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

JUN 21 1984

Dr. Mary Shoaf
Plasma Physics Lab.
P. O. Box 451
Princeton University
Princeton, NJ 08544

Dear ~~Dr. Shoaf~~: Mary.

Enclosed per your recent request to Bob Bernero are copies of 189's for FY 84 source-term related research. I have also enclosed a draft supplementary package which indexes the research projects to the major BMI-2104 codes, as well as related improved codes now under develoment. This package was prepared following the Berkeley meeting in anticipation of the needs of the Study Group. The supplementary package includes a brief summary of the objectives of each project and the most recent financial data for FY 84, 85, and 86.

The financial information for many of the 189's are out of date. Some of the projects in the supplementary package do not have a 189 because they are either new starts in FY 84 or international projects. Duplicate copies of all of this material will be available Thursday for the APS attendees.

If you have any questions, please call me on FTS 427-4737.

I look forward to seeing you on Thursday.

Sincerely,

Mel Silberberg

M. Silberberg, Assistant Director
for Research and Technical Support
Accident Source Term Program Office
Office of Nuclear Regulatory Research

cc: R. Bernero

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SUMMARY OF PLANS
FOR SOURCE TERM RELATED RESEARCH

ENCLOSURE 1 - Information on Computer Codes Related to the Source Term Reassessment

ENCLOSURE 2 - The NRC Analysis Program for Severe Accidents in LWRs

ENCLOSURE 3 - Brief Description of Related Research Projects

ENCLOSURE 4 - Project and Budget Proposal for NRC Source Term Related Research Projects (Forms 189)

U.S. Nuclear Regulatory Commission
Washington, D.C.
June 28, 1984

ENCLOSURE 1

INFORMATION ON COMPUTER CODES RELATED TO THE SOURCE TERM REASSESSMENT

Figure 1 identifies the Battelle suite of codes that was used for the source-term calculations in BMI-2104. In the following paragraphs we will try to answer the question: What new codes might replace the Battelle suite of codes and lead to improved results in the near future? We will also indicate how our research dollars are being spent for the development and validation of these codes. Two approaches will be described below.

One approach is the development of the MELCOR risk-assessment code, which is less detailed, but faster running than the Battelle codes. Thus, the intent of the MELCOR code is not to provide detailed models of all thermal, physical, and chemical effects associated with core-melt accidents, but rather to provide an approximate method to estimate the timing and extent of fuel degradation, fission product and aerosol release, and containment failure for an overall risk assessment.

The other approach is the development or modification of a collection of state-of-the-art codes on separate effects. These codes to the extent possible mechanistically treat the details of the thermal, physical, and chemical effects associated with fission product release and transfer from fuel, transport and deposition in reactor-vessel and primary-system components (piping, steam generator, etc.), and transport and deposition in the containment. Some of these codes are too complex to be used for repeated calculations in a suite of codes; others, with some modification, could replace present components of the Battelle suite.

MELCOR is a risk-assessment code that includes thermal-hydraulics modules, fission product behavior modules, ex-plant consequence modules, and economic consequence modules. As such, MELCOR has a broader scope than the BMI-2104 suite of codes, but it contains much less mechanistic detail. MELCOR is designed to be fast running, which is necessary for the large number of runs required in a risk assessment study. It is intended to replace the WASH-1400-generation codes, MARCH, CORRAL/MATADOR, and CRAC. While those codes had been

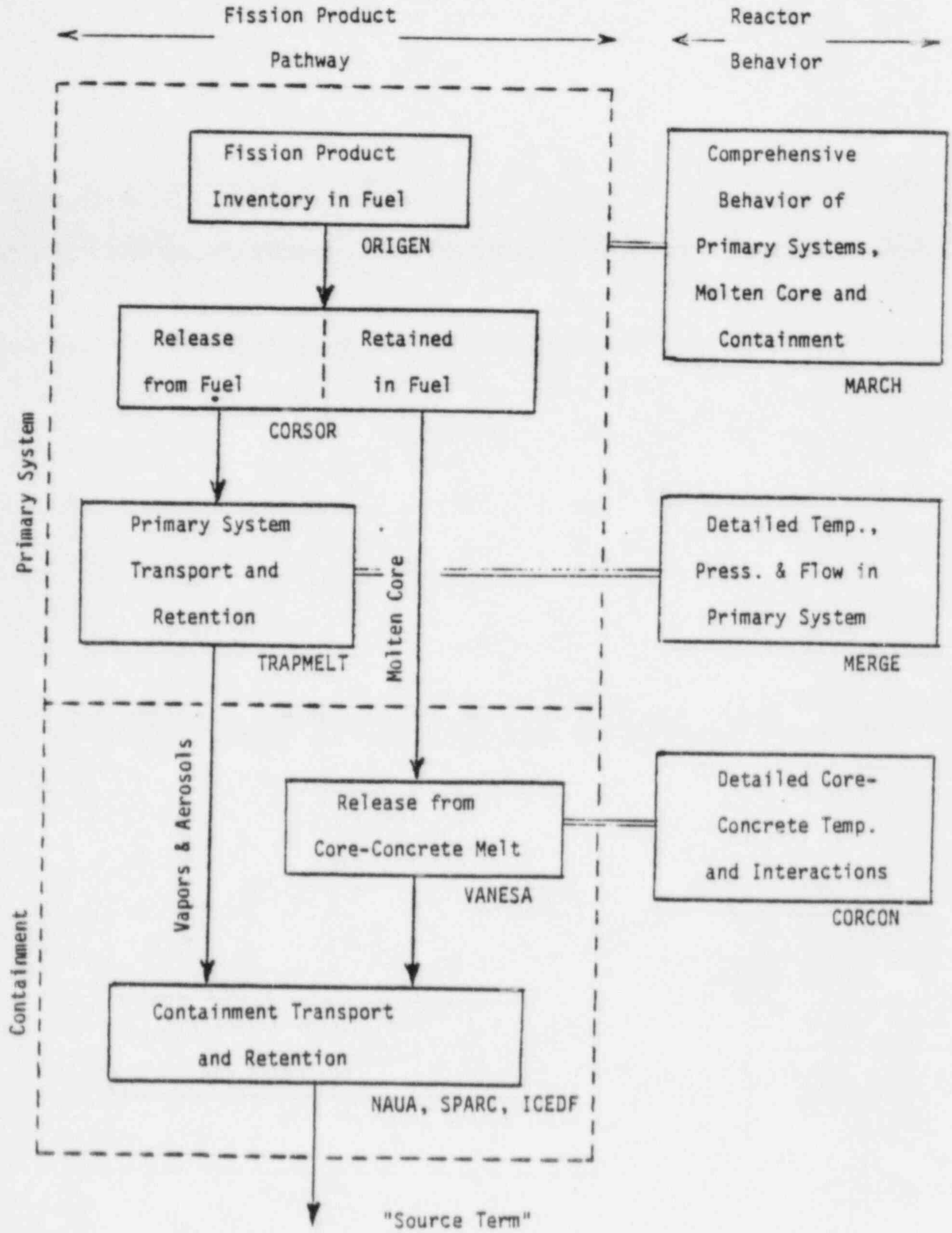


Figure 1. Battelle suite of codes as used in BMI-2104.

improved since the original WASH-1400 analyses, MELCOR will have improved, consistent treatments of severe accident phenomena and enhanced capabilities for sensitivity studies.

MELCOR will, in fact, employ slimmed-down or pseudo versions of some of the state-of-the-art codes to be mentioned later, and it will be extensively benchmarked to those more mechanistic codes. The initial version of MELCOR is scheduled for completion in September 1984, and QA testing will be done in FY 85. A running version of MELCOR with a full complement of models is scheduled for September 1985. Benchmarking will start in FY 85 and continue through FY 86.

Table 1 lists state-of-the-art codes on separate effects that could replace components of the Battelle suite of codes. A number of codes (e.g., TRAPMELT, CORCON, VANESA) appear in Table 1 as both BMI-2104 codes and newer codes. These codes are still under development or validation and improved versions can be expected.

Table 2 references the corresponding research projects for each of these newer codes. The research projects include code development and validation as well as experimental programs to provide a data base for the phenomenon in question. A brief description of each of these projects along with funding levels for FY 84, FY 85 and FY 86 is enclosed. Also enclosed is a draft paper entitled, "The NRC Analysis Program for Severe Accidents in LWRs," that gives additional information on our severe accident code strategy.

Linkages between some of the newer codes have been (or are being) developed. For example, a TRAC/MIMAS/MELPROG link has been developed to study the details of core melt progression. Melt progression and related phenomena have been shown by our source-term work to be very important in determining containment loads as well as fission product availability. A link between RELAP-5 and SCDAP, on the other hand, addresses core behavior up to the point of loss of fuel-rod geometry and should be more valuable for studying terminated accidents and mitigation features.

One could certainly conceive of a linkage between TRAC, MIMAS, MELPROG, VICTORIA, TRAPMELT, CORCON, VANESA, and CONTAIN. Such a linkage would

Table 1. Newer codes that have the potential for upgrading components of the Battelle suite of codes used in BMI-2104.

Description	BMI-2104	Newer Code
RCS Thermal-Hydraulics	MARCH, MERGE	TRAC, RELAP-5
Fuel Heatup and Degradation	MARCH	SCDAP, MIMAS, MELRPI, MELPROG
Fission Product Release from Fuel (in vessel)	CORSOR	GRASS, VICTORIA
RCS Fission Product Transport	TRAPMELT	TRAPMELT
Molten Fuel Interaction with Coolant	MARCH	WISCI
Debris-Concrete Interactions	CORCON	CORCON
Fission Product Release from Core-Concrete Melt (ex-vessel)	VANESA	VANESA
Containment Thermal-Hydraulics	MARCH	CONTAIN
Hydrogen Behavior	MARCH	CONTAIN (HECTR)*
Containment Fission Product Transport	NAUA, SPARC, ICEDF	NAUA, CONTAIN (SPARC, ICEDF, MAEROS)*

* Subroutines in CONTAIN

Table 2. Corresponding Research Project to Develop or Upgrade Newer Codes.

Newer Code	Description	Research Project Number
TRAC, RELAP-5	RCS Thermal-Hyrdaulics	3, 4, 7, 10, 14, 26, 31
SCDAP, MIMAS, MELRPI, MELPROG	Fuel Heatup and Degradation	1, 2, 3, 4, 5, 6, 8, 9, 10, 12, 13, 14, 15, 16, 17, 18, 19, 26, 28
GRASS VICTORIA	Fission Product Release from Fuel (in vessel)	4, 5, 9, 10, 11, 12, 13, 14, 15, 16, 18, 19, 20, 22, 26
TRAP-MELT	RCS Fission Product Transport	5, 13, 14, 15, 18, 20, 21, 22, 26, 30, 31, 35
CORCON	Debris - Concrete Interactions	7, 26, 27, 28
VANESA	Fission Product Release from Core-Concrete Melt (ex-vessel)	5, 20, 26, 27, 28
CONTAIN (HECTR)	Hydrogen Behavior	4, 16, 22, 26, 29, 34
NAUA, CONTAIN (SPARC, ICEDF, MAEROS, QUICKM)	Containment Fission Product/Aerosols Transport	16, 20, 22, 25, 26, 29, 31, 33, 34

constitute a complete substitute for the Battelle suite of codes and could be used for future detailed source term analysis. At this time, however, such a linkage is not planned. Instead, the newer state-of-the-art codes will be used individually or in small groups to understand in some detail the important physical phenomena affecting severe accidents. These codes will in turn be used to benchmark the fast-running MELCOR code, which will be used for the large number of code runs required for risk assessment.

ENCLOSURE 2

THE NRC ANALYSIS PROGRAM
FOR SEVERE ACCIDENTS IN LWRs

THE NRC ANALYSIS PROGRAM FOR SEVERE ACCIDENTS IN LWR'S

George P. Marino

I. Introduction

The analysis of severe accident consequences for light water reactors has been a project of major importance in the NRC prior to and, even more vigorously, after the event at Three-Mile Island Unit 2. In the development of a methodology for ascertaining the consequences of such events for use in risk analysis and source term studies, one must appreciate beforehand that the nature and complexity of the phenomena involved limit the extent to which an exact analysis capability is possible. An accurate analysis capability would require a set of experimentally validated models capable of treating in great detail all the phenomena occurring in the multi-phase, multi-component, nuclear steam supply system (NSSS) over temperature ranges from 300C to 2800C, pressure ranges from 15 psi to 2350 psi, and time periods over days, weeks, and possibly months. As a practical matter, the capabilities of an analysis technique of this type must be limited since:

1. Fully integral tests to validate all the models under all possible conditions for all types of plants would be prohibitively expensive, take decades to accomplish, and be difficult to evaluate.

2. Risk analyses and the necessary sensitivity and uncertainty analyses which are part of them require fast-running analysis codes so that many sequences for many plants can be made in a reasonable time frame. Therefore, fast-running computer codes require extensive model simplification to achieve this goal.

3. An exact quantification of the technology of all the processes expected to occur is not possible since uncertainties will always exist in experimental data for material properties, physical and chemical parameters, the models themselves, and the sequences predicted to occur.

Therefore, in order to make the goal of quantifying the consequences of severe accidents tractable, one must approach the problem in a more realistic manner. That is, recognizing that (1) uncertainties will always exist, (2) complete experimental validation is not feasible, and (3) analysis codes for risk studies must be fairly fast (i.e., less than a few hours on a CRAY for a given sequence and plant), a methodology must be developed that will result in a fast-running analysis code that evaluates the entire nuclear steam supply system and is composed of phenomenological models whose uncertainty can be quantified. Such a goal requires the accomplishment of two major prerequisite tasks; namely:

(1) The establishment of a data base to the extent practically possible in the temperature and pressure ranges of interest and,

(2) The development of specialized detailed analysis codes for specific components of the NSSS such as the primary system T/H, the core, and the containment. These codes should be validated using item (1) above.

Given the above two elements, the final goal of a fast-running overall NSSS code containing quantified uncertainties in its output can be achieved by benchmarking its simplified models against the corresponding models in the more mechanistic analysis codes. The uncertainty in the latter codes will be quantified -- to the extent possible -- by the established data base. It is to be expected that the uncertainty of a given model in the overall code will be greater than or equal to that in the more mechanistic code because of the need for fast-running times. However, whatever the magnitude of the uncertainty, it will be quantified and the results can therefore be used in decision making.

II. Current Program of the NRC

In order to achieve its goal of developing a fast-running NSSS analysis code with quantifiable uncertainties for analyzing the consequences of severe accidents in LWR's, the Office of Research established the Severe Accident Research Program (SARP) as part of its long-range plan shortly after the TMI-2 event. This program consists of approximately seventy research contracts to develop data bases for core behavior; primary system behavior; containment behavior; fission product release, transport, deposition, and re-evolution; hydrogen behavior; and detailed analysis codes for all of the above including severe accident sequence analysis (SASA); detailed risk analysis; and risk reduction studies. The current program utilizes approximately one-third of the annual Research budget, and is scheduled for completion in the 1986/87 time frame. The risk analysis code mentioned above (i.e., the fast-running NSSS computer code with quantified uncertainties) is under developm t and has been given the name MELCOR. The initial, unvalidated version, is scheduled for completion by the end of FY 1984.

The development of the supporting mechanistic analysis codes for benchmarking and quantifying MELCOR has been underway since 1980. This set of codes consists of a few newly developed codes, previously developed codes (such as TRAC and RELAP5), and linked packages of new and previously-developed codes. An extensive effort has been made to utilize previously developed analysis packages to avoid "re-inventing the wheel" and, of course, to minimize expenditures. In order to clearly show how all these codes (there are 25 of them) fit into the general scheme outlined in the Introduction, it will be necessary to graphically illustrate their function in relation to the NSSS. In order to do this we must first separate those codes intended for detailed analysis for benchmarking MELCOR from the current "risk" codes that MELCOR is intended to replace.

A. Current NRC Risk and Source Term Analysis Codes

This group of codes is currently being used in the current NRC evaluation of the source term for selected plants and sequences. Earlier versions of some of the models were used in the WASH-1400 study. It must be remembered that these codes are not "detailed" in the sense mentioned above; i.e., they are fast-running codes which necessarily requires considerable use of simplifying assumptions, empirically-based correlations, and user input options. They also have not had the benefit of an adequate data base from which to validate and assess their simplified models.

However, they represent our current best integrated analysis capability for source term analyses until the SARP program is completed and MELCOR has been fully validated and quantified. Table I gives a list of the codes and their application to the NSSS. Note that this family of codes is commonly referred to today as the "Battelle Suite of Codes" since the Battelle-Columbus laboratory has been prime contractor for their application. It should be noted that most of the models in this code series will be used directly in MELCOR with only slight changes. These models are ORIGEN, TRAP/MELT, VANESA, SPARC, CORCON, and ICEDF. Major improvements are expected for the MARCH, MERGE, CORSOR, and containment aerosol applications. MELCOR will, however, contain additional models for containment temperature and pressure response as well as ex-plant consequence models.

B. "Mechanistic Specialized Codes for Benchmarking MELCOR and Special Applications

This group of codes represent best-effort modeling with little emphasis on speed, but great emphasis on model accuracy. In other words, these codes are intended to represent the best state of technology for specialized phenomena and for applications to specific areas of interest. These codes, when completed, will be maintained as a best-estimate base of modeling expertise to be used when the necessarily simplified models in MELCOR are judged inadequate for highly specialized applications. The general philosophy being applied here is that the NRC staff must maintain "state of knowledge" expertise to be able to do in-depth studies of important phenomena whenever the need arises. However, as stated above, the primary application of this code group is to benchmark and

quantify the modeling uncertainties that will inherently be present in MELCOR, and, incidentally, to the Battelle Suite of Codes. Table II presents a summary of this group and where they are applied in the NSSS. With regard to the availability dates given in Table II, it should be understood that these codes will be continually updated and improved beyond that date as new experimental data for validation become available.

C. Linkages of Mechanistic Codes for Special Applications Where "Feedback" is Important

There are many situations where the input period from one specialized code is affected by the output of the receiving code over a given time period. For such cases, more accurate modeling is accomplished by "linking" the codes together so that input/output information can be interchanged during the run. An example of this is a TRAP/MELT calculation which relocates a significant part of the decay heat source into the upper plenum. This relocated heat source will affect the MERGE input on flows and structures in the upper plenum and other parts of the primary system and, therefore, future TRAP/MELT computations. Therefore, it is essential to develop hard links between the mechanistic fuel behavior modules (SCDAP, MELPROG) and system T/H codes (RELAP, TRAC) because of the intimate coupling between fuel degradation and the T/H behavior of the reactor coolant system. Note that MELCOR will be a fully integrated code package with inter-model feedback included as an inherent part of the programming.

For the detailed code series a few major code links are planned and are currently being implemented. They are shown in Table III. Completion dates for these linkages have not been firmly established, but they are expected to be available by mid 1985. The major reason for the SCDAP/RELAP5 link in addition to the TRAC/MIMAS/MELPROG link is that detailed modeling of the fuel pins is necessary for attentuated accidents but may not be necessary for "core on the floor" accidents. Therefore, using the less detailed MIMAS code in place of SCDAP will increase computational speed dramatically for non-attenuated accidents scenarios. Moreover, the SCDAP/RELAP5 link will give early capability to analyze both PWR and BWR systems for these less drastic events. It should be noted that the above

procedure is very cost-effective since the system codes have already been developed and validated as part of the ECCS research program, and the linkages can be accomplished in a very short time to give essentially universal applicability. That is, capability for PWR's and BWR's for "core on the floor events" and for attenuated events such as the accident in TMI-2. Finally, the linked codes can be used to benchmark a wide range of MELCOR's integrated package and quantify the effects of feedback in severe accident analyses.

III. Summary

The code development plan outlined above will provide the NRC with the analysis capability required for decision-making on severe accidents in LWR's. An integrated risk code is provided - MELCOR - for large-scale PRA and source term studies as well as special-application, more mechanistic codes for less broad, more specific decision-making processes. The plan utilizes to the broadest extent possible the codes developed for other purposes as well as the extensive new data base being developed under SARP for NSSS behavior under severe accident conditions.

Table IV summarizes how all these codes are used in NSSS analyses by classifying them by NSSS component application and by phenomenological categories. Finally, Figure 1 summarizes the codes graphically to show how they will "fit" together to accomplish their intended purpose. Note that some of the current "Battelle Suite" of codes will be used directly in the mechanistic set and will not be "re-invented". Note also that the CONTAIN code consists of many models from current codes used in a subroutine capacity such as CORCON, HECTR, etc.

