NUREG-0900
Rev. 1

Nuclear Power Plant Severe Accident Research Plan

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

G. P. Marino, Editor

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FOREWORD

This research plan is being issued (March 1986) at a time when major developments in the severe accident arena are occurring. The NRC staff report on reassessment of the science, NUREG-0956, has been issued for public comment and is in the final revision stage. A second NRC staff report, NUREG-1150, is in preparation; it will document application studies for six reference plants. The NRR implementation plan has been tendered to the Commission for review. These works notwithstanding, the NRC Office of Nuclear Regulatory Research is documenting in this report, Revision 1 to NUREG-0900, its continuing plans for research. Considerable specificity is offered for FY 1986 and FY 1987, and we anticipate (see Chapter 3) some additional research in FY 1988 and beyond.

This is not to say that the Agency can not close on issues related to severe accidents in advance of this planned research. To the contrary, we believe that the studies to be embodied in NUREG-1150 will provide the Office of Nuclear Reactor Regulation (NRR) with the audit function which they need to close issues on operating plants. Additionally, the six-plant study, and follow-on studies, should serve as the basis for generic action in areas such as revisions to emergency planning requirements, revisions to 10 CFR Part 100, and so on.

What we are saying in the issuance of this report is that while the regulator may have enough information from our completed studies to act, we are mindful of the need to continue research. This continued research will serve to quantify (and hopefully diminish) the rather large uncertainties which exist as of this writing.

Eventually, our goal is one of "truth in severe accidents" wherein we are satisfied the course of events and the uncertainty are sufficiently quantified.

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ACKNOWLEDGMENTS

Chapters 1, 2, and the Appendix were written by Dr. G. P. Marino with some revisions and additions by the RES staff. Chapter 3 was originally developed by BCL based on RES staff input and later revised by the RES staff. Chapters 4 and 5 were written by Dr's R. O. Meyer and C. N. Kelber, respectively. Review by the Division of Safety Review and Oversight (NRR) is appreciated.

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1 INTRODUCTION

Subsequent to the Three Mile Island Unit 2 accident, recommendations were made by a number of review committees (Refs. 1 and 2) to consider regulatory changes which would provide better protection of the public from severe accidents. Over the past six years a major research effort has been underway by the NRC to develop an improved understanding of severe accidents and to provide a technical basis to support regulatory decisions. The purpose of this report is to describe current plans for the completion and extension of this research in support of ongoing regulatory actions in this area.

1.1 Background

Operating reactors have been licensed, in accordance with 10 CFR 100.11, by their demonstrated ability to accommodate conditions which would hypothetically be produced in Design Basis Accidents without exceeding regulatory criteria (e.g., 25 rem at the site boundary). Although the prescriptions for the Design Basis Accidents that are analyzed in Safety Analysis Reports have some features of severe accidents (in particular, an assumed release of fission products to the containment from a severely-damaged core as developed in report TID-14844 (Ref. 3)), these accidents were not considered realistic characterizations of severe accidents. For example, the loads on the containment that could be produced in severe accidents can exceed those obtained in the analysis of Design Basis Accidents.

Prior to the TMI-2 accident the vast majority of reactor safety research funding was committed to confirming the ability of engineered safety features to perform under Design Basis Accident conditions in a manner which would satisfy regulatory criteria and prevent the escalation of transients or minor accidents into severe accidents. The various committees that reviewed the TMI-2 accident recognized that the regulatory approach taken in the past might not have provided the desired level of protection of the public from severe accidents. As a result of the recommendations made by these committees the NRC has taken the following actions:

A. A major research program was initiated (entitled the Severe Accident Research Program - SARP) to develop a better understanding of severe accident phenomena. The program was described in NUREG-0900 (Ref. 4) issued in 1983. From the beginning, the program was coordinated by senior management personnel from both the Office of Research and the Office of Nuclear Reactor Regulation.

B. The TMI Action Plan (Ref. 5) was issued to describe regulatory activities required to provide added protection against severe accidents. Among such regulatory actions, one hydrogen rule has subsequently been adopted to provide added protection against hydrogen burning loads during degraded core accidents and a second rule has been issued for comment.

C. In 1980, the NRC issued an advance notice of an intent to undertake a Degraded Core Cooling Rulemaking which would identify changes required in existing plants to accommodate severe accidents. That advance noticde of proposed rulemaking was withdrawn in 1985 with publication of the Severe Accident Policy Statement discussed below in Paragraph E.

D. In 1983 the NRC established an Accident Source Term Program Office (ASTPO) to develop more realistic source terms for severe accidents and to determine how the current treatment of source terms in the regulations should be modified. The major activities directed by ASTPO have been integrated with the severe accident evaluations that are being made for the six reference plants which will provide a basis to initiate the development of the regulatory implementation program for severe accidents in FY 1986. As stated above the full risk profile of each of these plants will be completed in FY 1986.

E. The Commission issued the Severe Accident Policy Statement (SAPS) in July 1985. It describes NRC's intentions for resolving safety issues related to severe accident issues. The NRC staff published concurrently with the SAPS a report (Ref. 6) that provides perspective on the development and implementation of the policy and describes the relationship of the SARP. This report (NUREG-1070) describes a strategy for which the following four regulatory questions need to be addressed:

   1. How safe are the existing plants with respect to severe accidents?

   2. How can the level of protection for severe accidents be increased?

   3. What additional research or information is needed?

   4. Is additional protection for severe accidents needed or desirable?

F. The NRC Offices of Regulation and Research (NRR and RES) have established a technical liaison with industrial groups such as IDCOR (Industrial Degraded Core Rulemaking group) to define and resolve complex technical issues associated with severe accident phenomena.

In view of the above actions, and the results obtained to date from them, it has become necessary to redefine and refocus the SARP as originally reported in NUREG-0900. It is the purpose of this report to serve as a revision to the original NUREG-0900 document and to reflect the required redirection in scope dictated by recent efforts in the field.

Prior to the general discussion of the SARP, it is important to define what is meant by "uncertainty" with respect to risk analyses and to define the major technical issues encountered to date. The following section is an attempt to condense these complicated subjects into a format that is easy to follow and is generic in nature. The interested reader can find much more detail on these subjects in Section 3 and Appendix A.

1.2 Uncertainty Evaluation and Major Technical Issues

1.2.1 Uncertainty Evaluation

In order to use PRA results more effectively in support of severe accident decisionmaking, the principal uncertainties associated with risk estimates should be identified and described quantitatively. In addition, there are many situations wherein uncertainties are too large to permit decisionmakers to distinguished among alternative actions. Therefore, the basic purposes of the SARP research are to characterize uncertainty and, to the extent possible, develop experimental and analytical programs to reduce uncertainty.

