS recommendation in this area.

The General Reactor Safety Research (GRSR) program within RES is currently involved in a project on "Flood Hazards and Flooding Effects," which is related, in part, to the issue of floods causing serious reactor accidents resulting in large radioactive release and subsequent contamination through the liquid pathway. The essential elements of this study are: (1) probabilistic models of flood hazards incorporating mixed events and site locations, (2) probabilistic models for flood protection structures, and (3) probabilities of radioactive release due to failure of protective structures. Operating plants will be categorized, based on site locations, and they will be evaluated further to determine the probability of radioactive release as functions of water levels. These probabilities will then be compared with the flood protection available in current design in order to determine the available margins of protection.

ACRS Comment (c) (page 14-1)
The topics relating to reactor systems analysis and licensing support should include the early development of an improved alternative to the single-failure criterion.

RES Response
The various reliability analyses already done and now underway such as the IREP continue to show the need for an alternative to the single failure criterion. It is not clear at this time whether the alternative path should be toward quantitative reliability standards for review or for a multiple failure deterministic approach whose structure is based on generic reliability analyses.

It is now intended that at each major milestone in the IREP, suggestions will be made to NRR by RES on changes to the review process. The early experience gained in the Auxiliary Feedwater System studies in 1979 produced reliability based changes in the deterministic licensing review of those systems. Later discussions have raised the possibility of scaling the number and nature of postulated failures to the probability of the initiating event being considered.

RES is also pursing the development of quantitative risk standards not only for health effects but for plant damage or failure evaluations. If such standards appear useful in the review process soon, RES will propose their adoption either to replace or to supplement deterministic review processes.

ACRS Comment (d) (page 14-1)
The work on nuclear fuel cycle risk should include a focus that will provide the NRC with improved bases for the promulgation of criteria for ultimate disposal of high level wastes.

RES Response
The work on nuclear fuel cycle risk already includes a focus specifically designed to provide improved bases for the promulgation of criteria for ultimate disposal of high level wastes. This focus is the Interoffice Waste Management Modeling Group (IWMG). The IWMG is composed of staff from NMSS, OSD, RES, and NRR, and its objectives are to: (a) build in-house expertise with the risk assessment models, (b) understand the capabilities and limitations of the risk assessment methodology, and (c) document the results of a series of increasingly complex problems culminating in the safety assessment of a repository site.

ACRS Comment (e) (page 14-2)
Other matters which should be addressed include projects to provide information needs outlined in the "Report of the Siting Policy Task Force" (NUREG-0625), and to provide data for assessing the advantages and disadvantages of multi-unit versus single-unit sites.

RES Response
A project is underway to validate the site population data to be used in assessing the site criteria. New meteorological data for the sites is also

being collected. Planning is underway to develop the multi-objective programming approach for assessing the costs/benefit of various siting alternatives including multi-unit versus single-unit sites.

ACRS Comment (f) (page 14-2)
The ACRS has previously recommended that the NRC attempt to develop quantitative risk acceptance criteria for public comment and for review by the Congress itself. The ACRS believes that this effort should be given high priority, well beyond that afforded it thus far by the NRC.

RES Response
The present methods of establishing quantitative risk criteria essentially involve consensus estimation by committee or special task force. Development of quantitative risk criteria by expressed preference and decision-theoretic approaches, developed in the earlier "How Safe is Safe Enough" project needs to be applied to broaden the methods of arriving at quantitative criteria. We have developed a new project to accomplish this. This project would have two distinct phases: (1) the expressed preference phase in which surveys were taken and analyzed, and (2) the decision-theoretic phase. This project would essentially apply the general methods discussed in the "How Safe is Safe Enough" project. This program is being actively supported by a variety of professional disciplines and professional engineering societies. Peer review functions are to be provided by the National Science Foundation, the National Academy of Science, and the American Statistical Association. Interest has been expressed by the Organization for Economic Cooperation and Development to participate actively in arranging for international discussions of the fundamental issues related to the development of numerical risk criteria. These discussions will be exceptionally beneficial in establishing guidelines for the required methodologies and techniques, and in attaining ultimate international accord.

We believe that this area is being given appropriately high priority in our overall program, and we will be discussing the possible studies with the ACRS in the near future.