TABLE I

BATTELLE SUITE OF CODES

FOR RISK AND SOURCE-TERM STUDIES

NAME OF CODE	APPLICATION IN THE NUCLEAR STEAM SUPPLY SYSTEM
ORIGEN	MODELS FISSION PRODUCT INVENTORY IN THE CORE PRIOR TO SCRAM
MARCH 2.0	PRIMARY SYSTEM T/H CORE PHENOMENA (CONTAINMENT T&P)
MERGE	IN-VESSEL GAS FLOW AND HEAT TRANSFER TO STRUCTURES. USED AS AN INTERFACE BETWEEN MARCH & TRAP/MELT
TRAP/MELT	MODELS FISSION PRODUCT AND AEROSOL TRANSPORT AND DEPOSITION WITHIN THE REACTOR COOLANT SYSTEM
CORSOR	MODELS FISSION PRODUCT AND AEROSOL RELEASE FROM THE CORE. AN EMPIRICAL CODE BASED UPON EX-PILE, FISSION-PRODUCT RELEASE EXPTS
CORCON	MODELS EX-VESSEL MOLTEN CORE INTERACTION WITH REACTOR CAVITY BASEMAT MATERIAL
VANESA	MODELS FISSION PRODUCT AND AEROSOL RELEASE DURING MOLTEN CORE/BASEMAT INTERACTION
NAUA-4	MODELS AEROSOL BEHAVIOR IN THE CONTAINMENT
SPARC	MODELS AEROSOL RETENTION IN SUPPRESSION POOLS (BWR ONLY)
ICEDF	MODELS AEROSOL RETENTION IN PWR ICE-CONDENSER CONTAINMENT SYSTEMS

TABLE II

DETAILED NRC SEVERE ACCIDENT COMPUTER CODES TO BENCHMARK AND QUANTIFY MELCOR

NAME OF CODE	APPLICATION IN THE NUCLEAR STEAM SUPPLY SYSTEM	AVAILABILITY DATE
SCDAP	DETAILED CORE BEHAVIOR TO LOSS OF ROD GEOMETRY, I.E., TO ABOUT 2400K	OCTOBER 1984
MELPROG	DETAILED CORE BEHAVIOR THROUGH MELTDOWN AND EXIT THE REACTOR PRESSURE VESSEL	DECEMBER 1984
MIMAS	SIMILAR TO SCDAP-BUT MUCH LESS DETAILED--USED AS INPUT TO MELPROG FOR UNATTENUATED "CORE ON THE FLOOR" EVENTS	1-D VERSION COMPLETE 2-D VERSION IN DECEMBER 1984
FASTGRASS	DETAILED FISSION PRODUCT RELEASE FROM INTACT FUEL USED IN SCDAP & MELPROG	OCTOBER 1984
TRAP/MELT	SEE TABLE I - THESE MODELS WILL BE MODIFIED & USED IN SCDAP, MIMAS, AND MELPROG	COMPLETED
VICTORIA	DETAILED FISSION PRODUCT AND AEROSOL RELEASE FROM MOLTEN FUEL FOR USE IN MELPROG	OCTOBER 1985
CONTAIN	INTEGRATED DETAILED CONTAINMENT MODEL: IE. IT USES SUBMODELS FOR T/H, AEROSOL AND FISSION PRODUCTS (MAEROS), CAVITY MODELS (CORCON, MEDICI, VANESA), HYDROGEN BURNING (HECTR) & ESF MODELS	JUNE 1984
HECTR	SEE CONTAIN CODE ABOVE. HECTR MODELS HYDROGEN BEHAVIOR IN CONTAINMENT. TREATS DEFLAGRATIONS, SPRAYS, HEAT TRANSFER, IGNITERS, SUPPRESSION POOLS, SUMPS, AND FANS	DECEMBER 1984

TABLE III

DETAILED NRC CODE LINKS

FOR SEVERE ACCIDENT ANALYSES*

CODES LINKED	APPLICATION TO NSSS
SCDAP/RELAP5	INTEGRATED PRIMARY SYSTEM/PARTIALLY DEGRADED CORE BEHAVIOR FOR ATTENUATED ACCIDENTS SUCH AS TMI-2
TRAC/MIMAS/MELPROG	INTEGRATED PRIMARY SYSTEM/MOLTEN CORE BEHAVIOR FOR "CORE ON THE FLOOR" EVENTS.

*NOTE: EACH PACKAGE WILL CONTAIN FULLY INTEGRATED FISSION PRODUCT AND AEROSOL RELEASE, TRANSPORT AND DEPOSITION MODELS FROM FASTGRASS, VICTORIA, TRAP/MELT, AND VANESA

TABLE IV

SUMMARY OF NRC SEVERE ACCIDENTS

COMPUTER CODES

(MELCOR MODELS ALL)

NSSS COMPONENT	RISK CODES	MECHANISTIC CODES
PRIMARY SYSTEM	MARCH, MERGE TRAP/MELT	TRAC, RELAP5, TRAP/MELT
CORE	MARCH, CORSOR	SCDAP, MELPROG, MIMAS, FASTGRASS, VICTORIA, TRAP/MELT
REACTOR SUMP	CORCON, VANESA	CORCON, VANESA, MEDICI
CONTAINMENT	NAUA-4, SPARC, ICEDF, MARCH	CONTAIN (INCLUDES SUMP CODES)

BY PHENOMENOLOGICAL CATEGORY

T/H CODES: MARCH, TRAC, MERGE, RELAP5

FISSION PRODUCT RELEASE CODES: CORSOR, FASTGRASS, VICTORIA, VANESA

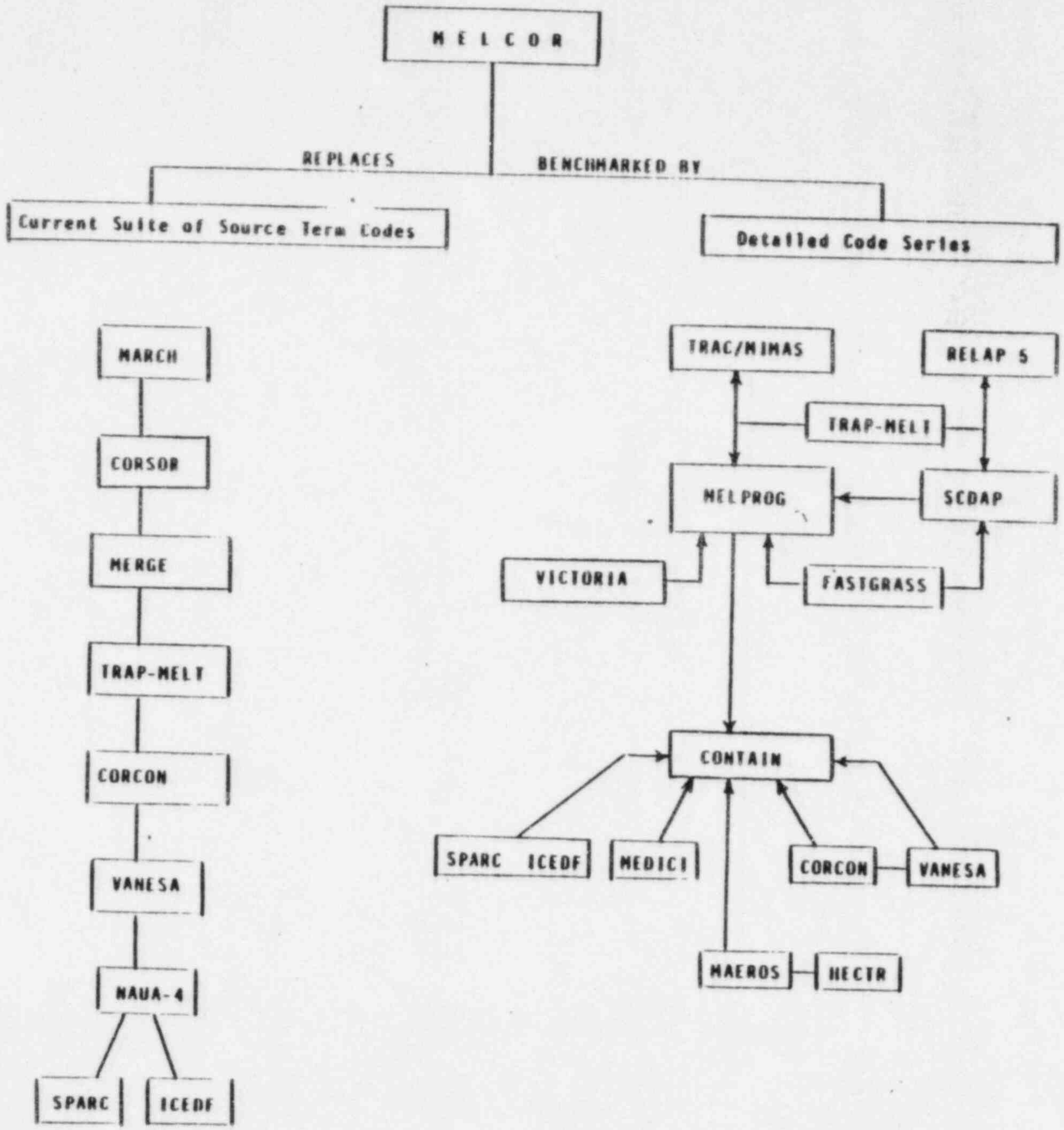
FISSION PRODUCT TRANSPORT AND DEPOSITION CODES: TRAP/MELT

AEROSOL BEHAVIOR CODES: NAUA-4, MAEROS (IN CONTAIN)

REACTOR CAVITY MODELS: CORCON, MEDICI

HYDROGEN BEHAVIOR CODES: HECTR (IN CONTAIN)

ESF CODES: SPARC, ICEDF (BOTH IN CONTAIN)



MELCOR
REPLACES
BENCHMARKED BY
Current Suite of Source Term Codes
Detailed Code Series
MARCH
CORSOR
MERGE
TRAP-MELT
CORCON
VANESA
NAUA-4
SPARC
ICEDF
TRAC/MIMAS
RELAP 5
TRAP-MELT
MELPROG
SCDAP
VICTORIA
FASTGRASS
CONTAIN
SPARC ICEDF
MEDICI
CORCON
VANESA
MAEROS
HECTR

ENCLOSURE 3

BRIEF DESCRIPTION OF RELATED

RESEARCH PROJECTS

Project 1 (FIN NO. B2455)

TITLE: SEVERE CORE DAMAGE MATERIALS PROPERTY TEST

OBJECTIVES: The primary purpose of this program is to generate data for use in modeling the behavior of fuel exposed to high temperature transients. A qualitative assessment of the so-called runaway Zr-O_2 reaction will also be made. Such an assessment could be important to understanding the severity and consequences of fuel damage where the maximum heat input is expected from reaction with steam.

Program tests are intended to provide the following data:

(a) Weight gain due to oxygen uptake as a function of temp.
(b) Rate constants for Zr/H_2O reactions as a function of temp.
(c) H_2/H_2O content as a function of time for a given temp
(d) H_2/H_2O content as a function of time during transient oxidation
(e) Uncontrolled temperature rise as a function of initial heating rate
(f) Effect of H_2/H_2O on K_p
(g) Effect of UO_2 on Zr/H_2O reaction rates
(h) Viscosity of Zr-UO_2 as a function of temp
(i) H_2 production from Zr-UO_2 up to 1700°C
(j) H_2 production from Zr-UO_2 from 1700°C - 2000°C

BUDGET ($K):	FY 84	FY 85	FY 86
	240	300	300

Project 2 (FIN NO. B2277)

TITLE: NRU COOLANT BOILAWAY AND DAMAGE PROGRESSION (CBDP) TESTS

OBJECTIVES: To provide well characterized full-length in-reactor test data on the severe fuel damage behavior of prototypic fuel rod clusters during and after coolant boilaway. In-reactor severe fuel damage tests on full-length prototypic fuel rod clusters are needed to validate oxidation and other models based on data from separate effects or short-length in-reactor tests.

The NRU CBDP tests will:

(a) evaluate length effects on damage progression
(b) determine the relationship of other SFD programs to NRU prototypic experimental test conditions measured from full length tests, including temperatures, temperature gradients, steaming rates, hydrogen generation, damage progression
(c) evaluate instrumentation performance under prototypic environments and temperature gradients.

The three tests in the current NRU SFD program are as follows:

MT-6, FY 84, 21-rod clad ballooning and low-temperature oxidation test;
FLHT-1, FY 85, 12-rod SFD rapid oxidation test to 2150;
FLHT-2, FY 86, 12-rod SFD rapid oxidation test to 2500K.

BUDGET ($K):	FY 84	FY 85	FY 86	
	2466	1450	2250	(NRC funds only)

Project 3 (FIN NO. A7303)

TITLE: TRAC/MELPROG INTEGRATION

OBJECTIVES: To develop two-dimensional thermal-hydraulic degraded-core models and a subcode to be used as a module in MELPROG under development at SNL. This project is an extension of the MIMAS code (developed at LANL for DOE) in which one-dimension thermal-hydraulics was used for degraded-core modeling. As part of the project, TRAC-PWR is being used to calculate flow recirculation in a PWR vessel; linking TRAC-PWR with MELPROG will be explored in the future to provide analysis capabilities for the reactor coolant system.

BUDGET ($K):	FY 84	FY 85	FY 86
	328	330	350

Project 4 (FIN NO. A1389)

TITLE: ACRR SOURCE TERM EXPERIMENTS - PLANNING

OBJECTIVES: Plan for a series of experiments in ACRR on fission-product release from reactor fuel, the chemical form, and aerosol formation under in-core severe-accident conditions at temperatures up to fuel melting. The purpose of these experiments is to provide a data base for the development of fission-product release-r models for the conditions of in-vessel severe fuel damage and core-melt progression. This planning is to include analysis of experiment requirements and diagnostics capabilities and engineering analysis of the proposed experiments through the preliminary design phase. Explicit attention will be given to obtain results for the following phenomena: release from solid fuel, release during fuel liquefaction, and release following fuel slumping. A report on this planning is to be submitted to NRC to form a basis for an NRC decision on whether to carry out these experiments as a major part of the alternative program to the PBF Phase II tests. Forseen is a possible program of ten experiments over a period of three years that includes development time.

BUDGET ($K):	FY 84	FY 85	FY 86
	300	920	2000

Project 5 (FIN NO. A1227)

TITLE: HIGH TEMP FISSION PRODUCT CHEMISTRY

OBJECTIVES: To investigate the high temperature chemistry of fission products in the vapor phase following release from the fuel. To determine the chemical interaction of fission products with RCS surface materials. To provide data for development of fission product transport models.

This project focuses on the kinetics of the chemical retention processes for fission products in the primary system during severe accidents. (Physical retention processes are being studied elsewhere). Research in FY 85 and 86 will include the study on the effects of radiation and pressure on the reaction kinetics.

BUDGET ($K):	FY 84*	FY 85	FY 86
	980	400	400

*The FY 84 funding includes 600K for the QUEST-I study.

Project 6 (FIN NO. A1335)

TITLE: LWR DEBRIS FORMATION & RELOCATION, MELT PROGRESSION, AND FISSION PRODUCT RELEASE

OBJECTIVES: To develop a data base and verified analytical models on the formation and relocation of LWR severely damaged fuel (core debris) and fission-product release applicable over the range of the risk-significant severe accident sequences. Separate-effects experiments up to full fuel-melt temperatures (3100K) are to be performed in the ACRR test reactor to give continuous-time data on the development of severe fuel damage, on debris formation by quenching at different times, on hydrogen generation, on debris relocation processes, and on fission-product release. Phenomenological models of the governing processes involved in the development of core-debris relocation, core-melt progression and in-vessel fission-product release are to be developed from the results of these experiments, the severe fuel damage tests in PBF, other in-pile tests, and laboratory experiments in both U.S. and foreign countries.

These experiments all use cinematography for time-continuous measurements of surface temperatures and of the damage-progression processes. The nine-experiment program includes four experiments on debris formation and relocation under core-uncovery conditions, two experiments on debris formation under reflood-quench conditions, and three experiments with pre-irradiated fuel on melt progression and fission-product release under in-vessel melt-progression conditions. Two experiments will be performed in FY 84, four in FY 85, and the final three experiments with pre-irradiated fuel in FY 86. This program is part of the integrated severe fuel damage research program of the Fuel Systems Research Branch.

BUDGET ($K):	FY 84	FY 85	FY 86
	2000	1500	2000

Project 7 (FIN NO. A1340)

TITLE: LWR CORE DEBRIS COOLABILITY

OBJECTIVES: To develop a data base on the dry-out coolability limits by reflooding and the post-dry-out behavior of LWR core debris by performing a series of experiments in the ACRR test reactor. To develop and verify relevant phenomenological debris-coolability models and codes for use in safety assessment. Emphasis is to be on verification of the existing LMFBR debris-bed coolability models for LWR specific conditions, in particular, high pressure, deep beds, coarse LWR debris, and inlet coolant flow as well as for the different coolant properties. All three experiments in this program utilize fission heating (simulating decay heating) of the artificial debris beds in ACRR, relatively deep (50cm) debris beds, and cover the full pressure range from 1 to 170 bars (2500 psi). The first two experiments have relatively fine and relatively coarse LWR debris operating, at dry out, in the laminar and the turbulent vapor-flow regiems, and will be performed in FY 84. The third experiment with a vertically stratified bed and variable inlet flow will be performed in FY 85. This program is part of the integrated severe fuel damage research program of the Fuel Systems Research Branch.

BUDGET ($K):	FY 84	FY 85	FY 86
	500	0	0

Project 8 (FIN NO. A1342)

TITLE: MELT PROGRESSION ANALYSIS (MELPROG)

OBJECTIVES: To develop analytical models and a computer code (MELPROG) to simulate LWR in-vessel melt progression from the onset severe core damage to core debris interaction with the structures in the vessel lower plenum and to reactor vessel failure. As part of the project, a subcode (VICTORIA) will be developed to model the fission product release from core debris; VICTORIA will be linked with TRAP-MELT as a module to be used in MELPROG to handle fission product release and transport in the vessel. MELPROG will use the two-dimensional thermal-hydraulic models under development at LANL and will also use some subroutines adopted from SCDAP for degraded core behavior. MELPROG will calculate the timing and release rates of the fission products and core debris from a failed vessel to the containment.

BUDGET ($K):	FY 84	FY 85	FY 86
	590	650	800

Project 9 (FIN NO. A2016)

TITLE: TRANSIENT FUEL RESEARCH & FP RELEASE

OBJECTIVES: To develop physically realistic models which describe fission product release from LWR fuels during thermal transients. This project centers on the development of a mechanistic fission product release code (and a fast running version) for the early phase of a severe accident. Work plans include the development of models to describe the effect of enhanced fission product release by fuel oxidation and the incorporation of the semi-volatile fission products such as Te, Ba, Sr, etc., into the code.

BUDGET ($K):	FY 84	FY 85	FY 86
	314	300	250

Project 10 (FIN NO. A2220)

TITLE: TMI FUEL EXAMINATION

OBJECTIVES: Conduct macro, microstructural, and chemical analyses of selected specimens removed from the core of the TMI-2 reactor pressure vessel to determine thermal and chemical changes that occurred.

With respect to the utility of TMI-2 examinations, there are a number of phenomena and for which TMI-2 data could potentially be of significant benefit. This includes the unknown degree of aerosolization of control rod material, and the chemical behavior of tellurium in the presence of metallic zircaloy. The principal TMI-2 core examinations at ANL will be carried out under DOE auspices at several facilities. The larger portions of the core will be first sent to EG&G hot-cells for dissemination of smaller units to satellite facilities for in-depth examinations.

Selected TMI-2 core debris specimens will be analyzed using optical and electron microscopy, X-ray, and electron microprobe chemical analyses, auger and scanning electron microscopy, to characterize the changes that occurred in the TMI-2 accident.

Following the beginning of TMI-2 defueling operations ANL will continue analyses on components including fuel and control material pieces, fuel assembly structural pieces, and core debris from primary system locations.

BUDGET ($K):	FY 84	FY 85	FY 86
	190	270	500

Project 11 (FIN NO. A2232)

TITLE: POSTTEST FUEL EXAMINATION

OBJECTIVES: To conduct post-test examinations (PIE) of the commercially irradiated fuel rods specimens subjected to high temperature fission product release testing at ORNL. To provide information on fission product release mechanisms for use in developing detailed models of fission product release from fuel under severe core damage and core melt accidents.

To accomplish the objective of determining the fission product release mechanisms from the ORNL tests, the investigator will look for diffusion along grain boundaries, interlinkage of voids, grain boundary sweeping, microcracking, etc., and compare to pretest fuel morphology. Post test fuel examination will also include the determination of the spatial distribution of the oxygen concentrations within the fuel and the locations of fission products remaining within the fuel pellet structure.

BUDGET ($K):	FY 84	FY 85	FY 86
	150	150	150

Project 12 (FIN NO. A6050)

TITLE: FUEL BEHAVIOR MODEL DEVELOPMENT

OBJECTIVES: To improve analytical models used in the FRAP-T6 code and in the MATPRO code. FRAP-T6 is used to simulate fuel rod behavior including clad ballooning, rupture, and fission product release during reactor transients and design-basis accidents. MATPRO is used to provide material properties input to FRAP-T6 and also to SCDAP which is for severe accident simulations. As part of the project, INEL will maintain and update both FRAP-T6 and MATPRO as new data become available.

BUDGET ($K):	FY 84	FY 85	FY 86
	380	200	0

Project 13 (FIN No. A6044 and A6057)

TITLE: PBF ENGINEERING & PLANT OPERATIONS

OBJECTIVES: Provide operating crew for and engineering support of the Power Burst Facility.

BUDGET ($K):	FY 84	FY 85	FY 86
	8000	5100	0

Project 14 (FIN NO. A6305)

TITLE: PBF SEVERE FUEL DAMAGE EXPERIMENTAL PROGRAM

OBJECTIVES: To provide a data base and analytical models on fuel behavior under the core-uncovery and reflood quench conditions of severe LWR accidents by performing integral multi-rod fuel-bundle tests in the PBF test reactor, by analysis of the test results, and by model development. Fission-product release and transport, hydrogen generation, and temperature distributions during the test transients are to be measured, and post-test characterization of the test fuel debris, including fuel relocation is to be made by neutron radiography and tomography and by post-irradiation examination (PIE). This program is part of the integrated severe fuel damage research program of the Fuel Systems Research Branch.