The first estimate of the potential release of radioactive fission products from Light Water Reactor power plants was made in 1957 (AEC report WASH-740). Other estimates followed, including TID-14844 (1962), the Reactor Safety Study - WASH-1400 (1975), and NUREG-0772 in 1981. The latter study did not actually estimate the source term, but provided a review of the state of knowledge at that time. It also led to the current NRC and industry (IDCOR) initiatives on estimating the source term as embodied in the BMI-2104 documentation (NRC-sponsored) and the final IDCOR document "Nuclear Power Plant Response to Severe Accidents." All of these attempts have suffered from a common critique: i.e., the lack of well-defined, statistically defensible, uncertainty bounds on the final results. Even a cursory study of the numerous calculations, assumptions, models, sequences, plant types, etc., involved in such analyses will convince the harshest critic that putting such bounds on source term and risk estimates is a very difficult task: ergo the lack of complete studies to date. In this Section an attempt will be made to elucidate in some detail the requirements and necessary limitations of such a study.

An initial attempt at an uncertainty analysis of the BMI-2104 results was recently completed by the Sandia National Laboratory (SNL) for the NRC. The program was entitled "Quantitative Uncertainty Estimate of the Source Term" (QUEST) and was necessarily limited (by time and money) to two plants: Surry and Grand Gulf (Ref. 7). Only two sequences were analyzed for Surry (TMLB' and S2D) and one for Grand Gulf (TC). For these limited cases, the QUEST study showed the following:

1. There are very large uncertainties in current source term predictions and, therefore, source term ranges (rather than specific values) should be considered in severe accident analyses.

2. Even with such large uncertainties, the "envelope" of the source terms for some cases fall well below the values quoted in previous reports such as WASH-1400. However, larger values are possible for early containment failure cases.

3. At least a factor of ten decrease in source terms can be expected if containment integrity can be maintained for some hours (> 10) after core melt begins.

4. Fission products such as cesium and iodine may be important for only a short time after core melt because other radionuclides such as tellurium, strontium, barium, lanthanum, etc. dominate at later times.

1.2.1.1 Definition of a Source Term

An overall risk profile for a given LWR reactor requires much more than a source term calculation for a given sequence. It begins with an an extensive study of all those initiating events (combined with a failure of a major system) that may lead to fission product release and/or core melt. For example, the much-studied TC event in BWR systems (also known as an ATWS event--Anticipated Transient Without Scram) consists of an initial transient which requires the reactor to scram. The initial transient has an occurrence frequency of about six per reactor per year. If this rather high frequency event is accompanied by a failure of the reactor scram system to function (a very low probability event - about 0.00001 per demand), the TC sequence is begun. Once begun, there are many possible paths through which the reactor system and its operators may traverse. Some of these paths may lead to complete recovery of the reactor with no damage or consequences whatever, whereas others may lead to severe core damage and extensive fission product release. The determination of this path structure is called sequence analysis and results in the so-called "Event Tree" for this sequence. Each path through the event tree has an associated probability which depends on "Fault Tree Analyses" of the reliability of the systems functioning or not functioning throughout a given path. Thus the TC sequence may have 20-30 possible outcomes each of which will have a source term associated with it. Most of these paths will be successful (i.e., complete recovery and no fission product release). Those paths with non-trivial source terms are combined by multiplying the probability of the specific path by the calculated consequences for the computed source term for that path and summing over all the paths. The result is the risk for the TC sequence for that plant. A risk profile for the plant is produced by summing the risks of all the significant sequences for that plant (TQUV, TW, etc. for a BWR plant). The latter is also known as a PRA--Probability Risk Assessement. The WASH-1400 study is a PRA for only two plants--Peach Bottom (BWR) and Surry (PWR). The BMI-2104 study did not constitute a PRA for any plant since it did not include all significant sequences nor all possible paths within each. However, the follow-on effort noted above (NUREG-1150) will complete the risk profiles for all the reference plants included in the study. The BMI-2104 study does contain source term estimates for one or two paths of the risk-significant sequences for the five reference plants studied.

It should now be clear that the uncertainty in a source term calculation is only a subset of the uncertainty in the risk of a plant. The latter includes uncertainties in the event trees, fault trees, reliability estimates and consequences as well as the individual source terms. Moreover, each uncertainty will be plant, sequence, path, and site dependent. The overall uncertainty in the risk of LWR power plants will be a combination of the above.

As seen from the above discussion, a "source term" calculation should be made for each and every path through an event tree sequence that leads to fission product release to the environment. The key word is "environment", because even a severely damaged reactor vessel may be surrounded by an intact leak-proof containment resulting in minimal or zero risk to the public. Most source term calculations are, therefore, very dependent on the mode, timing, and probability of containment failure. Thus, the sequence analysis described above must include an event tree for the containment behavior as well as one for the reactor system response.

Since the reactor system event tree path and the calculated response of the reactor system directly affect the loads on the containment, the event tree for the containment is not independant of either the reactor system tree or the source term calculation itself. What is usually done, therefore, is to compute the source term to (and the loads on) the containment for a given reactor sequence path. Once these are known as a function of time, they are combined and used in the containment event tree analysis to determine the source term to the environment. Sometimes, as in the BMI-2104 analyses where no containment event tree analyses were available, a containment failure time is assumed based on engineering judgement or independent calculations for specific sequential paths.

A basic requirement of any uncertainty analysis is the existence of a calculational methodology such as a set of computer models which allow for the computation of fission product release, transport, deposition, and location throughout the time span of a postulated accident sequence. In the past, no single code has been available for all the calculations required, so certain codes were used in series (i.e., allowing for no feedback between models) to provide the information necessary. Thus, the WASH-1400 study used the BOIL code for plant and core behavior, the CORRAL code for containment processes, hand calculations, and other codes as needed. The more extensive BMI-2104 study used an updated MARCH code for in-vessel core behavior, the MERGE code for in-vessel thermal hydraulics calculations, the TRAP/MELT code for fission product transport and deposition in the vessel, the CORSOR code for in-vessel fission product release from the fuel, the CORCON code for molten-core/concrete interactions, the VANESA code for fission product release during the core/concrete ex-vessel interaction, and the NAUA code for fission product behavior in the containment. In addition, depending on the reactor type, the SPARC code was used for suppression pool scrubbing of fission products and aerosols in BWR's, and the ICEDF code was used for fission product and aerosol attenuation in ice-condenser plants. Given the existence of an integrated code package, it is crucial to have a procedure in place for identifying and examining the possible ranges of parameters and assessing their significance. The boundary conditions on the analysis and assumptions made may be the largest contribution to the uncertainty in the results.