The program includes four Severe-Fuel-Damage (SFD) tests in the PBF test reactor, analysis of results, and Post-Irradiation Examination (PIE) of the damaged 32-rod fuel bundles. Test SFD 1-1, performed in 1983, used preconditioned (at power) fresh fuel as did the scoping test. Test SFD 1-3, to be performed in 1984, will use preconditioned pre-irradiated fuel as will the similar early 1985 test SFD 1-4 that will include As-In-Cd control rods. Irradiated-fuel tests 1-3 and 1-4 will have substantially improved fission-product diagnostics.

BUDGET ($K):	FY 84	FY 85	FY 86
	4200	2600	2000

Project 15 (FIN NO. A6321)

TITLE: IN-PILE FISSION PRODUCT STUDIES AT PBF

OBJECTIVES: To measure fission product behavior during the PBF Severe Fuel Damage tests (see Project 14). To design systems to obtain fission product release data. To collect, reduce, analyze, and report fission product data.

BUDGET ($K):	FY 84	FY 85	FY 86
	3200	2500	1000

Project 16 (FIN NO. A6352)

TITLE: RESIDENT SCIENTIST AT KFK KARLSRUHE, FRG

OBJECTIVES: To facilitate the exchange of nuclear safety-related information between the U.S. and Germany.

BUDGET ($K):	FY 84	FY 85	FY 86
	175	140	200

Project 17 (FIN NO. A6360)

TITLE: SEVERE FUEL DAMAGE MODEL DEVELOPMENT (SCDAP)

OJBECTIVES: To develop analytical models and a computer code (SCDAP) to simulate severe fuel damage and degraded core accidents in LWR's. SCDAP models fuel/clad melting and relocation, blockage formation, fission product release, and hydrogen generation. It has been used to simulate the TMI-2 accident and PBF severe fuel damage experiments. As part of the project, SCDAP will be linked with TRAP-MELT and RELAP5/MOD2 to provide system-wide modeling capabilities to calculate the timing and release rates of the fission products and hydrogen from a degraded core through a pathway to the containment during severe accidents in which the reactor vessel .remains intact. Close cooperation will be maintained between SCDAP developers and MELPROG developers.

BUDGET ($K):	FY 84	FY 85	FY 86
	712	700	800

Project 18 (FIN NO. A6829)

TITLE: MODIFICATIONS TO FISSION PRODUCT MEASUREMENT SYSTEMS FOR SFD SERIES-I TESTS

OBJECTIVES: Upgrade fission product measurement systems to obtain fission product release data during planned SFD Series 1-3 and 1-4 tests with pre-irradiated fuel. Obtain chemical form and aerosol data for fission products released in these tests. Collect, analyze, and report these fission product release and transport data.

Determine isotopic and chemical composition of aerosol samples collected on the filters of the gas sample bombs. Collect and analyze fission product data obtained from Ge and NaI gamma-ray spectrometers located on the steam line near the IPT. Determine chemical forms of fission products by analyzing samples with laser Raman spectrometer and fluorescence spectrometer. Improve analytical methods for identification of fission products for: (a) chemical forms in solution, (b) Beta-particle emitting radionuclides, and (c) hydrogen iodide and organic iodines. Incorporate aerosol measurement system on steam line near in-pile-tube and measure the time-dependent aerosol particle concentration and the particle size fractions. Estimate mean aerodynamic diameter size of aerosol particles.

BUDGET ($K):	FY 84	FY 85	FY 86
	1340	345	0

Project 19 (FIN NO. B0127)

TITLE: FP RELEASE FROM LWR FUEL

OBJECTIVES: Investigate experimentally the magnitude and physiochemical form of fission products and aerosols released from commercial irradiated LWR fuel under the elevated temperature and environmental conditions characteristic of severe fuel damage and core melt accidents.

This project consists of two tasks. Task A is a separate effect study which measures the quantitative release of fission products from a commerically irradiated fuel rod segment due to parametric variation. Task B is an integral study of the silver and boron control rod behavior in the presence of adjacent fuel rods. The simulant fuel rod bundle used here simulates a portion of the reactor core where a vertical as well as a radial temperature gradient exists.

Task A will focus on a more detailed study of the effects of time, temperature, steam flow, fuel burnup and heating rate on release. Task B will focus on the aerosolization of Ag and Cd after the control rod burst and the decomposition of CsI by the borate compounds.

BUDGET ($K):	FY 84	FY 85	FY 86
	1055	1460	1200

Project 20 (FIN NO. B0453)

TITLE: POST ACCIDENT FISSION PRODUCT CHEMISTRY

OBJECTIVES: Determine the chemical species of various fission products present in aqueous reactor solutions and their liquid/vapor distribution under representative reactor accident conditions.

The main objective of this project is to determine the gaseous iodine concentration in the containment atmosphere given that most of the iodine released went into solution in the containment sump. The re-evolution of the dissolved iodine at various conditions (particularly in the presence of radiation and silver) is the main concern of this project. Investigation will focus on the modelling of the iodine partition coefficient and the study on the organic iodine formation.

BUDGET ($K):	FY 84	FY 85	FY 86
	300	300	300

Project 21 (FIN NO. B0488)

TITLE: TRAP-MELT VERIFICATION PROGRAM

OBJECTIVES: To conduct small scale tests on fission product and aerosol transport to develop a data base for the early assessment of the TRAP-MELT code. Small scale experiments concentrating on the validation of phenomena of fission product and aerosol transport included into primary system transport and deposition code (TRAP-MELT). This includes an examination of aerosol transport through vertical pipe, using plasma torch to generate aerosols, for a spectrum of conditions, an examination of resuspension phenomena during fission product and aerosols transport, and an examination of upper plenum simulation on the effect of transport and deposition.

To participate in the review, planning, and implementation, and analyzing of the results of the severe fuel damage (SFD) series I tests; to participate in the review, planning, implementation, and analyzing of the results of the large scale fission product and aerosol transport tests in the MARVIKEN facility.

BUDGET ($K):	FY 84	FY 85	FY 86
	407	325	500

Project 22 (FIN NO. B0815)

TITLE: FP DEPOSITION OF AEROSOLS

OBJECTIVES: To examine fission product movement and interactions via aerosols by determination of the deposition capacity of aerosols, the rate of deposition, and the deposition of key fission product vapors onto aerosols relative to other structures.

Provide experimental data on the sorptive capacity of prototypic aerosol materials for selected important fission products and to determine the sorption rates and controlling deposition mechanisms (condensation chemisorption, etc.).

BUDGET ($K):	FY 84	FY 85	FY 86
	285	325	300

Project 23 (FIN NO. B0827)

TITLE: IN-PILE FISSION PRODUCT AND AEROSOLS TESTS: TECHNICAL SUPPORT

OBJECTIVES: Participate in the review process of all technical reports and other technical documents in relation ot PBF Series 1-3, and 1-4 tests and provide comments to support evaluation of and planning for PBF Phase I experiments.

Continue the review process of the results obtained during PBF scoping test and 1-1 test. Participate, in the collaboration with EG&G, in report(s) preparation.

Conduct, if necessary and warranted and as directed by NRC, independent analyses of selected samples obtained during the scoping and 1-1 tests to independently confirm findings and/or interpretation provided by others.

Participate in the review process of all technical reports and other technical documents in relation to the planned separate effect fission product and aerosols test at ACRR (SNL) and to the planned large integral tests at NRU (PNL/AECL) to support the evaluation of the technical approach and experiments planning. Integrate the on-going ORNL fission product release from fuel experiments (HI series) and small scale core-melt experiments into planned ACRR/NRU in-pile test program and test matrix.

BUDGET ($K):	FY 84	FY 85	FY 86
	143	150	150

Project 24 (FIN NO. B0831)

TITLE: MARVIKEN FISSION PRODUCT AND AEROSOL TESTS: TECHNICAL SUPPORT

OBJECTIVES: To provide technical support for the fission product and aerosol large tests of the MARVIKEN facility (Sweden) especially in the area of fission product and aerosols generation (fissium and corium) including efficiency determination, measurements and selected design aspects (upper plenum simulator design, PWR/BWR corium recipe, etc.).

Participate in the review process of all applicable MARVIKEN (as specified by NRC) technical documents and provide comments as a part of on-going review process to support and evaluate the experimental results.

As directed by NRC provide analyses and conduct small-scale experiments which might be needed to support the evaluation of the MARVIKEN experiments.

[illegible] directed by NRC participate and provide a technical support in the inter-laboratory review process (SNL, EG&G, BCL) of the MARVIKEN experiments.

As directed by NRC provide support for the U.S. scientist participation in the MARVIKEN experiments.

BUDGET ($K):	FY 84	FY 85	FY 86
	166	175	0

Project 25 (FIN NO. B0121)

TITLE: AEROSOL RELEASE & TRANSPORT

OBJECTIVES: Perform experiments to investigate phenomena affecting generation and subsequenct behavior of aerosols over a range of severe accident conditions. Develop a data base for validating prediction models on generic source terms for molten fuel within containment.

Two series of experiments are planned at ORNL to obtain quantitative information on the phenomena which affect either the characteristics or behavior of LWR aerosols.

The first series of experiments will be conducted in the Nuclear Safety Pilot Plant (volume of 38.3 m^3 vessel). The independent variables of interest include: relative humidity and all possible combinations (single and multi-component) of U_3O_8, Fe_2O_3, and limestone concrete aerosols.

The second series of experiments and will be conducted in the Aerosol-Moisture Interaction Vessel (volume of 0.56 m^3). The independent variables of interest include: relative humidity, aerosol components (U_3O_8, Fe_2O_3, limestone concrete, Ag + Cd), aerosol mass ratios, and aerosol concentrations.

The questions of interest include: at what level of relative humidity do the various aerosols or combinations of those aerosols change shape factors to look more like spheres and nt like chain agglomerates, what are the functional relationship among aerosol fallout behavior and the set of independent variables of interest, and what quantitative change in measured responses (mass concentration, aerodynamic size distribution, fallout as a function of time, microscopy) can be shown to be due to a given quantitative change in initial conditions (i.e., independent variables)?

BUDGET ($K):	FY 84	FY 85	FY 86
	1300	1300	1000

Project 26 (FIN NO. A1339)

TITLE: MELCOR

OBJECTIVES: Provide the replacement risk code (MELCOR) for the MARCH, MATADOR, and CRAC codes which: has a structure readily amenable to incorporation of new models based on the ongoing experiment research program; and permits the quantitative analysis of both "best estimate" severe accident consequences and the associated uncertainties.

BUDGET ($K):	FY 84	FY 85	FY 86
	1670	1050	500

Project 27 (FIN NO. A1019)

TITLE: MOLTEN FUEL-CONCRETE INTERACTIONS

OBJECTIVES: To characterize the chemical and physical phenomena associated with the interactions between molten LWR core materials and concrete likely to be encountered during hypothetical fuel melt accidents and to develop models for predicting these phenomena.

BUDGET ($K):	FY 84	FY 85	FY 86
	280	320	350

Project 28 (FIN NO. A1030)

TITLE: MOLTEN CORE-COOLANT INTERACTIONS

OBJECTIVES: To develop an understanding of the nature of core melt-coolant interactions during light water reactor hypothetical accidents;

To develop a data base for assessing the threat of core melt-coolant interactions to the integrity of the reactor vessel and containment;

To provide analysis and models for the reactor meltdown progression and the phenomena associated with core melt-coolant interactions; and

To perform experiments to investigate phenomena affecting core melt-coolant interactions.

These series of experiments are planned to obtain quantitative information on the phenomena which affect one or more of the three phases of molten-core coolant interactions: coarse mixing, triggering, or propagation. In order to obtain the most information per dollar spent, each series will be based on a statistically designed experiment, e.g., a fractional factorial design.

For all three series of experiments the questions of interest include: Is there a limit to the amount of melt mass which can participate in coarse mixing (series 2)? What is the functional relationships of melt mass and conversion ratio (thermal to kinetic) and what quantiative change in measured response (e.g., pressure, hydrogen production, debris particle size) can be shown to be due to a given quantitative change in initial conditions (i.e., independent variables)?

BUDGET ($K):	FY 84	FY 85	FY 86
	700	1115	1500

Project 29 (FIN NO. A1198)

TITLE: CONTAINMENT ANALYSIS

OBJECTIVES: To develop a computer program CONTAIN that models a power-reactor containment system and predicts the response of the system to accident-imposed conditions.

BUDGET ($K):

FY 84	FY 85	FY 86
1100	950	1100

Project 30 (FIN NO. B6747)

TITLE: RADIONUCLIDE RELEASE UNDER SPECIFIC LWR ACCIDENT CONDITIONS (BMI-2104)

OBJECTIVES: Provide analytical models and computer codes for application to the analysis of release and transport of fission products in LWR plants under severe accident conditions. Improvement to the primary system transport and deposition code (TRAP-MELT).

BUDGET ($K):	FY 84	FY 85	FY 86
	1036	0	0

Project 31 (FIN NO. B2444)

TITLE: ESF SYSTEMS EFFECTIVENESS EVALUATION

OBJECTIVES: Obtain and develop information that will aid in providing best estimates of the chemical and physical properties of environments associated with severe accident phenomena to predict performance of selected ESF systems (in these environments) emphasizing fission product depletion mechanisms. This includes filtration, sprays, suppression pools and ice condensers.

To develop mechanistic code for suppression pool effectiveness in removal fission product and aerosols (SPARC). To develop mechanistic code for the ice condenser effectiveness in removing fission product and aerosols (ICEDF). Validate SPARC using EPRI suppression pool experiments. Validate ICEDF by performing large integral tests or small separate effects experiments (planned).

BUDGET ($K):	FY 84	FY 85	FY 86
	675	875	800

Project 32 (FIN No. A1383)

QUEST I
(described elsewhere)

QUEST II (Planned)
(To be provided later)

Project 33*

TITLE: SUPPRESSION POOL SCRUBBING EXPERIMENTS (EPRI)

(Described elsewhere)

BUDGET: Not Known

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* Not an NRC experimental program, but it is expected to provide a valuable data base to validate the SPARC (NRC) and SUPRA (EPRI) suppression pool scrubbing codes.

Project 34**

TITLE: LWR AEROSOLS CONTAINMENT EXPERIMENTS (EPRI LACE PROGRAM)

OBJECTIVES: To provide a data base for validating containment aerosol and thermal hydraulic computer codes, and to investigate (in large scale) inherent radioactive aerosol retention behavior for postulated high consequence accident situations. This will include containment bypass sequences, early containment leakage or failure to isolate, and delayed containment failure (if aerosol resuspension proves to be significant, or core damage was similarly delayed)

BUDGET: Not known; foreign participation is being negotiated.

**Not an NRC experimental program, but it is expected to provide valuable data base to validate some of the containment aerosols transport codes, including the investigation of containment bypass release (V Sequences).

Project 35

TITLE: AEROSOL TRANSPORT TESTS (MARVIKEN PROGRAM)

OBJECTIVES: To provide information on the deposition rates in primary system components of selected fission product, control rod, and structural materials at temperatures and scale approaching those found in reactor systems under accident conditions.

The MARVIKEN experiments provide for a large, integral, high temperature system with a complex mixture of fissium and corium being injected into the reactor vessel and then transported through various pipes and components.

The results are intended to show the effects of multiple components along the flow path, fissium/corium interactions, fissium interactions with surfaces and flow irregularities in large vessels. The effects of scale and multiple flow components are not being studied elsewhere and, hence, the MARVIKEN experiments serve as a unique source for this type of information.

BUDGET ($K):	FY 84	FY 85	FY 86
	460	400	400

(NRC share 12.4% of the total budget.)

ENCLOSURE 4

PROJECT AND BUDGET PROPOSAL
FOR NRC SOURCE TERM RELATED
RESEARCH PROJECTS (FORMS 189)

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PROJECT AND BUDGET PROPOSAL
FOR NRC SOURCE TERM RELATED
RESEARCH PROJECTS (FORMS 189)

March 1, 1984

FY 1984 PROGRAM BRIEF

PROGRAM: AE

TITLE: MODIFICATIONS TO FISSION PRODUCT MEASUREMENT SYSTEMS FOR SFD SERIES 1 TESTS

FIN NO: A6829
CONTRACTOR: EG&G
SITE: INEL
STATE: IDAHO

NRC PROGRAM MANAGER: P. REED

EG&G PROGRAM MANAGERS: D. B. VAN LEUVEN, P. MACDONALD

PRINCIPAL INVESTIGATORS: P. MACDONALD, A. APPELHANS, R. HOBBINS, D. OSETEK

OBJECTIVES:

(1) TO UPGRADE FISSION PRODUCT MEASUREMENT SYSTEMS TO OBTAIN FISSION PRODUCT RELEASE DATA DURING PLANNED SFD SERIES 1-3 AND 1-4 TESTS.

(2) TO OBTAIN CHEMICAL FORM AND AEROSOL DATA FOR FISSION PRODUCTS RELEASED IN SERIES 1 SEVERE FUEL DAMAGE TESTS.

(3) TO COLLECT, ANALYZE, AND REPORT FISSION PRODUCT RELEASE AND TRANSPORT DATA DURING SERIES 1 SEVERE FUEL DAMAGE TESTS.

(4) TO INVESTIGATE FISSION PRODUCT TRANSPORT AND BEHAVIOR UNDER SEVERE ACCIDENT CONDITIONS.

BUDGET ACTIVITY: 60190201

FY 1984 OBLIGATION: $1010K

TASK A: MODIFICATIONS FOR SFD SERIES 1 TESTS.

PERFORM THE FOLLOWING UPGRADES FOR THE SFD SERIES TESTS:

1. INCORPORATE FILTERS ON SIX ADDITIONAL GAS SAMPLE BOMBS TO COLLECT PARTICULATES PRESENT IN THE GAS PHASE DURING THE TESTS.

FY 84 COSTS: $170K

   a. DETERMINE THE ISOTOPIC AND CHEMICAL COMPOSITION OF THE AEROSOLS AND THE CHEMICAL FORM OF THE FISSION PRODUCTS ASSOCIATED WITH THE AEROSOLS.

2. ADD Ge AND NaI GAMMA-RAY SPECTROMETERS TO THE STEAM LINE NEAR THE IPT

FY 84 COSTS: $345K

   a. $15K IS AUTHORIZED TO PREPARE TECHNICAL ASSESSMENT AND SCHEDULE ESTIMATES FOR INSTALLATION OF THE GAMMA-RAY DETECTORS TO VIEW THE STEAM LINE JUST ABOVE THE IN-PILE TUBE HEAD. THE ASSESSMENT SHOULD INCLUDE MODIFICATIONS TO PORTIONS OF THE WORKING PLATFORM, INSULATING JACKET, COLLIMATOR AND STEAM LINE SHIELDING AND DESIGNS FOR AN INCLINED COLLIMATOR TUBE, SHIELD BOX, STRUCTURAL SUPPORT SYSTEM, AND PLANT SUPPORT SYSTEMS (NITROGEN, AIR, ELECTRICAL, CONTROL, ETC.) PREPARE A REPORT ON THE DESIGNS AND MODIFICATIONS AND PROVIDE A FIRM SCHEDULE TO NRC.

   b. IF THE ASSESSMENT AND SCHEDULE ESTIMATE CONCLUDE THAT THE DETECTORS CANNOT BE PLACED ABOVE THE IN-PILE TUBE, THE DETECTORS SHOULD BE PLACED AS CLOSE AS POSSIBLE TO THE IN-PILE TUBE HEAD AND UPSTREAM OF THE MAIN FLOOR STEAM SAMPLE BOMBS.

3. IMPROVE FISSION PRODUCT COMPOUND IDENTIFICATION BY RAMAN AND FLUORESCENCE SPECTROSCOPY AND DETERMINE LIBRARY (REFERENCE) SPECTRA STANDARDS OF EXPECTED FISSION PRODUCT COMPOUNDS FOR THE LASER RAMAN SPECTROMETER AND FLUORESCENCE SPECTROMETER

FY 84 COSTS: $205K

a. MODIFY THE EXISTING MOLECULAR OPTICAL LASER EXAMINER TO INCLUDE A QUARTZ FLUORESCENCE SPECTROMETER.

b. DEVELOP A SET OF FISSION PRODUCT COMPOUND STANDARD REFERENCE SPECTRA FOR COMPARISON TO SERIES 1 TESTS SAMPLES.

c. DEVELOP CATALOG OF REF[illegible]CTRA.

d. DEVELOP METHODS FO[illegible]YSES OF TEST SAMPLES.

e. PREPARE REPORT PRESENTING CATALOG OF SPECTRA AND PROCEDURES DEVELOPED FOR TEST SAMPLE ANALYSES.

4. IMPROVE ANALYTICAL METHODS FOR IDENTIFICATION OF FISSION PRODUCTS FOR: a) CHEMICAL FORMS IN SOLUTION, b) BETA EMITTING RADIONUCLIDES, AND c) HYDIRDE COMPOUNDS, HYDROGEN IODIDE AND ORGANIC IODINES.

FY 84 COSTS: $220K

5. ADD A LIQUID COLLECTION SYSTEM TO THE LIQUID SAMPLE LINE AND DETERMINE FISSION PRODUCT CONCENTRATIONS IN THE GAS AND LIQUID PHASES OVER A PERIOD OF TIME FOLLOWING THE TESTS TO SIMULATE POST-ACCIDENT BEHAVIOR OF FISSION PRODUCTS.

FY 84 COSTS: $70K

a. THE WATER AND GAS SAMPLES SHOULD BE ANALYZED FOR FISSION PRODUCTS BEFORE AND AFTER THE ADDITION OF SODIUM BORATE TO THE WATER SAMPLES.