All analytical treatments of physical phenomena require the input of certain physical parameters such as diffusion coefficients, heats of vaporization, oxidation rate constants, heats of chemical reaction, etc. which have usually been obtained from experimental studies. All such parameters have uncertainties associated with them due to data scatter, extrapolation of values outside the range of known values, experimental error, and data analysis methods. These kinds of uncertainties are known as input uncertainties. Since source term analyses may involve thousands of such parameters, and since a statistically valid probability density function for each parameter requires large samples of measurements at each temperature, pressure, and environmental condition, it is necessary that source term analysts make some compromises in this respect. The compromise usually takes the form of using existing data, bounding the scatter with 95 percent confidence limits and extrapolating to other conditions of pressure and temperature when such dependences are known. Fortunately, the sensitivity of the source term to this type of material parameter is relatively small, as is evidenced by the QUEST results.

Another type of uncertainty that must be assessed is caused by the modelling of the pertinent phenomena. This type of uncertainty is by far the most important and the most difficult to determine. Conscientious engineers and scientists can and do make convincing arguments as to the validity of their models even if it gives almost opposite results to that of an equally reasonable model. In the absence of very specific, very conditional data, it is difficult to put reasonable bounds on the resulting output. Moreover, the output of the model may be used as input to another model wherein it may be the most important parameter affecting the output of the new model. In this way, the large uncertainty is propagated and even expanded throughout the entire analysis procedure.

As an example, consider as a parameter the amount of unoxidized Zircaloy present in the melted core just prior to vessel failure. The QUEST study showed that it can have a dramatic effect on the ex-vessel release of fission products via its role as an input parameter to the CORCON code and, depending on the containment failure time, a corresponding large effect on the resulting source term to the environment. Moreover, it can also dramatically affect the timing and probability of containment failure, thereby enhancing its importance by altering the sequence of critical events along the sequential path of the accident. Although the input constants used to model the oxidation rate of Zircaloy are well known and agreed upon by most analysts, the phenomenology used to determine the melt progression characteristics and the corresponding available surface area of Zircaloy able to react with steam is hotly debated. Some recent comparisons between IDCOR and BMI-2104 results for the TC sequence in Peach Bottom show factors of fifty or more difference in the fraction of Zircaloy oxidized in-vessel. This can yield an order of magnitude or more difference in the final estimated source term.

Inasmuch as there will never be enough data on this effect to even begin to determine a statistically defensible probability density function, source term analysts must rely on sound engineering and scientific judgements to define a proper range of uncertainty. By using new models and forming a study group of experts to analyze the results, it should be possible to define a reasonable range of uncertainties for these types of phenomena.

1.2.1.2 Definition of a Risk Profile

Given the set of individual uncertainties discussed above, a method is selected by which they are combined systematically to yield a range of values of a given output variable. Two major methods are available. Latin Hypercube Sampling or Response Surface Methodology. Both methods yield response variables with error bounds that represent the cumulative effect of consistently varying each of the individual uncertainties throughout their ranges. In addition, the methods also provide a determination of the major contributors to the overall uncertainty of the response variable. Specification of a subjective probability distribution (i.e., parameter ranges and degrees of belief) is crucial. Since the Latin Hypercube sampling method usually requires less computer runs and allows more judicious sampling of input ranges, it was chosen for the QUEST study and the MELCOR code.

In the staff's report on plant risk, NUREG-1150, uncertainties will be treated explicitly, compared to the past analyses which dealt with selected facets of risk evaluations. In these earlier analyses, the uncertainty was discussed in qualitative terms, if at all. NUREG-1150 will draw heavily on the QUEST results for source term uncertainties. Also included will be a more detailed analysis of source term uncertainties of a few selected sequences for comparison purposes.

Traditionally, a quantitative uncertainty analysis procedure has been developed for the severe accident frequencies. There, data sources are available which allow for probabilistic statements on core damage frequency.

In NUREG-1150, uncertainties in accident sequence frequencies, containment response, source terms, and ex-plant effects will be combined to produce a measure of overall uncertainty in risk. Engineering ranges (vis-a-vis classical statistical intervals, confidence bounds, etc.) are being developed for major parameters within each of these portions of the risk estimation method (typically 15 parameters in toto), subjective weights on the values within each range are being developed, and the ranges/weights are being combined using Latin Hypercube sampling techniques. The direct result of this combination is an engineering approximation of the range of uncertainty in risk. In addition, the relative contribution of each parameter's uncertainty to the overall uncertainty is being extracted and documented.

1.2.2 Major Technical Issues

The major technical issues cited above are listed in Table 1.1. These issues are a result of several years of NRC/IDCOR staff technical meetings and discussions. They very strongly correlate with issues defined by the American Physical Society (APS) review of the BMI-2104 documentation sponsored by ASTPO. Therefore, as shown in Table 1.1, the issue definition from the NRC/IDCOR staff interactions encompass all the APS and NRC issues defined in NUREG-0956. A more detailed discussion of these issues is presented in Appendix A and within Sections 2 and 3 of this report where the research programs discussed are related directly to their corresponding technical issue. It should be noted that some of these issues are being specifically studied as part of the NUREG-1150 risk analysis. In this report, quantitative assessments of the risk importance of these issues will be provided.

Table 1.1

Summary of Major NRC/IDCOR Technical Issues for Severe Accidents

I CORE HEATUP STAGE

ISSUE #1	-	FISSION PRODUCT RELEASE PRIOR TO VESSEL FAILURE
ISSUE #2	-	RECIRCULATION OF COOLANT IN THE REACTOR VESSEL
ISSUE #3	-	RELEASE MODEL FOR CONTROL ROD MATERIALS
ISSUE #4	-	MODEL FOR FISSION PRODUCT & AEROSOL DEPOSITION IN THE PRIMARY SYSTEM

II MELT PROGRESSION AND FUEL RELOCATION STAGE

ISSUE #5	-	MODELING OF IN-VESSEL H_2 GENERATION
ISSUE #6	-	CORE SLUMP, CORE COLLAPSE, AND REACTOR VESSEL FAILURE MODELS
ISSUE #7	-	ALPHA MODE CONTAINMENT FAILURE BY IN-VESSEL STEAM EXPLOSIONS

III EX-VESSEL STAGE

ISSUE #8	-	DIRECT HEATING OF CONTAINMENT BY EJECTED CORE MATERIAL
ISSUE #9	-	EX-VESSEL FISSION PRODUCT RELEASE MODELING
ISSUE #10	-	EX-VESSEL HEAT TRANSFER MODELS FROM MOLTEN CORE TO CONCRETE/CONTAINMENT
ISSUE #11	-	REVAPORIZATION OF FISSION PRODUCTS IN UPPER PLENUM
ISSUE #12	-	DEPOSITION MODEL FOR FISSION PRODUCTS IN CONTAINMENT
ISSUE #13	-	AMOUNT AND TIMING OF SUPPRESSION POOL BYPASS
ISSUE #14	-	MODELING OF EMERGENCY RESPONSE
ISSUE #15	-	CONTAINMENT PERFORMANCE
ISSUE #16	-	SECONDARY CONTAINMENT PERFORMANCE
ISSUE #17	-	HYDROGEN IGNITION AND BURNING
ISSUE #18	-	ESSENTIAL EQUIPMENT PERFORMANCE

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6. "NRC Policy on Future Reactor Designs: Decision on Severe Accident Issues in Nuclear Power Plant Regulation," NUREG-1070, July 1985.

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2 GENERAL DESCRIPTION OF THE SARP AND TECHNICAL ISSUES

The NRC and its predecessor, the AEC, have licensed many nuclear reactors with a safety analysis method that does not explicitly deal with the threat of core melt accidents except by the requirement that there be a robust containment around the reactor, and the conduct of the simple 10 CFR Part 100 calculation to measure the effectiveness of that containment. The Three Mile Island accident led to a call for more explicit evaluations of the threat of severe accidents; i.e., those entailing severe core damage or core melt. The SARP and consequent severe accident regulatory decisions are the answers to that call. The SARP is producing a substantially greater data base to understand and predict severe accident behavior as well as generating plant-specific severe accident assessments upon which to base regulatory decisions.

The NRC severe accident research program was planned shortly after the accident at the Three Mile Island Unit 2 power station in March, 1979. Major research experiments did not actually begin until late FY 82 since lead times for such complex efforts are considerable. The program plan was documented and published in January 1983 as NUREG-0900 "Nuclear Power Plant Severe Accident Research Plan." The report listed thirteen separate elements of research to be completed under the plan. Although most of the planned work has been completed and much of it has been redirected due to changing needs, the document still serves a purpose as a general guideline.

In order to simplify the discussion of the remaining research to be accomplished over fiscal years 1986 and 1987, the original elements have been be combined initially into three general phenomenological areas and expanded into six sub-areas for general discussion purposes. A detailed discussion of the individual programs will be presented in Section 3. Moreover, major deliverables such as code packages will be discussed separately in Section 3 and as an integrated package in Section 4.

Figure 2.1 gives a general representation of how severe accident research is structured to provide NRR and I&E with the necessary tools for their mission. Figure 2.2 shows the information sources available to NRR or I&E to implement the Severe Accident Policy Statement. Note the three general areas of research depicted at the bottom of the figure. Note also that considerable information is already available from the SARP, ASTPO and IDCOR work efforts to date. Each of the three phenomenological areas of research will be discussed briefly below and in detail in Section 3.

2.1 Risk Evaluation and Sequence Analysis Research

This area of SARP is divided into two major sub-areas which address the frequencies of severe accidents in LWR's and the sequence definitions for those accidents. They are (a) Risk Reduction and Evaluation and (b) Severe Accident Sequence Analysis (SASA).