May 14, 1984

REVISED FY 1984 PROGRAM BRIEF

PROGRAM: AE

TITLE: TRAC/MELPROG INTEGRATION

FIN NO: A7303
CONTRACTOR: LANL
SITE: LOS ALAMOS
STATE: NEW MEXICO

NRC TECHNICAL MONITOR: J. T. HAN

PRINCIPAL INVESTIGATOR: R. HENNINGER

OBJECTIVE: TO PERFORM DETAILED THERMAL-HYDRAULIC CALCULATIONS IN THE UPPER PLENUM REGION TO STUDY FISSION PRODUCT TRANSPORT AND DEPOSITION.

BUDGET ACTIVITY: 60190201

FY 1984 OBLIGATION: $208K

THIS ORDER: $120K
TOTAL $328K

FY 1984 SCOPE REVISED (10/01/83 - 09/30/84):

1. ISSUE A REPORT FOR A TRAC NATURAL-CONVECTION CALCULATION IN THE UPPER PLENUM REGION OF THE SURRY PWR VESSEL.

2. DEVELOP TWO-DIMENSIONAL DEGRADED-CORE THERMAL-HYDRAULIC CAPABILITIES FOR TRAC/MELPROG.

3. ISSUE A TRAC/MELPROG USER MANUAL INCUDING THE DESCRIPTION OF THE COUPLING BETWEEN THE MELPROG AND TRAC-PF1 CODES.

4. COOPERATE WITH SNL IN MELPROG DEVELOPMENT.

5. PROVIDE ON-CALL TECHNICAL ASSISTANCE TO NRC.

GENERAL RESEARCH HISTORY

BEGINNING TO 1974

- o ATOMIC ENERGY COMMISSION
- o DEVELOP REACTOR TECHNOLOGY
- o DATA BASE FOR SAFETY ISSUES
  - ECCS/THERMAL HYDRAULIC
  - METALLURGY
  - FUELS

1975 TO PRESENT

- o NUCLEAR REGULATORY COMMISSION
- o SAFETY RESEARCH
- o CONFIRMATORY
- o EXPLORATORY

HISTORY OF SEVERE ACCIDENTS

1957 WASH-740

- o A CONSEQUENCE ANALYSIS OF CORE MELT WITH DIFFERENT CONTAINMENT PERFORMANCE

1963 TID-14844

- o A TECHNICAL RATIONALE FOR REGULATORY TREATMENT OF SEVERE ACCIDENTS

1975 WASH-1400

- o A RISK ASSESSMENT OF LARGE POWER REACTORS
- o A PRIMER FOR ACCIDENT CAUSES
- o A MECHANISTIC ANALYSIS OF CORE MELT BEHAVIOR--PRODUCING A SOURCE TERM

1978 IMPROVED REACTOR SAFETY PROGRAM

- o RISK BASED IMPROVEMENTS
  - CORE CATCHERS
  - FILTERED VENT CONTAINMENT
  - ADD ON DHR SYSTEMS
  - UNDERGROUNDING
- o RELIED ON WASH-1400 MECHANISTIC ANALYSIS OF CORE MELT

HISTORY OF SEVERE ACCIDENTS (CONT.)

1980 REVIVAL OF INTEREST IN CORE MELT

- RECOGNITION THAT CORE MELT ACCIDENTS DOMINATE RISK
- TMI ACTION PLAN--DEGRADED CORE RULEMAKING
- LETTER FROM STRATTON, MALINAUSKAS, AND CAMPBELL
- NRC INTEROFFICE COMMITTEE ON DEGRADED CORE RULEMAKING
- PAPER BY LEVENSON AND RAHN

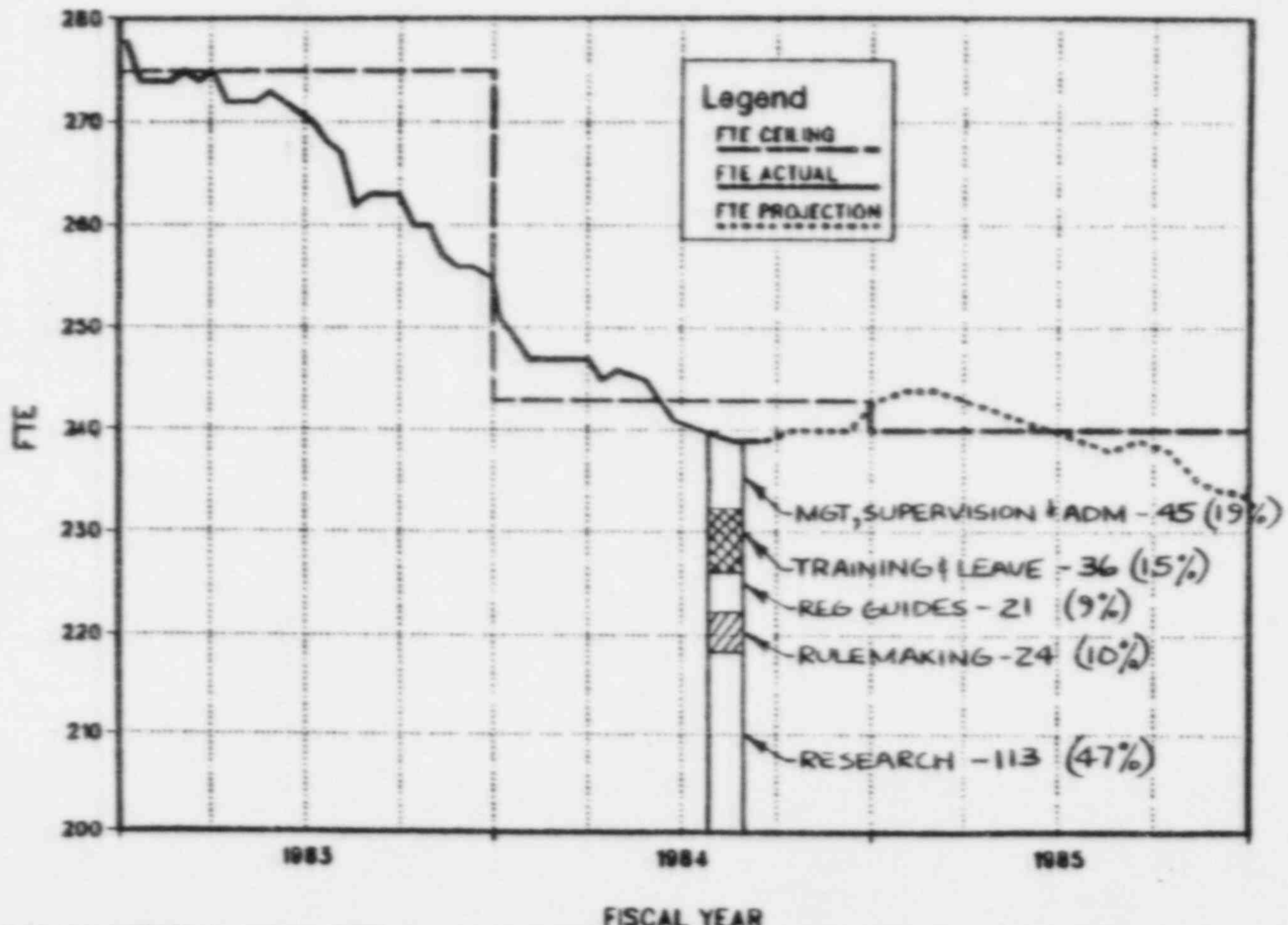
1981 INITIAL EVALUATION

- NUREG-0772
- WASH-1400 SOURCE TERMS APPEAR TO BE PESSIMISTIC
- SARP PLAN BASED ON PHASE I WITH WASH-1400 SOURCE TERMS, PHASE II TO CONFIRM OR REDUCE

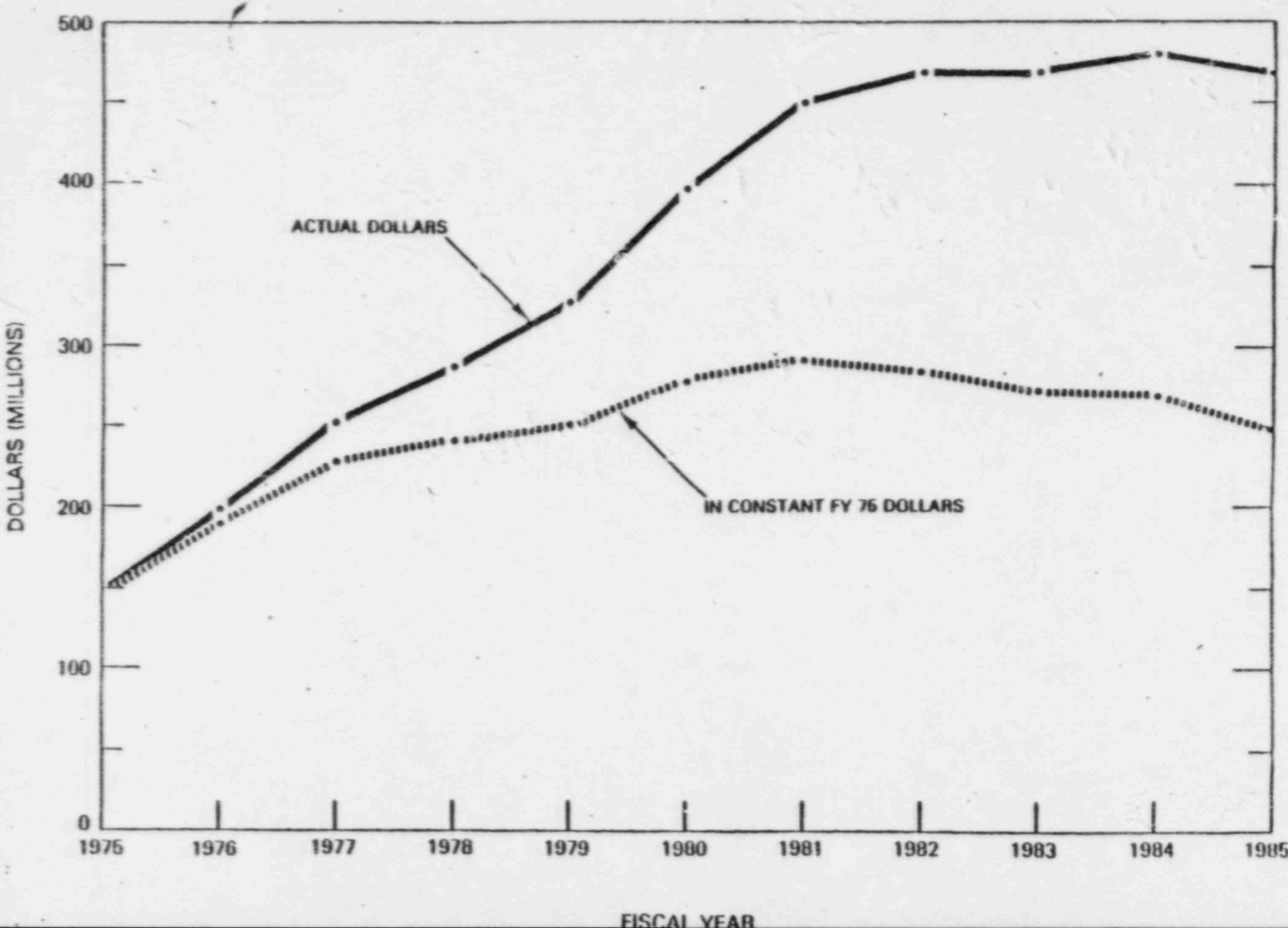
1982 DEMAND FOR NEW SOURCE TERM

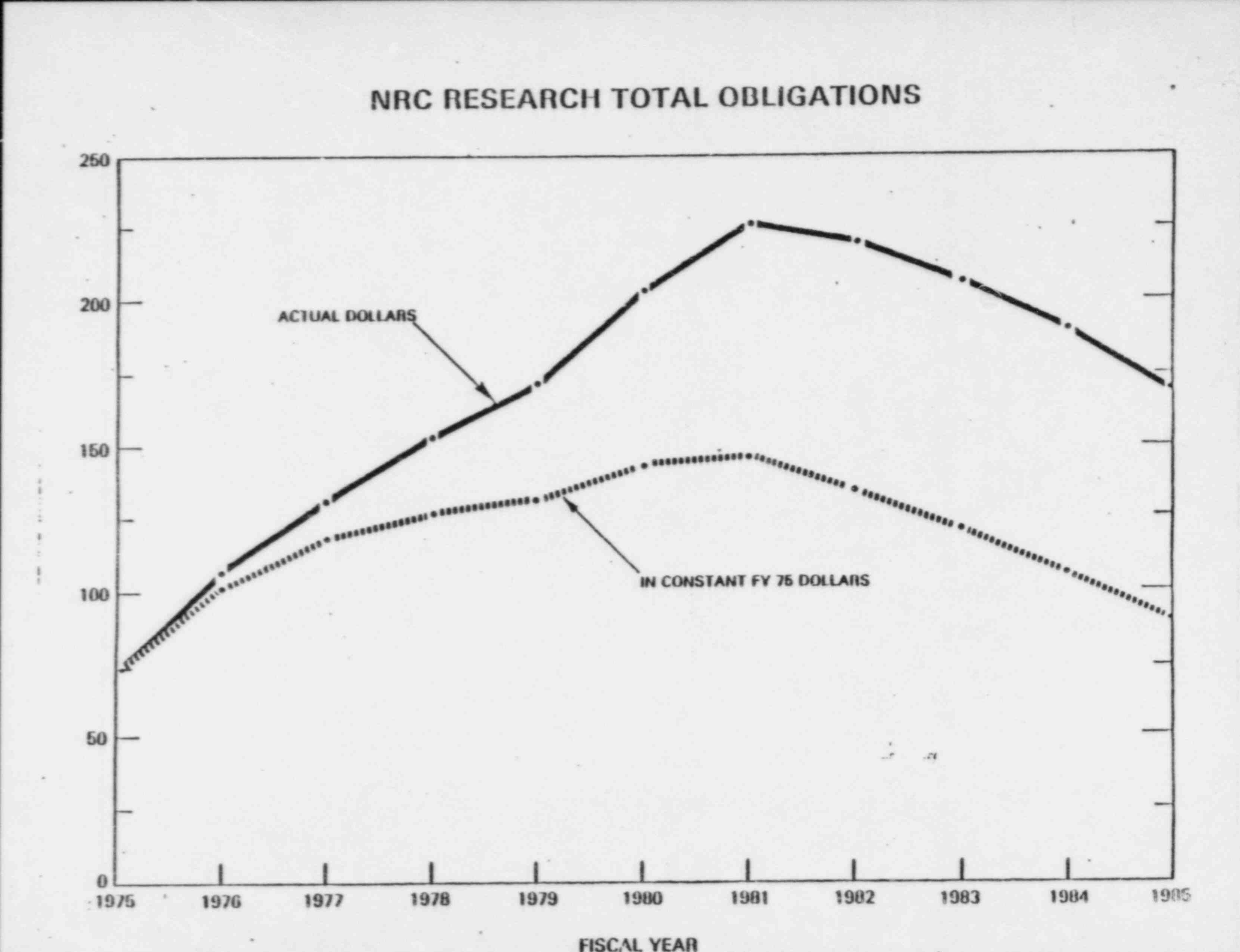
- SOURCE TERM PLAN
- SOURCE TERM PROGRAM OFFICE
- DELAY OF SARP PHASE I

RES FTE MANAGEMENT PLOT — 6/13/84
INSTANTANEOUS FTE USAGE RATE



NRC TOTAL OBLIGATIONS





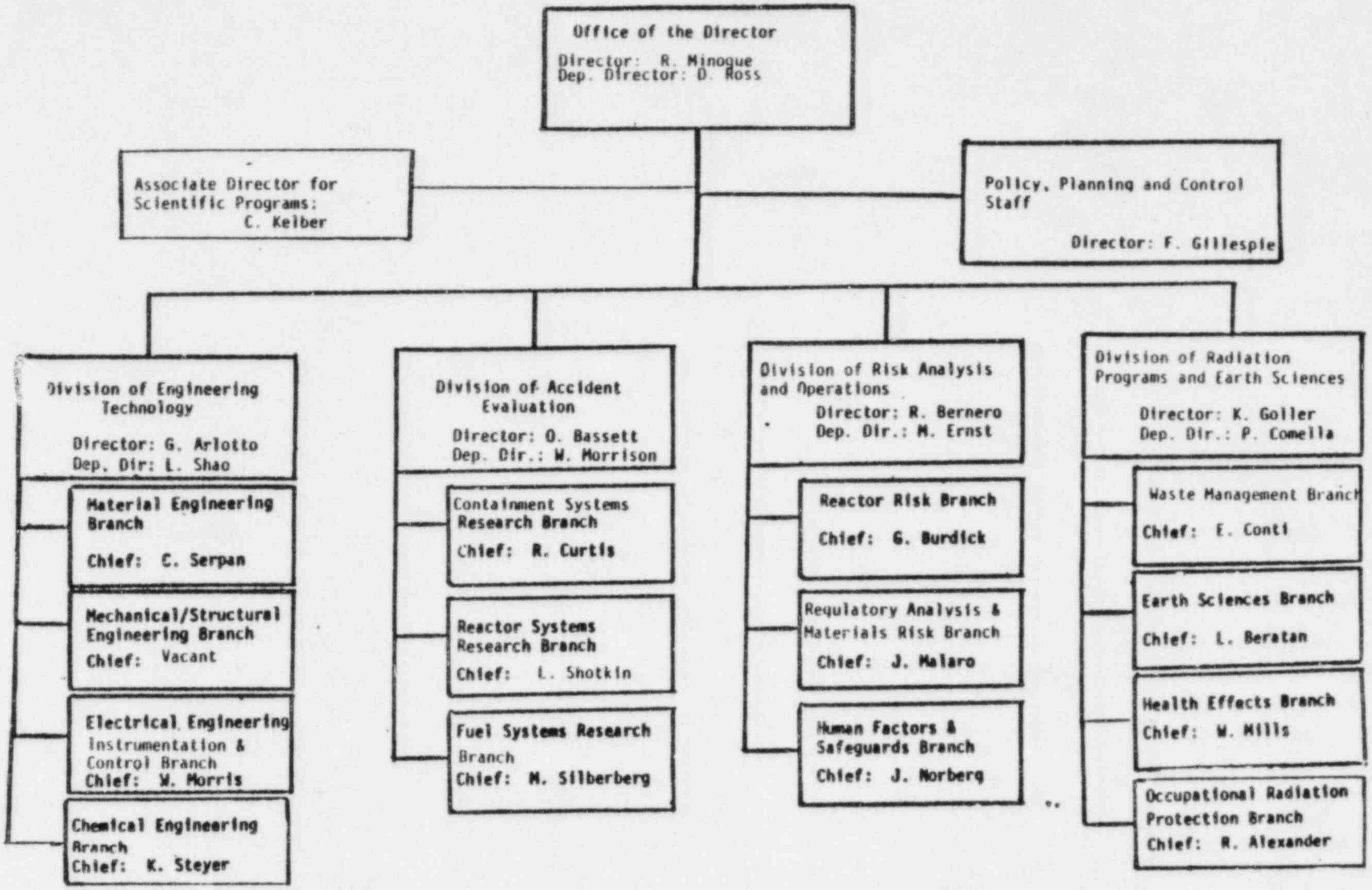
NRC RESEARCH TOTAL OBLIGATIONS
250
200
150
100
50
0
ACTUAL DOLLARS
IN CONSTANT FY 75 DOLLARS
1975
1976
1977
1978
1979
1980
1981
1982
1983
1984
1985
FISCAL YEAR

ORGANIZATION AND FUNCTION

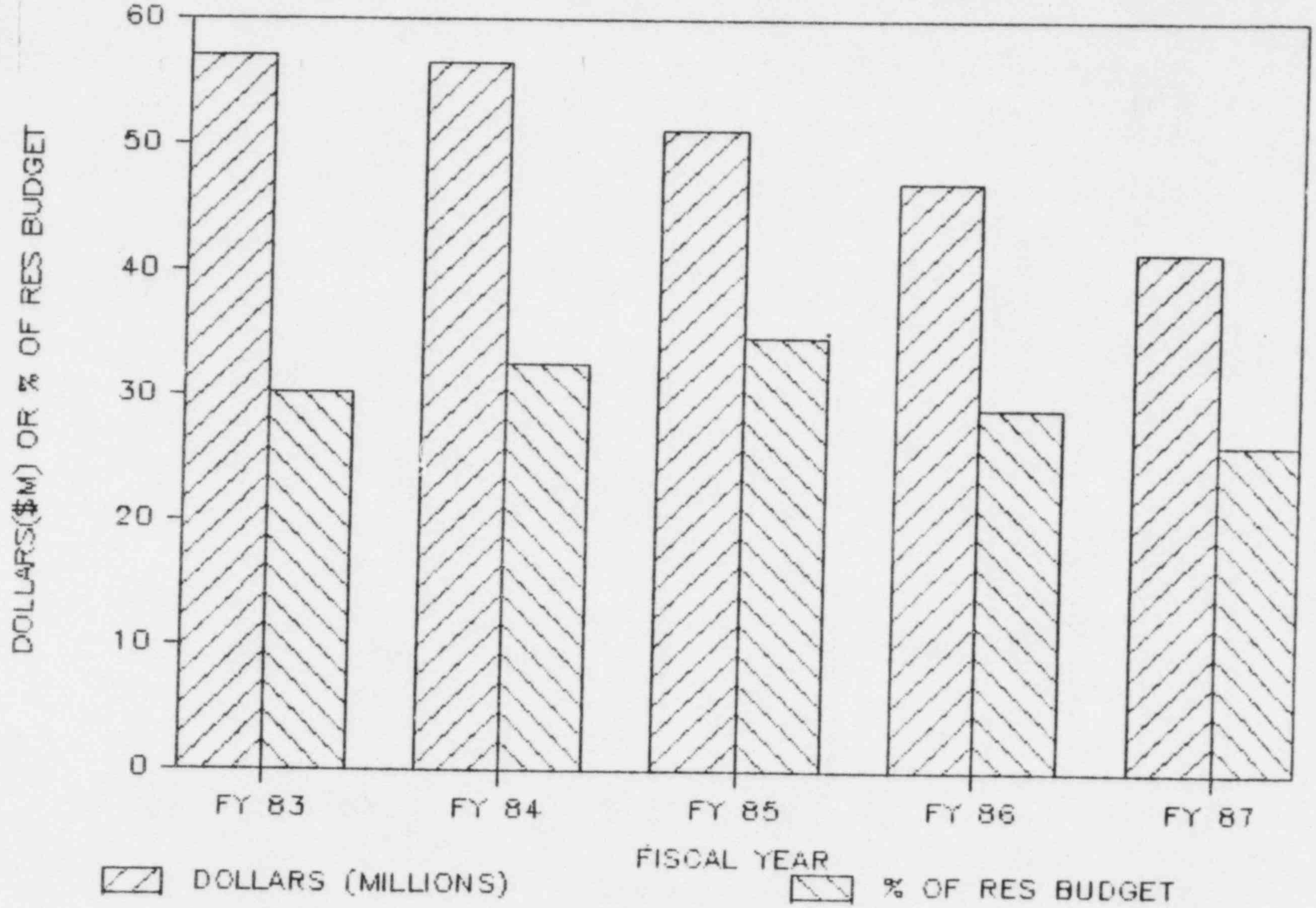
1. WHO IS DOING THE RESEARCH?