GENERAL S.A.R.P. APPROACH FOR PROVIDING INFORMATION TO NRR

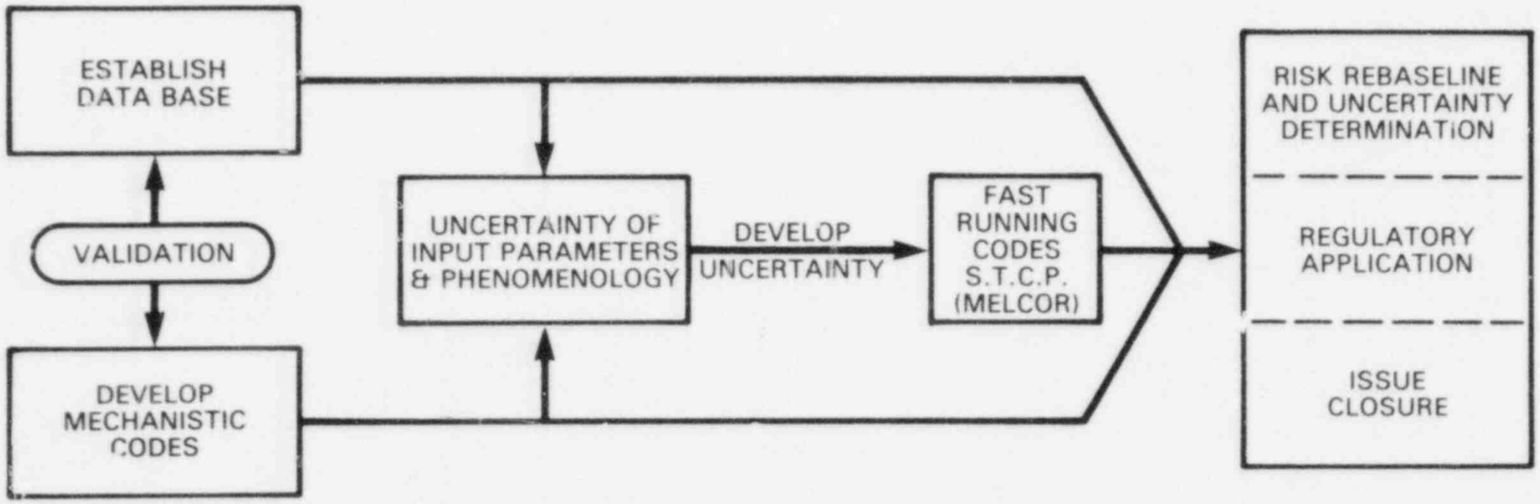


Figure 2.1

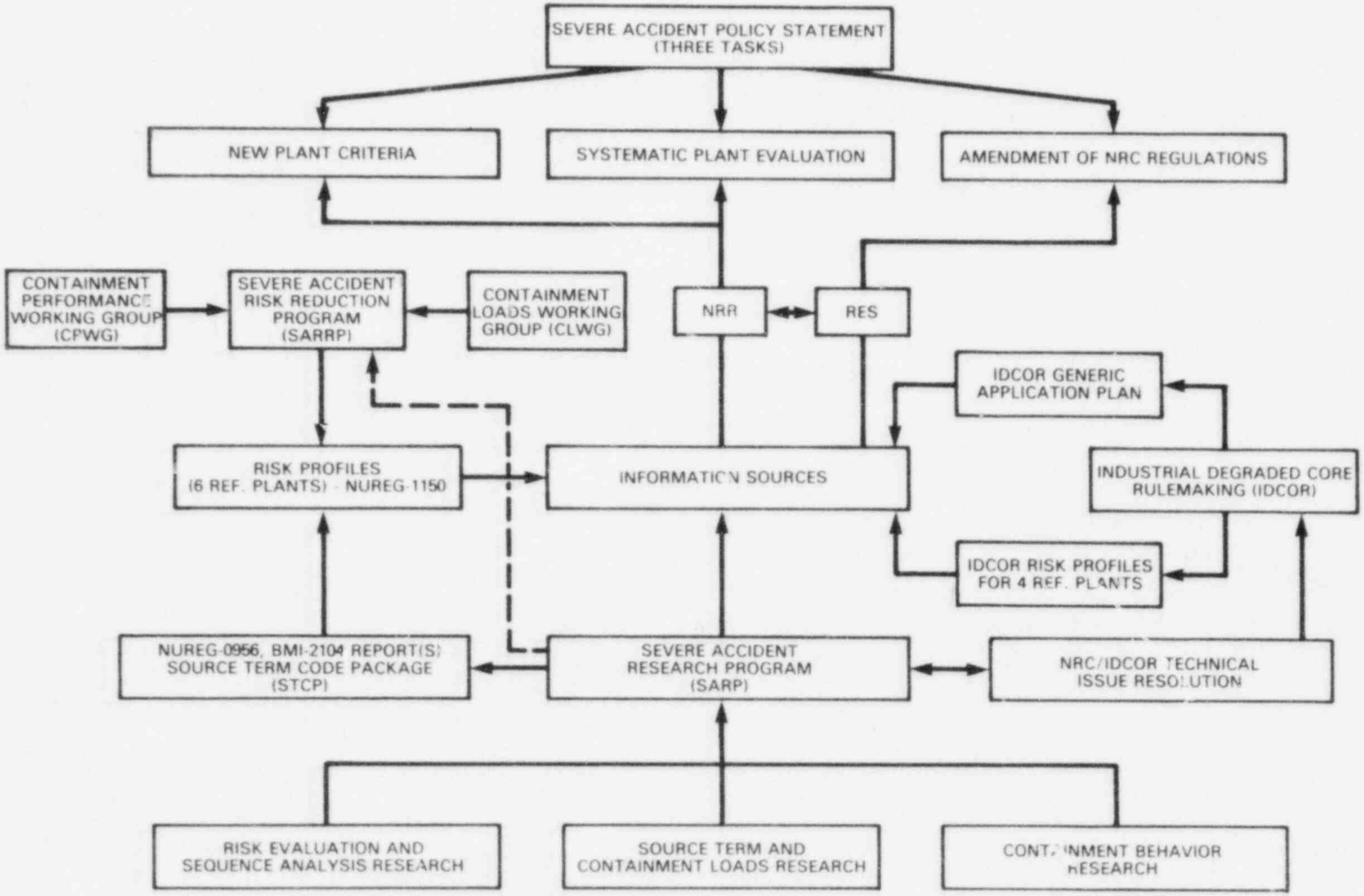


Figure 2.2 Information Sources

2.1.1 Risk Reduction and Evaluation Research

This sub-area consists of programs related to determining the sequence event trees and probabilities for the reactor system and the containment, the consequence models for a given sequence and source term, and the development of the MELCOR computer code. The latter program is the long-term replacement for the Source Term Code Package (STCP) developed by ASTPO and discussed in the ASTPO report NUREG-0956. The research in this sub-area is conducted by the Division of Risk Analysis and Operations (DRAO), and is depicted as the left branch of Figure 2.3.

The principal thrust of the severe accident risk research program in FY 1985 has been directed toward the preparation of an integrated risk assessment report characterizing the risk at six operating plants covering the basic containment types employed today. The plants selected for this study are Surry, Peach Bottom, LaSalle, Grand Gulf, Sequoyah, and Zion (representing subatmospheric, Mark I, Mark II, Mark III, ice condenser and large dry containments, respectively). Such a report is needed to provide the NRC with updated risk information for typical reactor designs; to provide examples and insights on the use of such information could be used to enhance regulatory decisionmaking, considering the large uncertainties inherent in risk information; and to support NRR's efforts to assess the plant evaluation methodology developed by IDCOR. This effort will utilize source term calculation methods set forth in NUREG-0956, Reassessment of the Technical Basis for Estimating Source Terms. This effort will continue through FY 1986, and work is planned by FY 1987 to extend the reference plant analyses to a B&W and a CE plant and to evaluate, as needed, residual risk-important issues and new research results.

Major programs conducted by DRAO include: (1) The Accident Sequence and Evaluation Program (ASEP); (2) the performance of additional STCP runs; (3) the Severe Accident Risk Reduction and Rebaselining Program (SARRP); (4) the MELCOR code development program; (5) consequence modeling; and (6) statistical and uncertainty analysis program. ASEP provides a rebaselining of accident sequence likelihoods for the six reference plants as well as generic accident sequence information for all operating and near operating LWRs. ASEP research consists of compiling and analyzing current PRA findings, operating experience, and post-TMI fixes.

Risk profiles for the reference plants are being developed based on state-of-the-art knowledge of accident frequencies, containment behavior in severe accidents, and radionuclide source terms. Although a number of sequence analyses were performed with the BMI-2104 code suite to support the Source Term Reassessment Study, the coverage of sequences was not adequate to support the characterization of risk for SARRP. In order to identify the minimum set of source term analyses required, a binning assessment was made for each plant and additional source term calculations are subsequently to be made at BCL using an integrated version of the suite of source term codes called the Source Term Code Package (STCP).