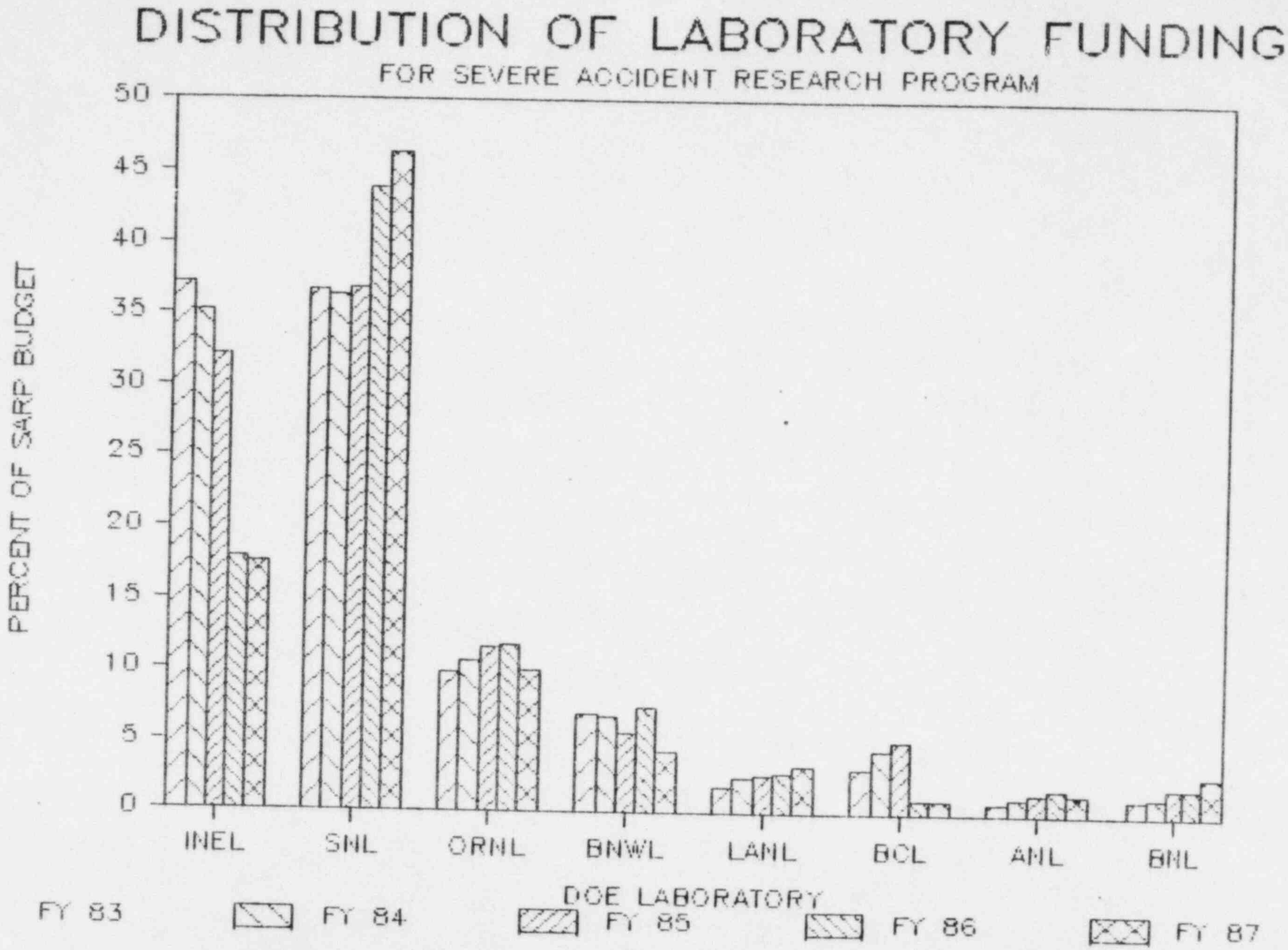
   - NRC OFFICE OF NUCLEAR REGULATORY RESEARCH
   - SANDIA NATIONAL LABORATORY (SNL)
   - IDAHO NATIONAL ENGINEERING LABORATORY (INEL)
   - OAK RIDGE NATIONAL LABORATORY (ORNL)
   - PACIFIC NORTHWEST LABORATORY (PNL)
   - LOS ALAMOS NATIONAL LABORATORY (LANL)
   - BATTELLE COLUMBUS LABORATORY (BCL)
   - ARGONNE NATIONAL LABORATORY (ANL)
   - BROOKHAVEN NATIONAL LABORATORY (BNL)
   - SELECTED CONTRACTORS

2. HOW ARE THEY SELECTED?


Office of the Director
Director: R. Minogue
Dep. Director: D. Ross
Associate Director for Scientific Programs: C. Keiber
Policy, Planning and Control Staff
Director: F. Gillespie
Division of Engineering Technology
Director: G. Arlotto
Dep. Dir: L. Shao
Material Engineering Branch
Chief: C. Serpan
Mechanical/Structural Engineering Branch
Chief: Vacant
Electrical Engineering Instrumentation & Control Branch
Chief: W. Morris
Chemical Engineering Branch
Chief: K. Steyer
Division of Accident Evaluation
Director: O. Bassett
Dep. Dir.: W. Morrison
Containment Systems Research Branch
Chief: R. Curtis
Reactor Systems Research Branch
Chief: L. Shotkin
Fuel Systems Research Branch
Chief: M. Silberberg
Division of Risk Analysis and Operations
Director: R. Bernero
Dep. Dir.: M. Ernst
Reactor Risk Branch
Chief: G. Burdick
Regulatory Analysis & Materials Risk Branch
Chief: J. Malaro
Human Factors & Safeguards Branch
Chief: J. Norberg
Division of Radiation Programs and Earth Sciences
Director: K. Goller
Dep. Dir.: P. Comella
Waste Management Branch
Chief: E. Conti
Earth Sciences Branch
Chief: L. Beratan
Health Effects Branch
Chief: W. Mills
Occupational Radiation Protection Branch
Chief: R. Alexander

SARP FUNDING BY FISCAL YEAR

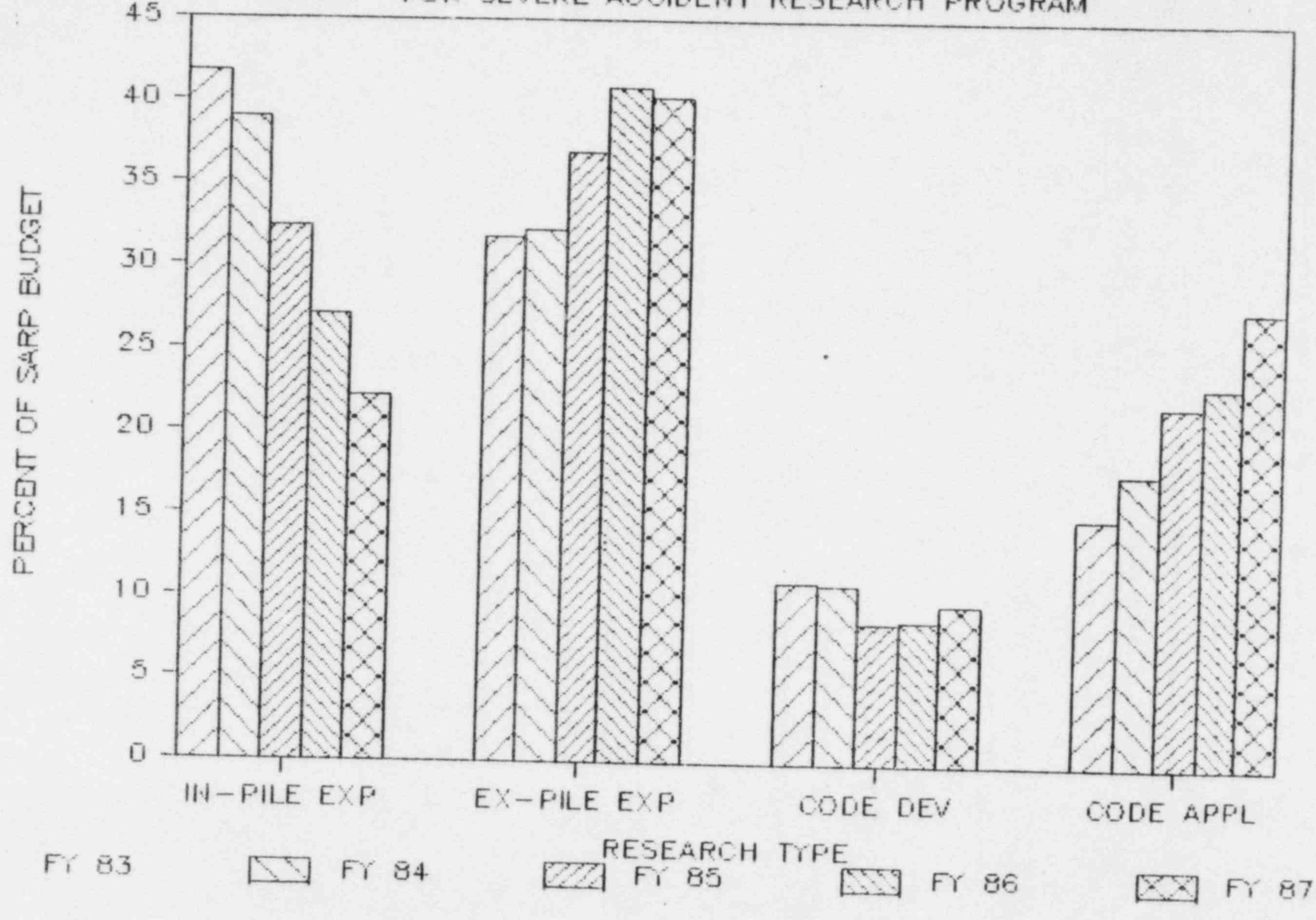




DISTRIBUTION OF LABORATORY FUNDING
FOR SEVERE ACCIDENT RESEARCH PROGRAM
PERCENT OF SARP BUDGET
50
45
40
35
30
25
20
15
10
5
0
INEL
SNL
ORNL
BNWL
LANL
BCL
ANL
BNL
DOE LABORATORY
FY 83
FY 84
FY 85
FY 86
FY 87

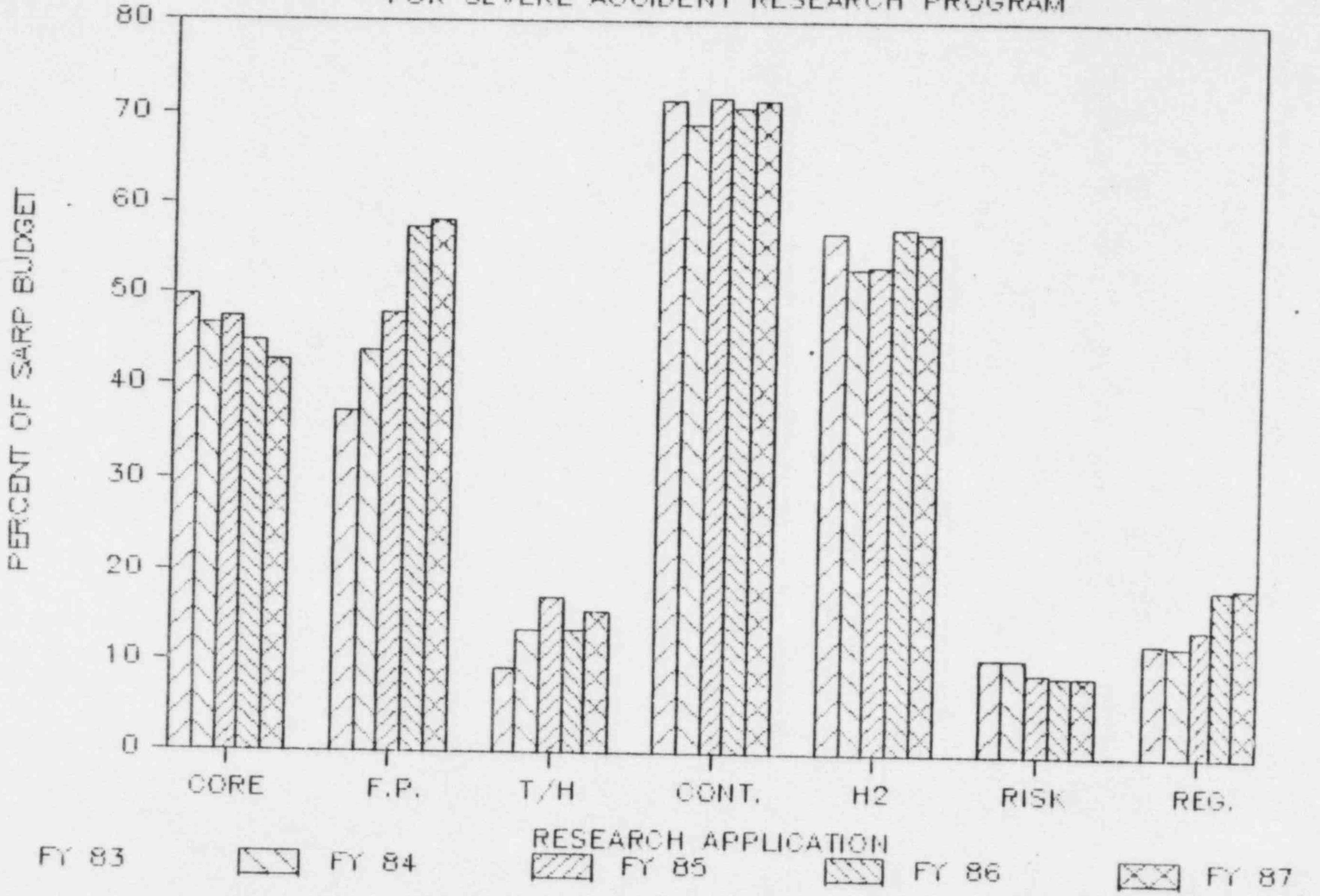
DISTRIBUTION OF FUNDS BY RESEARCH TYPE

FOR SEVERE ACCIDENT RESEARCH PROGRAM



DISTRIBUTION OF FUNDS BY APPLICATION

FOR SEVERE ACCIDENT RESEARCH PROGRAM



3. WHAT INTERNAL AND EXTERNAL ADVISORY AND TECHNICAL REVIEW COMMITTEES ARE IN PLACE?

   A. HOW ARE MEMBERS CHOSEN?

   B. WHO WRITES THEIR CHARGES?

      o RESEARCH REVIEW GROUPS

      o ACRS: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

         - ANNUAL REVIEW OF RESEARCH BUDGET AND PLANNING

         - SPECIAL SUBCOMMITTEE REVIEWS

      o SELECTED SUBJECTS PEER REVIEW

         - INDIVIDUAL

         - GROUP

         - STANDING COMMITTEE

4. WHO ARE THE PROGRAM MANAGERS AND WHAT PROTOCOLS GOVERN THEIR PARTICIPATION IN THE DESIGN OF THE RESEARCH PROGRAMS?

5. HOW DO YOU MANAGE QA/QC?

6. HOW ARE THE RESEARCH RESULTS DISSEMINATED BOTH INTERNALLY AND EXTERNALLY?

SOURCE TERM RESULTS

A. NUREG-0956

- o SOURCE TERM TECHNOLOGY
- o BMI-2104 REFERENCE PLANTS REPRESENT SPECTRUM OF DESIGN
- o CONTAINMENT GROUPS' NEW PERSPECTIVES
- o GENERAL ESTIMATES OF LWR RISK AND REGULATORY SIGNIFICANCE

B. SARRP REPORTS

- o BASED ON SOURCE TERM AND CONTAINMENT GROUPS
- o SPECIFIC ESTIMATES OF REFERENCE PLANT RISKS
- o COST-BENEFIT ANALYSES FOR REFERENCE PLANTS
- o EXTRAPOLATIONS

NUCLEAR REGULATORY RESEARCH

APPLICATION OF RESEARCH PROJECTS TO REGULATORY PROCESS

SEPTEMBER 15, 1982

PRC .ITLE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
Semiscale	Evaluate PWR systems behavior under simulated accident sequences leading to degraded core conditions.	10CFR50, App. A Criterion 34, Appendix K, and proposed rulemaking	. Provided data on: - Core uncovery and two-phase natural circulation. - Pumps-on versus pumps-off effects. - Small break tests with natural circulation effects. . Will investigate UHI plant safety features.
		NRC Action Plan (NUREG-0660 and -0737)	Pumps on/off resolution, two phase natural circulation including inert gas effects, small, intermediate, and large break data for non-UHI and UHI, loss of power transients, transients involving steam generator tube breaks, secondary side initiated transients, and general NRR support.
BWR FIST	Evaluate BWR systems behavior under a variety of simulated accident sequences	10CFR50, App. A Criterion 34 & 50, Appendix K, and planned rulemaking	Will provide data from simulated BWR small and intermediate break LOCA, ATWS and other transients to: - improve understanding of BWR transients - assess TRAC-BWR calculational capability - improve response of operators to BWR transients - evaluate system design improvements.
Separate Effects Exp. and Model Development	Measure BWR & PWR bundle thermal-hydraulic phenomena and develop predictive models. Unresolved safety issue support Evaluate BWR ECCS response under accident conditions. Advanced instrumentation to meet research needs and reactor safety needs.	10CFR50, App. A Criterion 35 & 50, Appendix K, and planned rulemaking	. Provided data on: - Bundle heat transfer under degraded core cooling conditions. - S.G. performance under accident conditions. - Pressurized thermal shock to vessels. - PWR ECCS systems performance. - Bundle blockage effects under severely damaged core conditions. - BWR separate effects data to support TRAC-BWR assessment - Natural circulation long term cooling.
2D/3D Program	Conduct large-scale test of emergency core cooling during refill-reflood portion of large-break loss-of-coolant accident, and natural circulation during a small-break loss-of-coolant accident.	10CFR50, App. A Criterion 35, & App. K.	Provide reflood data concerning steam binding assumptions in Appendix K. Provide small break natural circulation data. Provide blocked bundle data for reflood assumptions in Appendix K and degraded core rulemaking. Provide large-scale ECC bypass information for design base accident.

PRC ITLE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
Code Improvement and Maintenance	Develop computer codes to analyze reactor operational transients, small breaks, and design base accidents.	10CFR40, App. A Criterion 35, 38 & 50, App. K, and planned rulemaking	Codes used in audit of PWR and BWR plants. Codes used to calculate plant response to LOCAs and operational transients Codes used to evaluate operator guidelines.
Code Assessment and	Assess the accuracy of computer codes to analyze reactor operational transient small breaks and design basis accidents. These will be used to evaluate plant response and licensee conservatism.	10CFR50, App. A Criterion 35, 38 & 50, App. K, and planned rulemaking.	Evaluate vendor margins in design base accident analysis. Evaluate ECCS response under operational transients and LOCAs. Provide analysis base for licensing and safety issues, such as pressurized thermal shock.
Fuel Behavior Under Operational Transients	Determine fuel and clad behavior under anticipated operational transients and design basis accidents.	10CFR50, App. A Criterion 35, 50 & 60, and App. K	Previous data have been used to assess and confirm these areas in 10CFR50. Current work to confirm clad ballooning and blockage for DBA-LOCA will complete this effort.
Loss-of-Fluid-Test (LOFT)	In a scaled PWR, test the hydraulics and fuel thermal transients during large break LOCAs and operational transients and anticipated transients without scram (ATWS). During these accidents, assess the response of plant instrumentation and new operator diagnostic and display techniques in helping the operator to identify and respond appropriately to the accident. Assess the effectiveness of prescribed plant recovery procedures.	10CFR50, App. A General Design Crit. .GDC 13 Instru. & Control .GDC 15 Reactor Coolant System Design .GDC 35 Emergency Core Cooling System App. K-ECCS Evaluation Model <u>Task Action Plan (NUREG-0660)</u> I.D.5 Control Room Design II.E.2 ECCS LOFT - specifically referenced.	Helps evaluate the degree of conservatism in 10CFR50, App. K. Tests confirmed that the emergency core cooling system required by 10CFR50 prevents major core damage during design basis loss-of-coolant accidents. Large-scale data on ECC bypass used to improve codes. Small-break pumps off/on tests provided data needed to determine the licensing position on issue of tripping reactor coolant pumps during small break. Operational transients and multi-failure accidents provided data to confirm corrective operating procedures and effectiveness of various cooldown procedures including natural circulation. Intermediate sized break used to assess codes and models. Large break LOCA test L2-5 with pressurized fuel will address the margins to clad ballooning and rupture in FY82. The generic issue of plant response to an anticipated transient without scram will be addressed by two ATWS tests in FY82.

PR TITLE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
Loss-of-Fluid Test (cont'd)		I.A.2 Training & Qualification of Operators	The generic issue of plant response to an anticipated transient without scram will be addressed by two ATWS tests in FY82.
		I.A.4 Simulator Use & Development	Five anticipated transients will study the boron dilution accident, the control rod withdrawal accident, and the steam generator tube break which recently occurred in the Ginna reactor.
		I.C Operating Procedures	
		II.B Degraded Cores	Large break fuel damage test in FY83 will demonstrate core coolability when in a degraded cooling condition, and confirm criteria for clad ballooning and burst in a large fuel bundle in actual nuclear core accident conditions.
Accident Evaluation and Mitigation	Research on coolability of severely damaged cores and severe accident hydrogen and radiological source terms. Severe accident sequence analysis and mitigation of effects of combustible gases and retention of molten core materials.	Planned rulemakings	Support for accident management procedures for the prevention, early termination and/or mitigation of severe reactor accidents. Specific application to the NRC development of hydrogen control rules, reevaluation of the technical basis for radiological source term for use in the development of policy on siting and emergency preparedness and development of policy on equipment qualification requirements. Direct inputs to the improvement of the phenomenological base for reliability and risk assessments.
Fast Breeder	An integrated research program to provide NRC with data and methods of analysis to regulate breeder reactors when required.	10CFR50 10CFR100	Support for NRC licensing actions on CRBR. Development of Generic Design Criteria, Siting Criteria and Regulatory Guides and Standards for LMFBR.
Advanced Converter	Development of safety and licensing related data and evaluation methods for gas-cooled, graphite moderated nuclear power reactor plants (HTGRs). Work currently focuses on preparations to enable licensing of the new, more advanced design HTGRs including an early examination of siting characteristics.	10CFR50 10CFR100	Support for NRC licensing actions and for preapplications and for preapplication review for large demonstration plant.

P	TITLE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
	Mechanical and Structural Engineering	Develop methodologies and provide information to characterize the behavior of structures, components, systems and equipment under operational, environmental and postulated accident conditions. Verify mechanical/structural computer codes used to perform safety analyses. Determine the behavior of structural and mechanical systems under earthquake and other accident loads in a probabilistic format to risk. Assess how loads are combined. Evaluate qualification criteria of mechanical components and equipment.	10CFR50, App. A GDC 1-5 10CFR50, App. B 10CFR50, 50.55a 10CFR100, App. A	Assess and recommend improvements to the Standard Review Plan Sections 2.4, 2.5, 3.5, 3.7, 3.8 and 3.9. Provide research support for unresolved safety issues pertaining to asymmetric blowdown, seismic design, snubbers, tornado missiles and safety relief valve dynamic loads. Support rulemaking on degraded core cooling by determining safety margins and modes and probabilities of failure for structural components. Support Systematic Evaluation Program for operating plants. Support development of Regulatory Guides applicable to mechanical/structural components and systems. Support rulemaking on qualification of mechanical equipment, including research to develop regulatory guides on equipment, qualification in support of NRR action plan.
	Fracture Mechanics	Development of elastic-plastic fracture mechanics analysis, test method and data base.	10CFR50, App. A, GDC 14, 15, 30, 31; App. A, G.	Assess RPV and pipe toughness in accidents
		Develop thermal shock analysis and test validation.	10CFR50, App. A.	Modify piping design basis accident; assist resolutions of asymmetric load problem.
		Integrity of cracked, degraded piping.	10CFR50, App. A. ASME Boiler and Pressure Vessel	Upgrade, rewrite piping design rules. Assist S.E.P. evaluation of past pipe fracture consequences.
		Probability and structural consequences of pipe breaks.		

PRO. FILE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
Operating Effects	Embrittlement of RPV Steel.	10CFR50, App. G, H.	Modify or confirm vessel toughness requirements.
	Fatigue crack growth	10CFR50, App. A, GDC 30, 31.	Assess future safety of component with growing flaw per ASME-XI.
	Stress corrosion cracking of BWR, PWR piping.	10CFR50, App. A. GDC 14, 15, 31	Assess "fixes" prop. by vendors; modify water chemistry procedures.
	Integrity of steam generator tubing.	10CFR50, App. A, GDC 14, 15, 31. ASME Boiler and Pressure Vessel Code	Assess safety of cracked tubes; assess proposed design and operating.
Nondestructive Examination	Reliability of flaw detection and evaluation of current code methods.	10CFR50, Sec. 50, 55a, App. A, GDC 1, 32.	Tighter inspection of ASME-XI will find smaller flaws.
	Improved flaw detection and accuracy of characterization during ISI.	App. B, Criterion XII.	Improved techniques and equipment to find smaller flaws.
	Continuous monitoring of reactors for on-line evaluation of structural integrity.	ASME Boiler and Pressure Vessel Code	Early warning of cracking, and guide for better ISI during shutdowns.
Fuel Cycle Facility Safety	Develop experimental data and validated analysis methods to assess radiological source terms for accidents of major consequences in fuel cycle facilities.	10CFR40.32 10CFR50.35 10CFR50.50 10CFR50.57 10CFR70.23 10CFR70.31 10CFR72	Provide data on possible design basis accidents and methods for assessing safety in fuel cycle facilities.
	Develop experimental data and validated analysis methods to assess criticality safety for configurations applicable to fuel fabrication, shipping and storage.	10CFR70.22 10CFR70.24 10CFR72 10CFR71.33 10CFR50	Provide NRC staff with improved capability to verify criticality safety for configurations and conditions not previously assessed.

PRO. TLE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
Fuel Cycle Facility Safety (continued)	Collect and analyze experimental data to assess the effects of storing spent LWR fuel in a dry environment.	10CFR72	Provide data to establish and verify existing criteria for storing spent fuel.
Decommissioning	Determine levels of radioactivity present in nuclear facilities, assess techniques, safety and costs for reducing residual levels to specified levels and assess methods to verify these residual levels.	10CFR20 10CFR30 10CFR40 10CFR50.82 1[illegible]R70 10CFR72	Provide technical data to support the development of decommissioning standards and guides and to evaluate licensee implementation.
Effluent Control	Provide experimental data on the radionuclide concentrations in waste streams in LWRs and the performance of methods to reduce releases to the environment.	10CFR20.106 10CFR20, App. B 10CFR50.34 10CFR50.34a 10CFR50, App. I	Provide a technical basis for licensing evaluation of waste stream characteristics and the performance of waste treatment systems to assure effluents are ALARA.
Electrical Equipment Qualification	Evaluate qualification test methodologies including test sequencing and aging effects.	10CFR50, App. B. Criterion III	Provide a technical basis for licensing evaluation of proposed qualification test programs.
Fire Protection	Evaluate and validate positions in present and proposed fire protection.	10CFR50, App. A Criterion 3 and App. R	Provide technical bases for revision to App. R and for new regulations establishing fire protection and requirements for new plants.

PROJ ILE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
Human Engr/Man-Machine	Evaluate ways to enhance human performance for safe operations of nuclear facilities.	Quantification of human performance for use in probabilistic risk analysis	Regulations - 10CFR34, staffing requirements. 10CFR54, conditions of licensee - fitness for duty. 10CFR50, Appendix A, General Design Criterion 19 - Control Room Design; and 10CFR55, Operator Training Requirements. 10CFR50.54 Control Room Staffing.
		Improved reactor safety through human engineered equipment and facilities.	Results providing input to Control Room Review Guidelines, NUREG-0700 and standard review plans relevant to human factors. Providing input to Reg. Guides: 1.8 Operator Training 1.47 Status Monitoring 1.97 Accident Instrumentation 1.97 Accident Instrumentation 1.149 Training Simulators HF-608-4 ISFSI Certification and Training
		ANSI 3.1 Selection, Qualification and Training of Personnel at Nuclear Power Plants	Standards: ANS3.5 Simulators ANS N-660 Operator Action Time IEEE Colors and Symbols IEEE Subcommittee 7 on Human Factors Human Engineering Design Requirements IEEE 566 and 567, Control Room Design
		ISAdS67.14 Certification and Training of Instrument and Control Technicians	Guidelines and criteria for computerized diagnostic and display systems.
Emergency Preparedness	Improve the capability of Federal, state and local government and licenses to licensees to mitigate the consequences of an accident.	10CFR50.33, 50.47, 50.54 and App. E	Provide input and upgrade Reg. Guides 1.101, 2.6 and 3.42 and emergency preparedness regulations.
		10CFR30, 40, 70	Establish emergency preparedness requirements for fuel cycles and material licensees.

PROJE. ΊLE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
Instrumentation and Control	Evaluation of Safety Implication of Control Systems and Associated Electric Power Systems	10CFR50, App. A Criteria 2, 13, 17, 18 and 26.	Principal effort in Resolution of USIA-47
	Evaluation of design, manufacture, installation, operation, periodic testing and maintenance practices of I&C and electric equipment important to safety.	10CFR50, App. A, Criteria 2 and 13	Technical Basis for Guidelines on the design, manufacture, installation, operation, testing and maintenance of electric equipment important to safety.
	Assessment of the state-of-the art and current practices for measuring important parameters during the cause of and following an accident.	10CFR50, App. A, Criterion 13	Data Base needed for regulatory decisions on Implementation of R.G. 1.97.
	Evaluate regulatory impact of use of programmable digital computers with associated isolation devices and other advanced concepts in safety, control, alarm and information systems.	10CFR50, App. A,	Regulatory Guides on Programmable Digital Computers and Isolation devices.
	Diagnostic Instrumentation Research to Monitor Reactor Safety	10CFR App. A, Criteria 2 and 13	Technical Basis for Licensing Criteria for Diagnostic Instrumentation.
Occupational Protection	Assess the sources of high radiation exposures in LWRs and evaluate safety, effectiveness and cost of alternative decontamination methods and operational approaches to reduce these exposures.	10CFR20.1 10CFR20.101 10CFR20.105 10CFR50.34 EPA Occupational Exposure Guidelines	Provide a comprehensive experimental basis for assessing licensee plans to assure that collective (manrems) occupational exposures are ALARA; a technical basis for modification of approaches to ensure ALARA.

PROJEC .LE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
Communicated Threat Assessment	Develop capability to perform comprehensive assessments of nuclear threats.	10CFR73	Provide advice and recommendations to the NRC, FBI, etc., on credibility of nuclear threats.
Reactor Vulnerability Studies	To identify power reactor systems which are vulnerable to insider sabotage and safeguard measures to offset this vulnerability.	10CFR73	Provide NRC the technical basis for assessing the adequacy of current regulations with regard to insider sabotage.
Safeguards Aspects	To identify human factors considerations which could impact negatively on safeguards systems and to identify a way to mitigate or eliminate such impacts.	10CFR73	Provide the NRC with a technical basis for assessing the adequacy of current regulations with regard to human factors in safeguards.
Power Reactor Safety/ Safeguards Interface	To identify and resolve safety/ safeguards interface problems at nuclear power reactors.	10CFR50 and 73	Provide the NRC with technical basis for assessing current regulations with regards to safety/safeguards interface problems.
Integration of Vital Area Products	Develop, demonstrate, and transfer a method of organizing and rapidly retrieving existing site-specific systems information in response to reactor safeguards events.	10CFR50 and 73	Permit timely retrieval of site-specific information to support NRC's analysis of and response to safeguards events at power reactors.
ISFSI Design Study	To identify probable designs which will be used for storing spent fuel.	10CFR72 and 73	Provide information which can be used for assessing adequacy of safeguards at ISFSIs.
Standards and Measurement Control for NDA	To improve measurement methods standards, for calibration, calibration methods, and procedures for the quality control of measurement systems.	10CFR70	To provide the technical basis for regulatory decisions associated with measurement technology.

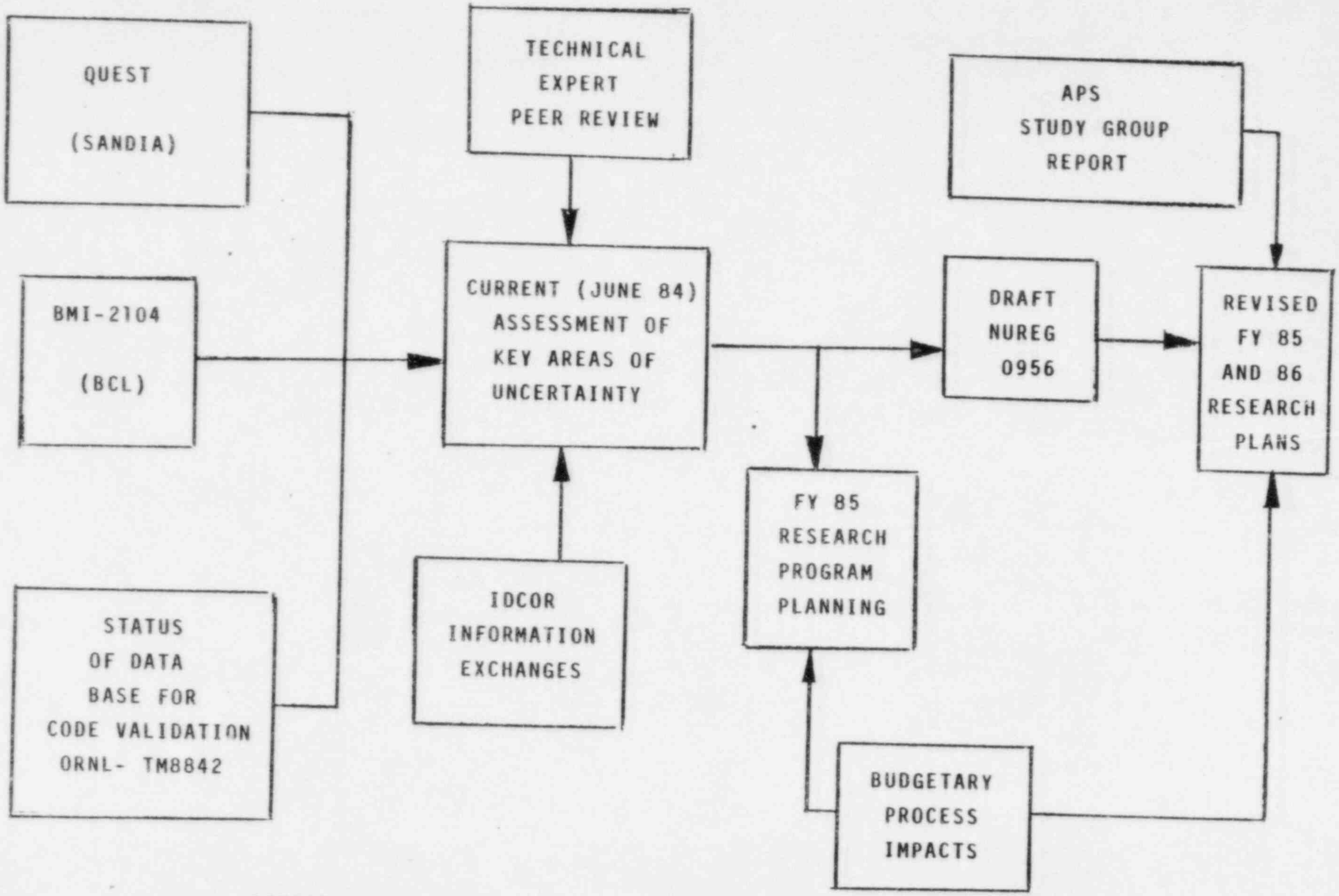
PROJE. .LE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
Perimeter Alarm Testing	Evaluate under harsh winter conditions specific perimeter alarms systems used at power reactors.	10CFR73	Provide the NRC with a technical basis for assessing adequacy of current regulations in regards to perimeter alarms.
Statistical Methods for Nuclear Material Accountability	Provide standardized statistical procedures to be used by the nuclear industry.	10CFR70	Will serve as the basis for revision of several statistical regulatory guides.
Recommendations for Strenghtening IAEA Safeguards	Analyze past and current initiatives for strengthen-IAEA safeguards in order to identify areas for further NRC efforts.	10CFR75	Will identify areas of 10CFR75 for revision or update.
Passive NDA Reference Manual	Documents current uses of Passive NDA as used by the Nuclear Industry.	10CFR70	To be used as a guide for licensing process.
13 Statistical Deficiencies	To develop methods and approaches that will address the deficiencies noted in SECY 80-514.	10CFR70	To help standardize statistical terminology and to address deficiencies in the use of statistics in NRC regulations.
Estimation Methods for Material Holdup	Describe experimental programs to better estimate the nuclear material held up in process equipment.	10CFR70	Provides guidance for licensee implementation of proposed prompt accountability requirements.
Reactor Risk Analysis Methods Development & Data Evaluation	Develop methods for analyzing reactor risk	10CFR50, App. A	Review susceptibility of reactors to human error and equipment failure. Develop methods to apply safety goal and decision theory to safety regulation.
	Predict operator and maintenance errors.	10CFR50, App. A	Operator training, procedures and control room design.
	Assess equipment failure rates in reactor plants	10CFR50, 34	Review industry analysis of reliability data and recommend acceptable values for use in risk assessments.

PK TITLE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
Reactor Risk Analysis	Identify risks of nuclear power plants.	10CFR50, 100	Discover strengths and weaknesses in regulatory program.
	Study sensitivity of risk prediction models to explore safety issues	10CFR50, App. A 10CFR50, App. E 10CFR100	Support reactor safety standards development in rulemakings, resolution of generic issues, and regulatory reform.
	Improve methods for predicting consequences of large releases of radioactivity.	10CFR50, App. E 10CFR100	Siting, emergency planning, rulemaking and policy studies.
	Train selected NRC regulatory staff members in risk assessment methods and results.	N/A	Risk Assessment is becoming a licensing technique.
Transportation and Materials Risk	Develop methods for analyzing risk for fuel cycle facilities	10CFR30, 40, 70, 72	Evaluate safety, over-regulation, and weak spots in standards for fuel cycle, material facilities, and transportation.
	Assess the ability of products containing licensed nuclear materials to safely respond to use, accident, and disposal environments.	10CFR30.33 10CFR40.32 10CFR70.32	Provide experimental demonstration of the adequacy of the current licensing approach or a technical basis for modification to ensure safety.
	Define and/or provide capability to evaluate radioactive material transportation package safety under normal and abnormal transport conditions. Establish a set of radioactive material package performance tests based on mode dependent transport conditions.	10CFR71, App. A.4 10CFR71.37 10CFR71.36 10CFR71, App. B.1 10CFR71, App. A.2 10CFR71, Subpart C, App. B	Clarify ambiguous performance test specifications; provide state-of-the-art analytical tools to establish compliance. Provide technical basis for decision on revision of 10CFR71 to adopt modal standards.