The SARRP work is an integrating effort conducted by the Sandia National Laboratories for the NRC which will provide a major portion of the technical basis for the Severe Accident Regulatory Implementation Program. Within its workscope, SARRP will develop the detailed containment event trees, assure coordination with the accident sequence, consequence, and uncertainty analyses, and evaluate

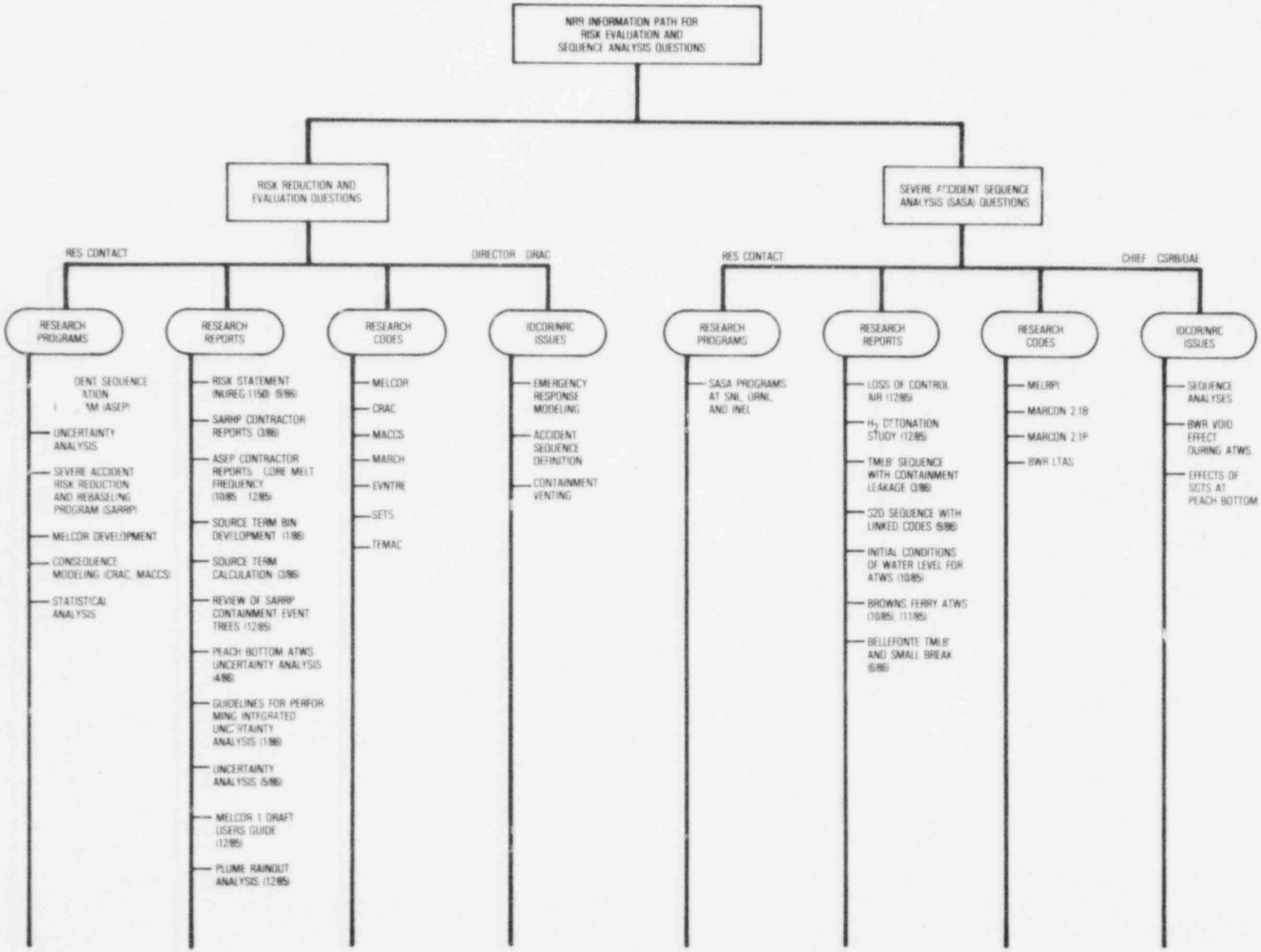


Figure 2.3 Risk evaluation and sequence analysis research

the risk to the public for the six reference plants. The change in risk associated with proposed modifications in the design and operation of each plant will also be addressed. Cost-benefit analyses will be performed to determine if the cost of the proposed modification is warranted by the reduction in risk.

As part of the SARRP work, the uncertainty in risk, and the major contributors to this uncertainty, are being evaluated. The principal method involves the systematic combination of 10-20 parameters judged to be the most important to this uncertainty. An advanced simulation technique (Latin-Hypercube sampling) is being used for this study. To check the adequacy of the use of a limited number of parameters, a more detailed uncertainty/sensitivity analysis (i.e., using more parameters) is being performed in parallel for one reference plant (Peach Bottom).

The new consequence model, MACCS, incorporates explicit modeling of time dependent releases, adds additional exposure pathways, and contains improved dosimetry and health effects models. In addition, it is designed to accommodate a variety of graded emergency responses and considers road network evacuation.

As stated above, the MELCOR code package is designed to replace the current STCP for risk analyses. However, it has many additional features. It is designed to predict the physical and chemical processes of core meltdown sequences, containment loads and performance, the release and transport of fission products, and the resulting off-site consequences through its integration with the consequence analysis program (MACCS). All the above efforts will include comprehensive uncertainty and statistical analyses to provide estimates of the confidence limits to be imposed on the final results.

During the last half of FY 1985, a substantial reorientation of ongoing programs in risk assessment research was effected to support the preparation of this report, identified as NUREG-1150. As presently conceived, NUREG-1150 is intended to present:

A. An assessment of core melt frequency and risks associated with each of the reference plants.

B. Perspectives on the elements driving risk-dominant accident sequences.

C. A characterization of related uncertainties (qualitative and quantitative) in the assessments.

D. The results of the evaluation of the effectiveness of several risk-reduction features for the reference plants.

E. Discussions of insights and perspectives gained from plant-specific studies, compared to insights gained from more generic plant models and data bases (such as existing in the Accident Sequence Evaluation Program (ASEP).

F. Insights on extrapolation of the methodology to assess the risks for plants without PRAs.

G. Recommendations for future regulatory utilization of reliability and risk information.

This report is scheduled to be published at the end of FY 1986. The draft report will include only the six reference plants. The B&W and CE plant analyses will be published subsequently.

Risk Assessment Report: Program Detail - The Projects supporting the preparation of NUREG-1150 have been divided into the six principal areas which are described below. Detailed project summaries for the major projects are provided in Section 3.1 of this report. The six principal program areas and related FIN's are as follows:

1. Accident Frequency Rebaselining - Updating and recalculating the accident frequencies of the six reference plant PRA's, using the most up-to-date information (FIN's A1228; A6301; B0445; A7225).

   a. Develop event trees for the dominant accident sequences drawing heavily upon the ASEP modeling and data base.

   b. Perform plant visits (except for Zion) to update plant design and procedures. This effort includes an augmented human reliability assessment of the ATWS sequence at Peach Bottom, the principal contributor to risk for Mark I designs.

   c. Provide support from two data programs to improve the plant specific data base used for five of the reference plant analyses.

   d. Complete the accident sequence analyses using the results of the visits, insights from all existing PRA's and up-to-date PRA methodology (to the extent practical).

   e. Provide extensive QA and QC efforts on the above analyses.