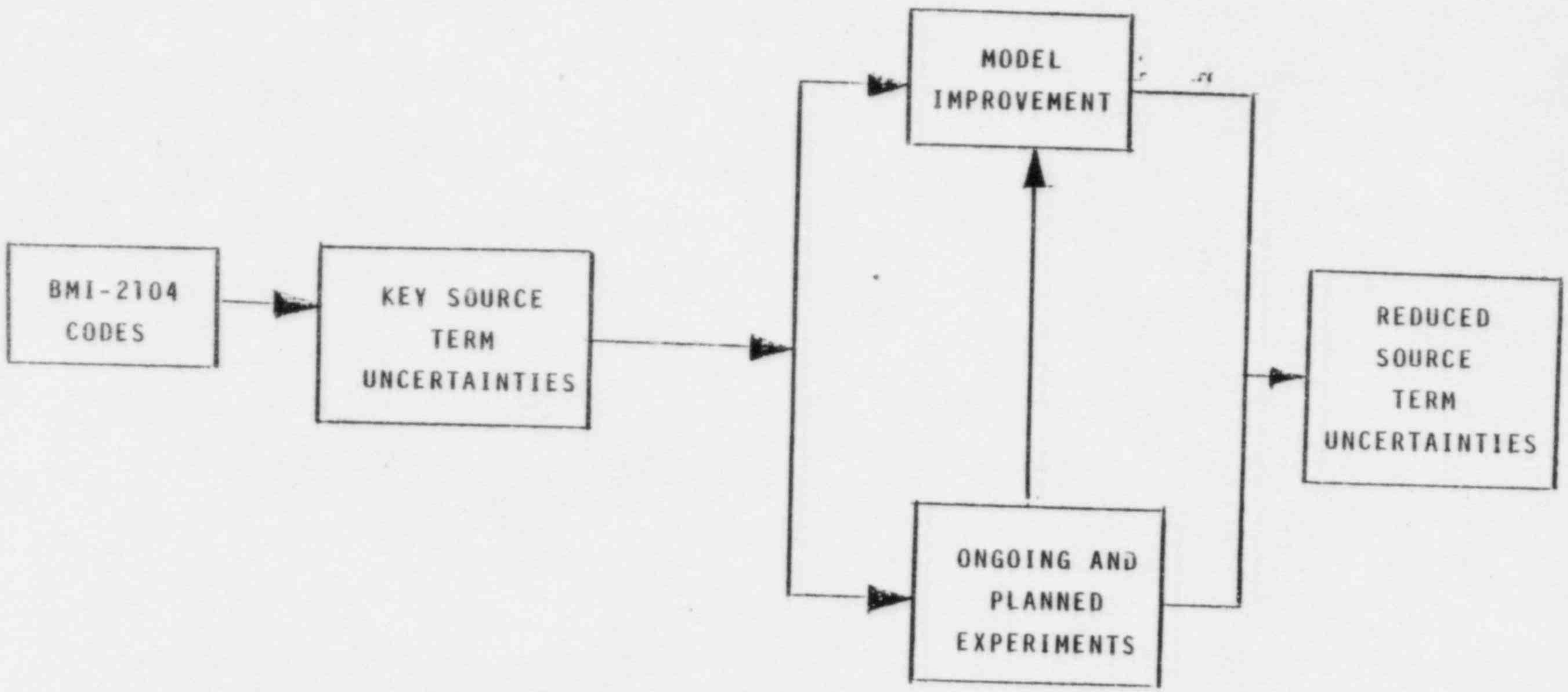
PROJE .LE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
High Level Waste (HLW) Package and Geochemical Interaction	Test methods to predict long-term performance of proposed matrix materials for high level waste forms and containers and assess release rates of radionuclides	10CFR60, para. 60.111	Provide technical data and test methods that will be used to guide the development and to assess the performance of waste forms and packages with respect to containment and controlled release of radionuclides.
HLW Repository Siting	Develop information to identify critical technical requirements for HLW repositories.	10CFR60, 20 EPA Standard 40CFR191	Provide technical bases for formulating regulatory guide rules and regulatory decisions, for ensuring that the host rock will safely accommodate repository structure and that site and environs provide adequate long-term natural barriers to radionuclides.
HLW Repository Design Operational Safety and Performance Reliability	Assess methods to evaluate reliability and safety of design, structures and operational plans and the reliability of sealants, backfill materials and monitoring methods.	10CFR20, 60 (EPA and State regs.)	Ensure that regulations will provide containment capability of repositories, and permit compliance with health and safety requirements and with environmental quality requirements.
Systems Analysis and Risk Assessment of HLW	Develop and demonstrate the applicability of methods for risk predictions of HLW repositories in various media	10CFR60 and EPA Standard 40CFR191	Methods developed will be used to enable NRC to use applicant's plans and data to independently predict reliability of proposed repository performance and help determine the adequacy of characterization information.
Low Level Waste (LLW) Form and Container	Test methods and data to confirm the properties of low level waste forms and containers to assess their performance in shallow land burial facilities.	10CFR61	Provide technical bases for technical directives, and regulatory guides with respect to performance of waste forms and containers.
Shallow Land Burial of LL Radioactive Wastes and Alternative Disposal Methods	Evaluate performance of existing shallow land burial sites; test validity of site characterization methods, facility designs and monitoring and site closure methods.	10CFR30, 51, 61, 150	Provide technical bases for assessing and improving regulatory actions for siting, designing, operating and monitoring shallow land burial and alternative disposal methods. Works cooperatively with states (Kentucky and New York) in developing site closure methods.

PROJ. TITLE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
Uranium Recovery Waste Characterization, Operations; Management Decommissioning and Surveillance	Evaluate operational methods and identify alternative procedures to reduce risk and evaluate decommissioning and surveillance techniques to prevent radioactive and toxic substances from escaping from uranium mill tailings	10CFR20, 40, 51	Provide technical bases for uranium mill tailings regulatory criteria and tested methods for assuring that unacceptable amounts of radon, airborne particulates and liquid wastes are not discharged into the environment.
Uranium Recovery Siting, Pathways Impacts	Assess methods for selecting sites for uranium mill tailings and for predicitng the risks to surface water and groundwater contamination by radioactive and toxic substances from uranium mill tailings sites and solutions mining.	10CFR20, 40, 51	Provide technical basis for regulatory standards and confirmatory data to support licensing actions to ensure the long-term effectiveness of mill tailings and solution mining management procedures and assure protection to the health and safety of the public is adequate.
Site Safety			
Seismology-Geology	Studies of earthquakes and other geologic hazards to develop realistic assessment of potential for ground shaking ground rupture, soil foundation failure, and volcanic effects at nuclear facilities sites. Assessment of feasibility of seismic regionalization.	10CFR100, App. A (Seismic and Geologic Siting Criteria for Nuclear Power Plants)	Provide basis for future rulemaking on Appendix A 10CFR100. Clarify ambiguities in the Regulation. Improve capability to utilize advances in the state of knowledge in earthquake sciences. Eliminate redundancies in license applications concerning regional earthquake potential.
Meteorology-Hydrology	Acquisition and analysis of high quality field test data on meteorological dispersion of accidentially released radionuclides out to 50 miles and over a variety of terrains typical of nuclear power plant sites.	10CFR100, App. E	Verify models and calculations used to satisfy requirements for site evaluation and emergency response. Update methods and bases for revision of Draft Regulatory Guide 1.145.

PR	TITLE	DESCRIPTION	RELEVANCE	APPLICATION TO REGULATORY PROCESS
		Characterize severe storm hazards (tornados, lightning and floods) in different regions	10CFR50, App. A	Verify levels of conservatism of design bases for natural hazards.
		Studies to develop criteria.	Generic Site criteria for existing & future sites considering effects of Class 9 accidents	Develop siting criteria and specify mitigation practices to prevent contamination of public water supplies and damage to the environment through this segment of the liquid pathway.
	Aquatic & Airborne Effluents - Environmental Impacts	Radionuclide transport in surface waters & effects of effluents on biota	10CFR20 10CFR50, App. I 10CFR51	Technical bases to support licensing judgments and for environmental impact statements to implement regulations.
		Airborne effluent pathways; models and codes		Environmental impact; population dose assessments.
	Health Effects	Assess the impact of radiation exposure from facilities and activities regulated by NRC on human health and safety.	10CFR20	To develop regulatory guidance, set exposure standards and provide technical data for risk/benefit analyses.
	Siting	Assess potential sites from the perspective of population density and land use, and develop methods for evaluating alternative sites.	10CFR100 10CFR51 10CFR50, App. Q	Siting rulemaking, Early Site Review, Environmental Impact Assessment, Evaluation of Alternative Sites.
		Assess social and economic impact of nuclear reactors	National Environmental Policy Act (1969) 102,(2),C 10CFR51, App. Q	Environmental Impact Assessment, Cost-Benefit Analysis, Evaluation of Alternative Sites.



SOURCE TERM RESEARCH PROGRAM PLANNING PROCESS



RESOLUTION STRATEGY FOR KEY SOURCE TERM UNCERTAINTIES

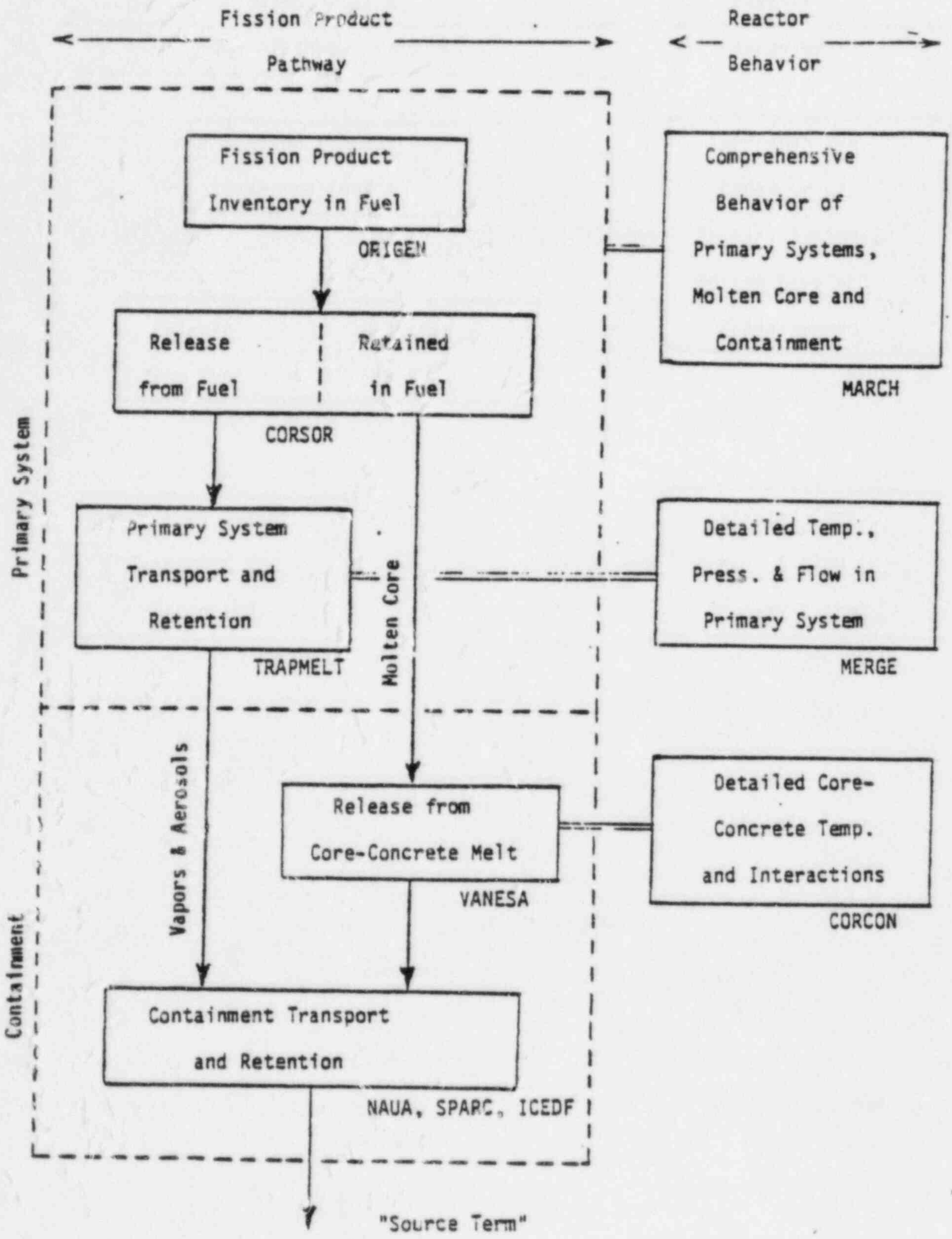


Figure 1. Battelle suite of codes as used in BMI-2104.

MAJOR UNCERTAINTY AREAS IN BMI-2104 SOURCE-TERM ANALYSIS

DESCRIPTION OF PHENOMENA	BMI-2104 CODES	AREAS OF MAJOR UNCERTAINTY
PRIMARY SYSTEM THERMAL-HYDRAULICS	MARCH, MERGE	UPPER PLENUM RECIRCULATION
FUEL HEATUP AND DEGRADATION	MARCH	CORE MELT PROGRESSION
FISSION PRODUCT RELEASE FROM FUEL (IN VESSEL)	CORSOR	CONTROL ROD VAPORIZATION Te RETENTION
PRIMARY SYSTEM FISSION PRODUCT TRANSPORT	TRAPMELT	FISSION PRODUCT DEPOSITION AND RETENTION
DEBRIS-CONCRETE INTERACTIONS	CORCON	
FISSION PRODUCT RELEASE FROM CORE-CONCRETE MELT	VANESA	Te RETENTION
CONTAINMENT THERMAL-HYDRAULICS	MARCH	
HYDROGEN BEHAVIOR	MARCH	
CONTAINMENT FISSION PRODUCT TRANSPORT	NAUA, SPARC, ICEDF	

Area of
Uncertainty: Upper Plenum Recirculation

Description: In the MERGE code, it is assumed that the flow of gases and aerosols in the upper plenum is one-dimensional. In reality, it would be expected that circular flow patterns would be established in this region because of steep temperature gradients. Accounting for this natural recirculation would significantly increase the calculated effective area for aerosol deposition and vapor condensation causing a substantial increase in predicted fission product retention in the primary system.

Resolution: There is underway a small experimental program sponsored by EPRI at Westinghouse to investigate the natural convection pattern in simulated upper plenum designs. These results and the utilization of more detailed codes like TRAC and RELAP can be used to investigate the consequences of one-dimensional simplistic modeling. Present work at SNL with a TRAC/MIMAS code linkage will quantify this deficiency. Modifications to MELCOR in this area will be made if indicated by the TRAC/MIMAS studies.

Area of
Uncertainty: Core Melt Progression

Description: The modeling of core melting, slumping, and vessel melt-through is uncertain because of the absence of large-scale tests, and uncertainties in related hydrogen generation and melt composition (and quantity) have been identified as very important because of their effects on containment loads and ex-vessel releases respectively.

Resolution: Major experimental programs in the ACRR, PBF, and NRU test reactors are underway to study core melt progression. Improved out-of-pile meltdown tests are also planned at ORNL (Parker). Revised out-of-pile tests to study fission product and aerosol release at ORNL (Lorenz) and KfK (SASCHA) are also expected to provide some melt-progression information. These experimental programs are strongly coupled to SCDAP (before melting) and MELPROG (after melting) modeling work at INEL and SNL.

Area of

Uncertainty: Control Rod Vaporization

Description: Earlier KfK tests produced a lot of silver vaporization while ORNL tests produced only a little. Using this common data base, NRC's contractors assume significant Ag aerosol generation in their model while IDCOR's contractors assume none. The timing of any control rod vaporization is also very important because Ag aerosols can be carriers for fission products and affect deposition only if they are present when the fission products are being released.

Resolution: Most of the core-melt-progression experiments include tests with silver-alloy control rods, and the new out-of-pile tests at ORNL (Parker) and KfK (SASCHA) will specifically address this issue. TMI-2 control rod leadscrew examinations are also contributing data on Ag aerosol deposits. Explanation to reconcile the earlier KfK and ORNL test results have bee put forward, and the expected new data should lead to model improvements. The new modeling will be incorporated in the VICTORIA subroutine of MELPROG.

Area of
Uncertainty: Te Retention

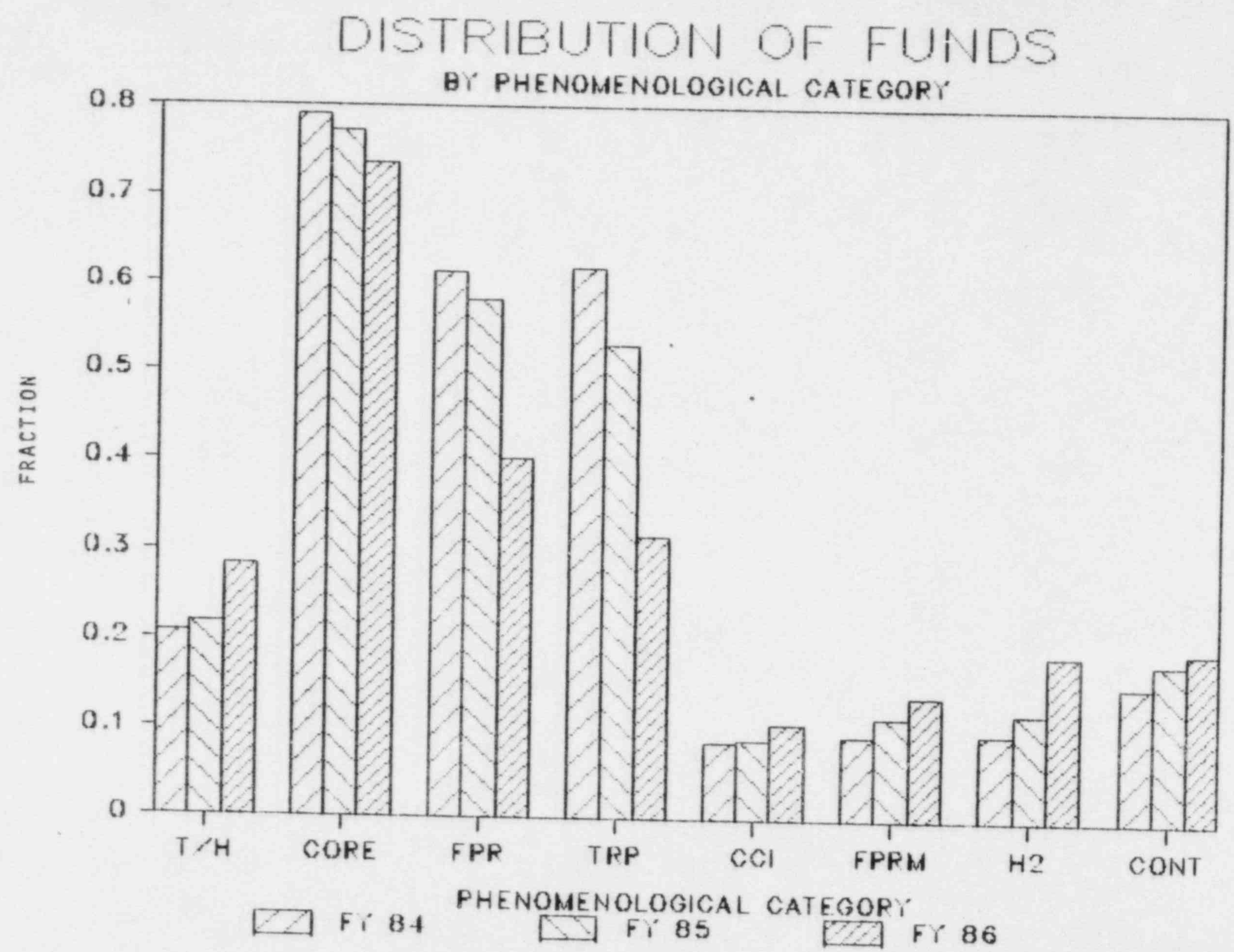
Description: Tellurium is a very reactive element that readily forms tellurides with other metals. Evidence exists that Te is retained in unoxidized Zr cladding, and rudimentary modeling of this effect is included in BMI-2104. However, it is not known exactly how Te will be released from the Zr as Zr oxidation progresses in the vessel or how Te will be released from the core melt during its reaction with concrete in the containment. Furthermore, it is not known if Te will react with stainless steel surfaces in the vessel or in the containment and be retained there after release from the core.

Resolution: The fission-product-release programs at ORNL (out-of-pile) and PBF are all specifically investigating the behavior of Te. The TMI-2 core examinations will also attempt to determine the disposition of Te. Recently initiated mass-spectrometry studies of the reactions of Te with Zr will provide basic vaporization and reaction data. The high-temperature chemistry program at SNL should also provide supporting information. Modeling of these results will be done in VICTORIA and TRAPMELT (in-vessel) and VANESA (ex-vessel).

Area of Uncertainty: Fission Product Deposition and Retention

Description: Deposition and retention of fission products during transport through the upper plenum and primary system piping are calculated by the TRAPMELT code. The predicted retention on surfaces in many cases is substantial, but the uncertainty regarding the amount of deposited fission products is large because several potentially important phenomena are neglected (chemical reactions between deposits and surfaces and between vapors and deposits, and local addition of decay heat where fission products are deposited). Thus local temperature increases due to the deposits themselves may cause revaporization of some of the fission products.

Resolution: A high temperature chemistry program is underway at SNL to determine.chemical interactions with primary system surface materials of fission products in the vapor phase. A study at ORNL of fission products and aerosols will provide experimental data for the controlling deposition mechanisms. Large aerosol transport tests (MARVIKEN, LACE) will also contribute to improve current modeling. Presently we have no experiments to examine the revaporization aspect, but EPRI is conducting small-scale experiments at ANL to investigate aspects of revaporization from metal surfaces. We also have under investigation the possibility of utilizing mass spectrometry techniques to setup single experiments addressing revaporization of selected fission products from surfaces. Other proposals are also being evaluated. Modeling improvements reflecting new results will be made to TRAPMELT.