2. Containment Analysis - Development and analysis of containment event trees to determine containment failure probabilities for each reference plant (FIN's A1322, G1084; A3293). This work consists of the following basic elements:

   a. Development of questions appropriate for determining how the accident will progress and threaten containment integrity.

   b. Development of containment event trees using the above questions and optimistic, central, and pessimistic judgments regarding the answer to each question and the likelihood of each answer.

   c. Calculation of the likelihood of various types of containment failure modes from an optimistic, central, and pessimistic perspective.

   d. Identification of the major factors driving risk, performance of sensitivity analyses of their importance, and characterization of the containment failure likelihoods and uncertainties.

   e. Performance of a QA review of the above work.

3. Source Term and Consequence Analyses - Calculation of accident source terms and consequences for all the risk-important sequences not included in the BMI-2104 report and other sequences subsequently identified in the rebaselining

program (FIN's A1042; A1322; A3290; B7499; D1106; A1339). This work consists of the following basic elements:

a. Perform needed source term code package runs for the risk significant sequences.

b. Perform a QA review of the results of the STCP runs.

c. Develop the MACCS code on an expedited basis to support the reference plant analyses, since MACCS will permit the use of multiple puff releases, new health effects models, and other significant refinements to the CRAC-2 code.

d. Develop the BWR analysis capability for MELCOR on an expedited basis to support uncertainty analyses for the Peach Bottom ATWS sequence and subsequently for additional source term and consequence analyses.

e. Develop the PWR capability for MELCOR to support source term, risk, and uncertainty analyses subsequent to publication of the draft NUREG-1150.

4. Risk Reduction Analysis - Execution of value-impact analysis (risk reduction effectiveness) of a number of plant modifications using updated risk and cost data (FIN's A1322; A6842; A3293). This work consists of the following basic elements:

a. Assessment of baseline risk using the above work on accident sequences, containment event trees, source terms, and consequence codes.

b. Identification of risk reduction options, including those generic items identified in NUREG-0900, as well as potential plant specific options identified in the reference plant analyses.

c. Performance of an in-depth procedures and engineering analysis of the benefits of venting for Peach Bottom.

d. Performance of a risk reduction and value-impact analysis, considering important uncertainties.

5. Uncertainty Analysis - Provide estimates (qualitative and quantitative) of the uncertainties in accident frequency and consequences (FIN's A1393; A1322). This work consists of the following basic elements:

a. Perform limited uncertainty analyses using the results of a finite number of sensitivity analyses for all six of the reference plants to develop reasonable bounds on risk and core melt frequency.

b. Perform an integrated and comprehensive uncertainty analyses for the Peach Bottom ATWS sequence using MELCOR.

c. Perform uncertainty analyses on a few accident sequences using the STCP.

6. Preparation of NUREG-1150 - Provide contractor support for writing various technical sections of appendices of NUREG-1150 (FIN's D1106; A1322). This work consists of the following basic elements:

a. Performance of limited additional technical analyses on important risk parameters.

b. Preparation of technical portions of the draft NUREG and supporting technical appendices.

c. Administrative support for advisory review meetings.

d. Conduct of a peer review workshop.

Note: The bulk of NUREG-1150, including all conclusions and recommendations, will be prepared by DRAO.

Program Schedule and Major Milestones

The following is the broad chronology and schedule which has been adopted for the preparation of NUREG-1150:

Decision to Expand Program	Late May 1985
First Plant Visits (Surry, Peach Bottom)	July 1985
Complete Source Term Runs	Early 1986
Complete Refinement of all Accident Sequences	Early 1986
Final Drafts of Contractor Risk Reports	Spring and Summer 1986
Draft NUREG-1150 for Public Comment & Peer Review	End of FY 1986
Peer Review Underway	Early FY 1987
Final NUREG-1150 Issued	Spring 1987
B&W and CE Plants and Future Work as Needed	CY 1987

It is important to recognize that these proposed milestones are both ambitious and optimistic. They are ambitious because of the extensive new analyses required and optimistic because the schedule assumes no problems with code package linkage or code runs, timely access to all plant information, that no additional source term runs will be needed based on information gained from containment event tree and accident analyses, and that substantive changes in these risk assessments will not be needed to respond to the comments of peer review groups.

2.1.2 Severe Accident Sequence Analysis (SASA) Research

The second sub-area of research under this general phenomenological heading is conducted by the Division of Accident Evaluation (DAE). Detailed studies of dominant accident sequences, identified by the Accident Sequence Evaluation Program (ASEP), are analyzed with best-estimate deterministic codes. Severe accidents, with and without operator actions, are studied to provide an assessment of the potential for management of the accident. The BWR portion of the SASA program identified the source term differences from PWR's and provided the inclusion of the specific BWR severe accidents into the containment loading analyses for the source term study. In addition, the most detailed BWR core models in currently released versions of BWR codes were developed by the ORNL/SASA program. In the PWR area, overall thermal/hydraulic modeling is provided by RELAP5, which is dynamically linked to both SCDAP and TRAP/MELT.

Specific NRR questions on sequence details should be directed towards the programs in this sub-area. Figure 2.3 shows deliverables, codes, and technical issues involved in this research area. (FIN's A1258; A6354; B0452).

2.2 Source Term and Containment Loads Research

As can be seen from Figure 2.4, this general area consists of two sub-areas: namely, In-Vessel Research, and Ex-Vessel/Containment Loads Research.

2.2.1 In-vessel Research

In-vessel research is conducted to ensure the necessarily simplified computer models used in the STCP (and its replacement MELCOR) are correctly modelling all of the significant aspects of the complex phenomena that occur during a severe accident sequence. This research consists primarily of highly focussed experiments which are intended to reduce the major uncertainties in core melt progression phenomena and fission product and aerosol behavior. This research is also the subject of the Joint International Severe Fuel Damage and Source Term Research Program consisting of the NRC, EPRI, and eleven foreign partners. About one quarter of the financial support for the program comes from the foreign partners as well as invaluable in-kind research results from their own programs. The major deliverables from this sub-area (in addition to the documentation of the experimental results) are two highly-detailed computer codes which describe in-vessel core degradation and fission product behavior for all types of accident sequences. For melt progression phenomena, the mechanistic codes are SCDAP/RELAP and MELPROG/TRAC. For fission product release and transport, the detailed codes are VICTORIA, FASTGRASS, and TRAP-MELT.