DISTRIBUTION OF FUNDS
BY PHENOMENOLOGICAL CATEGORY
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DISTRIBUTION OF FUNDS BY FY

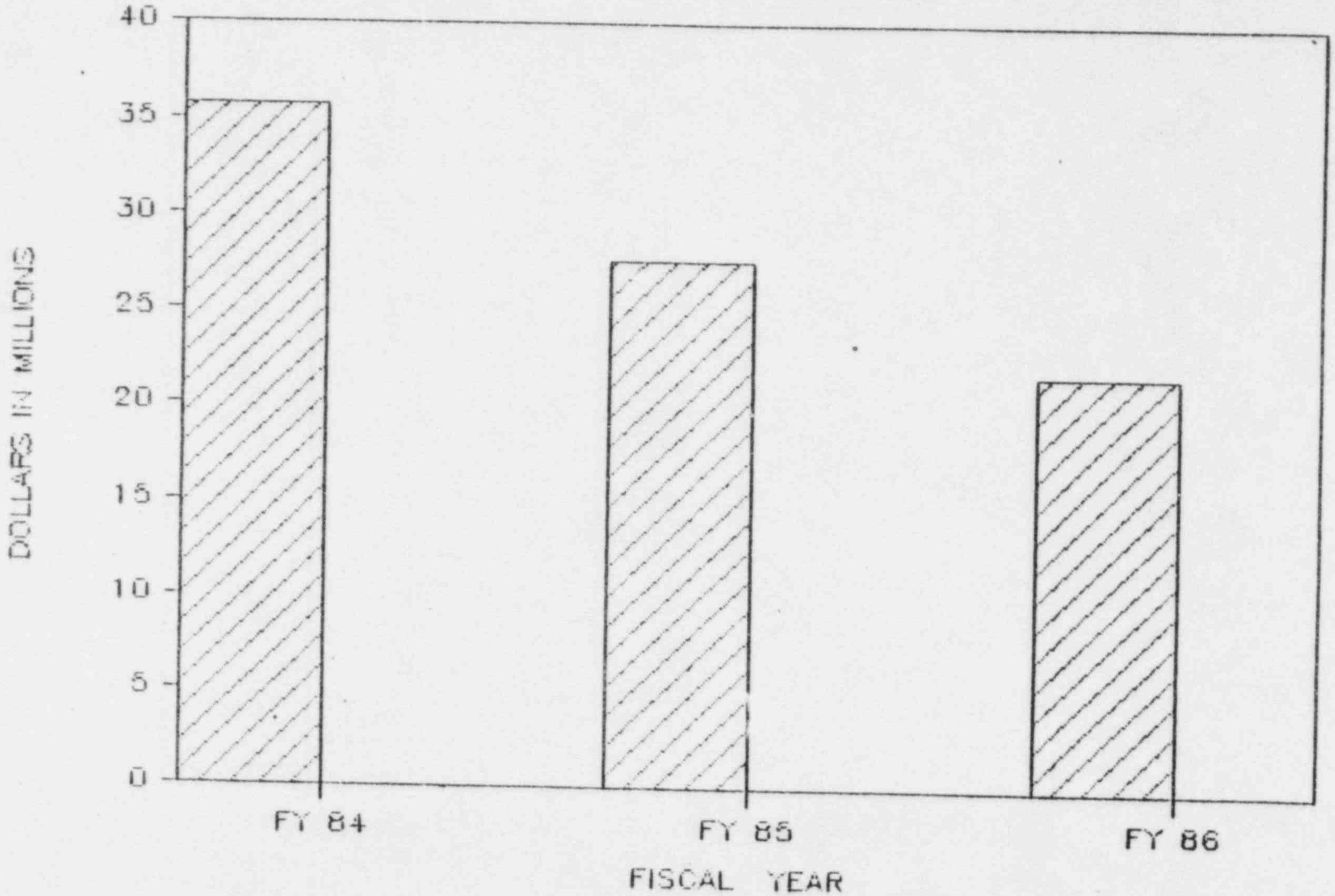


Table 1. Newer codes that have the potential for upgrading components of the Battelle suite of codes used in BMI-2104.

Description	BMI-2104	Newer Code
RCS Thermal-Hydraulics	MARCH, MERGE	TRAC, RELAP-5
Fuel Heatup and Degradation	MARCH	SCDAP, MIMAS, MELRPI, MELPROG
Fission Product Release from Fuel (in vessel)	CORSOR	GRASS, VICTORIA
RCS Fission Product Transport	TRAPMELT	TRAPMELT
Molten Fuel Interaction with Coolant	MARCH	WISCI
Debris-Concrete Interactions	CORCON	CORCON
Fission Product Release from Core-Concrete Melt (ex-vessel)	VANESA	VANESA
Containment Thermal-Hydraulics	MARCH	CONTAIN
Hydrogen Behavior	MARCH	CONTAIN (HECTR)*
Containment Fission Product Transport	NAUA, SPARC, ICEDF	NAUA, CONTAIN (SPARC, ICEDF, MAEROS)*

* Subroutines in CONTAIN

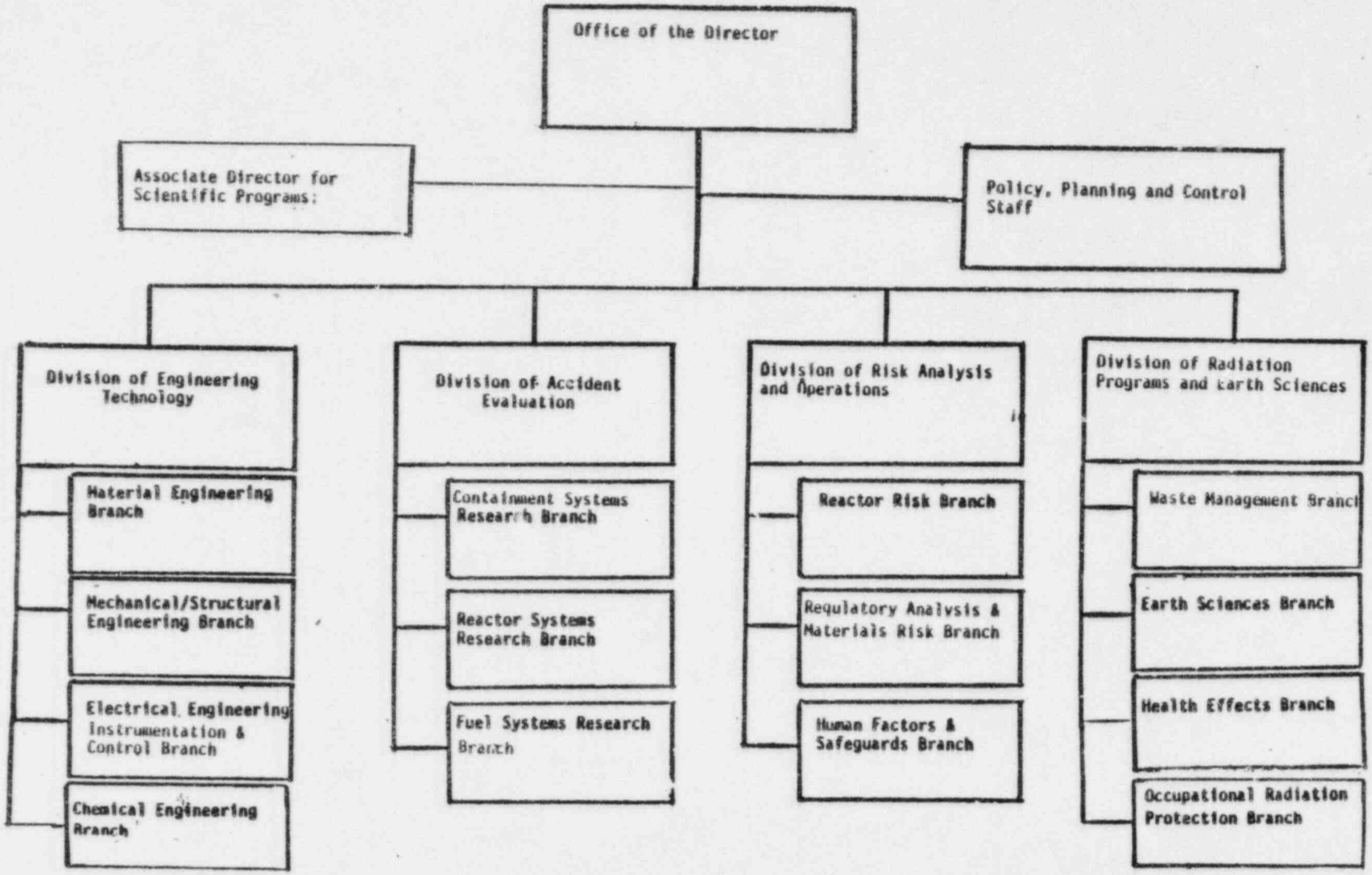
Table 2. Corresponding Research Project to Develop or Upgrade Newer Codes.

Newer Code	Description	Research Project Number
TRAC, RELAP-5	RCS Thermal-Hyrdaulics	3, 4, 7, 10, 14, 26, 31
SCDAP, MIMAS, MELRPI, MELPROG	Fuel Heatup and Degradation	1, 2, 3, 4, 5, 6, 8, 9, 10, 12, 13, 14, 15, 16, 17, 18, 19, 26, 28
GRASS VICTORIA	Fission Product Release from Fuel (in vessel)	4, 5, 9, 10, 11, 12, 13, 14, 15, 16, 18, 19, 20, 22, 26
TRAP-MELT	RCS Fission Product Transport	5, 13, 14, 15, 18, 20, 21, 22, 26, 30, 31, 35
CORCON	Debris - Concrete Interactions	7, 26, 27, 28
VANESA	Fission Product Release from Core-Concrete Melt (ex-vessel)	5, 20, 26, 27, 28
CONTAIN (HECTR)	Hydrogen Behavior	4, 16, 22, 26, 29, 34
NAUA, CONTAIN (SPARC, ICEDF, MAEROS, QUICKM)	Containment Fission Product/Aerosols Transport	16, 20, 22, 25, 26, 29, 31, 33, 34

ORGANIZATIONAL CHART

Office of Research

March 1984
Includes the NRC reorganization.



DIVISION OF ENGINEERING TECHNOLOGY

Plans, develops, and directs comprehensive research and standards programs for nuclear safety and nuclear materials safety in the design, qualification, construction, inspection, testing, operation, and decommissioning of nuclear power plants, nuclear reactors, and fuel cycle facilities with emphasis on the mechanical, structural, materials, electrical, instrumentation and control engineering, and chemical engineering aspects of these facilities and materials; establishes or recommends policy, planning, and procedures for the research and standards programs as required to carry out the functions of the Division; coordinates these research and standards programs with other NRC offices to ensure that the programs are responsive to their needs; provides technical assistance within NRC regarding resolution of generic issues and the development and application of research and standards to the solution of specific safety problems; provides funding guidance to NRC contractors, DOE laboratories, and other government agencies within the Division budget and consistent with NRC policy; maintains liaison with and provides technical input to other Federal Agencies, ANSI, professional societies, international agencies, and other organizations in assigned areas.

Mechanical/Structural Engineering Branch

Develops, recommends, plans, evaluates, and manages research programs and develops standards for the design, qualification, construction, inspection, testing, and operation of nuclear power plants, nuclear reactors, and fuel cycle facilities with emphasis on the mechanical engineering and structural engineering aspects of structures and components. Specifically, this branch has the responsibility for these mechanical and structural engineering aspects, including qualification of mechanical components and related personnel, effects of natural phenomena (e.g., seismic analysis, tornado missiles), classification of mechanical components, load combinations and associated design limits, inservice testing for functional adequacy of mechanical components, vibration, soil structure as a support material, soil/structure interaction, spent fuel shipping casks, waste disposal casks, and cranes. This branch has the lead responsibility for the coordination and interfacing activities associated with Sections III and XI of the ASME Code.

Materials Engineering Branch

Develops, recommends, plans, evaluates, and manages research programs and develops standards for the design, qualification, construction, inspection, testing, and operation of nuclear power plants, nuclear reactors, and fuel cycle facilities and transportation of radioactive materials with emphasis on the materials engineering aspects of structures and components. Specifically, this branch has the responsibility for these materials engineering aspects, including inservice inspection for structural integrity, corrosion, fracture mechanics, thermal shock, effects of environment on materials, and overall nondestructive examination programs and provides general support to other branches within and outside the Division of Engineering Technology for their materials-related needs.

Electrical Engineering, Instrumentation and Control Branch

Develops, recommends, plans, evaluates, and manages research programs and develops standards for the design, qualification, construction, inspection, testing, and operation of nuclear power plants, nuclear reactors, and fuel cycle facilities with emphasis on the electrical and instrumentation and control engineering aspects of these facilities. Responsibilities for the electrical engineering aspects include qualification of electrical components and related personnel, classification of electrical systems and equipment, effects of natural phenomena on electrical equipment, research on protection against electricity-related fires, electric power generating distribution equipment, and the effects of plant electrical transient and anomalies on electrical equipment. Responsibilities for the instrumentation and control engineering aspects include ensuring that the methods of control and the instrumentation used in nuclear facilities are designed, applied, and utilized to minimize the probability of abnormal operation or accidents and to ameliorate the consequences of an accident if one does happen. Also included is the consideration of unusual operating conditions, hostile environments, equipment and material failures, system interactions, and man/equipment interactions.

Chemical Engineering Branch

Develops, recommends, plans, evaluates, and manages research programs and develops standards for the design, qualification, construction, inspection, testing, and operation, and decommissioning of nuclear power plants, and nuclear reactors, and fuel cycle facilities and for nuclear materials with emphasis on chemical engineering aspects of these facilities and materials. Specifically, this branch has the responsibility for these chemical engineering aspects, including water and corrosion chemistry, criticality, decontamination, chemical cleaning, decommissioning, hydrogen control, fission product control, ventilation, and fuel handling, onsite waste treatment and storage, onsite and independent fuel storage, and fuel cycle unit processes. This branch has the lead responsibility for development and coordination of the agency's decommissioning program.

DIVISION OF ACCIDENT EVALUATION

Plans, develops, and directs comprehensive research and standards programs for predicting nuclear power plant behavior under normal and abnormal conditions. These activities include developing systems analysis capabilities related to all plant systems with emphasis on the primary and secondary coolant systems, the containment system, and the full system and considering their interactions with each other and the balance of the plant. Also included is the administration of the support facilities needed to carry out test programs. Establishes or recommends policy, planning, and procedures for the research and standards programs as required to carry out the functions of the Division. Coordinates these research and standards programs with other NRC offices to ensure that the programs are responsive to their needs. Provides technical assistance within NRC regarding resolution of generic issues and the development and application of research and standards to the solution of specific safety problems. Provides funding guidance to NRC contractors, DOE laboratories, and other government agencies within the Division budget and consistent with NRC policy. Maintains liaison with and provides technical input to other Federal Agencies, ANSI, professional societies, international agencies, and other organizations in assigned areas.

Reactor Systems Research Branch

Plans, recommends, evaluates, and manages analytical and experimental research programs on the behavior of the primary coolant system of nuclear power plants, including interactions with the balance of plant under normal, abnormal, and severe accident conditions. These programs include the development of verified codes and models of coolant behavior under accident conditions, the development of an understanding of special thermohydraulic phenomena encountered in abnormal or severe accident conditions and their implications for reactor safety and improved reactor safety system design, and the development of an understanding of the implications of selected precursor events and dominant accident sequences for primary system safety. This branch is responsible for a program of code and model development and verification; the conduct of appropriate test programs to provide the necessary empirical data, including administration of the facilities associated with these programs that are dedicated to NRC; the detailed analysis of selected precursor events; and the development of a plant analyzer and other aids to reactor plant analysis.

Containment Systems Research Branch

Plans, recommends, evaluates, and manages analytical and experimental research programs on the behavior of the containment systems of nuclear power plants, including interactions with the balance of plant, under normal, abnormal, and severe accident conditions. These programs include the development of verified codes and models of the content and behavior of containment atmospheres and the transport of aerosol and fission products under accident conditions; the development of an understanding of how abnormal and severe accidents may be managed so as to reduce the frequency of challenges to containment integrity or mitigate their consequences; and the development of an understanding of the implications of selected precursor events and dominant accident sequences for containment system safety. This branch is responsible for a program of code and model development and verification; the detailed analysis of selected precursor events; the conduct of appropriate experimental programs to provide empirical data necessary to support the assigned analysis and regulatory support responsibilities, including administration of any associated facilities that are dedicated to NRC; and the coordination of LMFBR safety research within the Office of Research.

Fuel Systems Research Branch

Plans, recommends, evaluates, and manages analytical and experimental research programs on the behavior of fuel systems of nuclear power plants under normal, abnormal, and severe accident conditions, including interactions with the primary system boundary. These programs include the development of verified codes and models of fuel behavior in the primary system under the above conditions, including the release and transport of fission products and hydrogen from the fuel into the containment; the devlopment of an understanding of the coolability limits of the fuel system; the determination of how safety and control system interactions affect the fuel system under dominant accident sequences; and the possible implications of selected precursor events. The branch is responsible for a program of code and model development and the conduct of fuel tests and the administration of the test facilities that are dedicated to NRC, verification and analysis of fuel system test data under severe accident conditions, and the coordination of research on the radiological source term and on HTGR safety within the Office of Research.

DIVISION OF RISK ANALYSIS AND OPERATIONS

Plans, develops, and directs comprehensive research and standards programs for independent risk analysis of all elements of NRC-regulated nuclear activity. The activities include systematic evaluation of reactor issues; development of analysis techniques to determine reactor reliability and overall risk, including risk assessments and reliability analysis associated with the nuclear fuel cycle; human factors; and safeguards. Develops and improves risk methodology and translates these efforts into effective tools to aid in making licensing and other regulatory decisions; establishes or recommends policy, planning, and procedures for the research and standards programs as required to carry out the functions of the Division; coordinates these research and standards programs with other NRC offices to ensure that the programs are responsive to their needs; provides technical assistance within NRC regarding resolution of generic issues and the development and application of research and standards to the solution of specific safety problems; provides funding guidance to NRC contractors, DOE laboratories, and other government agencies within the Division budget and consistent with NRC policy; maintains liaison with and provides technical input to other Federal Agencies, professional societies, international agencies, and other organizations in assigned areas.

Reactor Risk Branch

Develops and uses systematic analysis techniques for assessing reactor plants to determine system or overall reliability; overall risk, including radioactive releases and consequences. Prepares reports, standards, and regulations related to its areas of responsibility. The Reactor Reliability Section develops new techniques for systems analysis, systems reliability engineering, and accident sequence prediction and has lead responsibility for applications of risk assessment to reactors in such programs as the Reactor Safety Study Methodology Applications Program and the Interim Reliability Evaluation Program. The Reactor Risk Section develops probabilistic techniques for analyzing the response of reactors in serious accident sequences once core damage has begun. This includes the processes of core melt, energy release, and containment response and the analysis of fission product release and transport within plant systems and in the environment. The Reactor Risk Section develops techniques to use comparative probabilisti analyses as a means to identify and evaluate useful alternatives in reactor siting, emergency planning, and plant design. Develops probabilistic analys methods, assists in collecting reliability data, and prepares reports, standards, and regulations that cover the use of specific probabilistic analysis methods and reliability data in the regulatory process. Work includ research data to obtain component failure rates, human error rates, and the frequency of occurrence of multiple failures of common causes and research t develop probabilistic analytical methods for determining component, subsyste and system reliability. These methods cover challenges and failures from bo external and internal causes and include, to the extent possible, means to identify and quantify uncertainties. This research supports development of quantitative criteria for acceptable reliability and acceptable risk and develops methods to assess compliance with such criteria

Human Factors and Safeguards Branch

Develops and manages research and standards programs on human factors for the safe design, construction, and operation of nuclear facilities. These activities deal with safety-related aspects of the man-machine interface; plant procedures and tests; qualification, training, and licensing of persons in certain functions; and the organization and management of the plant operating staff and the licensee corporate staff as a whole. The pursuit of these activities requires close coordination with the NRR Division of Human Factors Safety, the NRR Quality Assurance Branch, and the Office of Inspection and Enforcement. Develops research and standards programs on safeguards and the protection of certain nuclear materials and facilities. These activities support the NRC's objective of ensuring protection of the public health and safety and the national defense and respond to the unauthorized possession, theft, diversion, or use of special nuclear material and the sabotage of nuclear facilities. The pursuit of these activities requires close coordination with the NMSS Division of Safeguards.

Regulatory Analysis & Materials Risk Branch

Carries out a systematic evaluation (which includes the use of PRA) of relevant safety issues and the information needed to address those issues. Uses the results of NRC and other research programs to identify regulation changes needed to correct deficiencies in safety-significant areas or eliminate unnecessary regulatory constraints. Proposes or initiates rulemaking, as appropriate, and manages complex rulemakings that span the technical or organizational responsibilities of several RES branches or that involve novel or complex questions of regulatory policy. Plans, organizes, and manages a research program directed toward improving the effectiveness of the NRC regulatory process (i.e., the process used in reaching and implementing decisions on regulatory issues) through the use of value/impact analyses and PRA. Monitors and analyses administrative, judicial, and legislative developments that could affect NRC regulatory and research programs. Develops, documents, and implements policies and procedures for developing regulations, including preparation of a regulatory analysis on the impact of a proposed regulatory activity handling of petitions for rulemaking, and RES interactions with the CRGR. Prepares standards, guides and regulations for the production and use of industrial, commercial and consumer products containing regulated quantities of radioactive materials. Develops and implements an agency-wide technology transfer program to train a cadre of PRA practitioners capable of evaluating PRA submittals and applying PRA techniques to regulatory problems, apprise NRC management and selected staff of PRA results having an impact on the regulatory process, and expand the use of PRA technology to support the NRC safety goal.

DIVISION OF RADIATION PROGRAMS AND EARTH SCIENCES

Plans and directs research and standards programs in the earth sciences related to the performance of nuclear facilities in the management of radioactive wastes, natural phenomena, siting factors affecting public health and safety, and health effects and occupational protection from ionizing radiation. Establishes or recommends policy, planning, and procedures for the research and standards programs as required to carry out the functions of the Division and coordinates these research and standards programs with other NRC offices to ensure that the programs are responsive to their needs. Provides technical assistance within NRC on the resolution of generic issues and specific safety problems; provides funding guidance to NRC contractors, DOE laboratories, and other government agencies within the Division budget and consistent with NRC Policy; and maintains liaison with and provides technical input to other Federal Agencies, international and domestic agencies, and other organizations in assigned areas.

Waste Management Branch

Develops and manages research and standards programs to provide regulations, criteria and guides and manages contractual work relative to the overall performance of nuclear facilities and the management of radioactive waste. This involves research to improve understanding of the factors and phenomena resulting from routine licensed operations that significantly affect the public health, accidental releases of radioactive material, and waste facility system performance. These factors and phenomena include external events that affect facility safety; institutional and physical factors that affect the consequences of routine operations and accidents; and the operating, engineering, and system performance factors that affect waste isolation and containment. The branch develops standards and guides and recommends regulations in the areas of its responsibility. Carrying out these functions requires extensive coordination with other NRC Divisions, DOE, EPA, and other Federal and State agencies.