The SCDAP/RELAP code is a best-estimate reactor system and core model intended to provide the NRC with accurate analysis capability for accidents which involve severe core damage, but do not proceed to vessel melt-through. That is, it is intended for analysis of accidents such as occurred at TMI-2 in March 1979. Because it is to be used for such attenuated sequences, considerable attention is given to detailed modelling of individual fuel rods, fission product release, and fission product deposition throughout the reactor coolant system, and the coolability limits of severely damaged cores upon reflooding. SCDAP has been linked with the RELAP5 thermal/hydraulics code, and it also incorporates the mechanistic FASTGRASS fission product release model and the TRAP-MELT fission product transport methodology. SCDAP is now operational and will be essentially complete by FY 1987. Fiscal 1987 funding will be primarily used for code maintenance, validation, and applications. The code has been and is currently undergoing considerable assessment using the results of the PBF severe fuel damage program (completed in FY 1985) and the ACRR and NRU experimental programs at SNL and BNWL respectively. Moreover, the TMI-2 core examination and the recent LOFT FP2 experiment have been and will continue to be instrumental in its validation. In fact, the code was used very successfully for FP2 pretest predictions of fuel temperatures and test operating specifications. Future use of this code will be in assessing the range of validity of corresponding simplified models in the STCP and MELCOR. It will also be used for highly specific calculations required by the SASA program during accident progression and management studies, and to address specific questions on severe accidents that cannot be adequately answered using the STCP.

The MELPROG/TRAC/VICTORIA code is a best-estimate mechanistic core-melt progression Code package for analysis of accidents which proceed through melt-through of the reactor vessel. It differs from the SCDAP/RELAP code in that considerably less attention is given to detailed fuel rod modelling since rod geometry will be lost prior to melt-through. Detailed calculations, however,

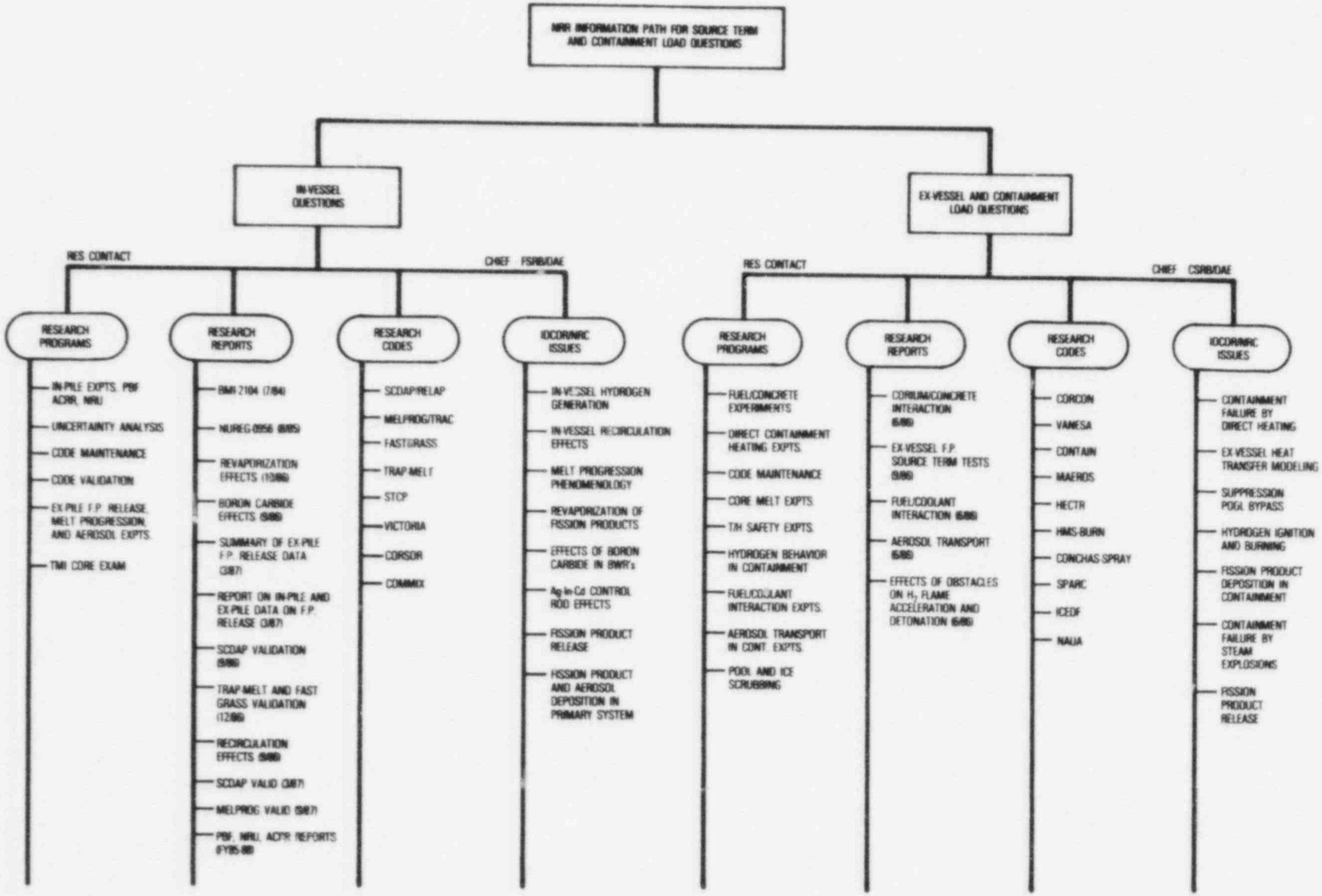


Figure 2.4 Source term and containment loads research

are made for such core melt progression phenomena as the attack on the core-support structure and the reactor vessel, fission product and aerosol release and in-vessel attenuation, core temperature distributions, core debris composition, and the mode of vessel failure and the conditions at vessel failure. This code is important for risk studies because the large uncertainties in melt progression models are major contributors to the overall uncertainty in source term calculations. For example, it has been shown in the QUEST source term uncertainty study that the amount of oxidation of the Zircaloy cladding that occurs during the in-vessel stage of a severe accident can change the resulting source term estimate by several orders of magnitude. Such a large effect is due to the complex chemical interaction of Zircaloy with the fission product tellurium and to the interaction of hydrogen produced by ex-vessel oxidation of the remaining Zircaloy with lanthanum and plutonium oxides during the core/concrete interaction phase of the accident. For some reactor designs and sequences, the hydrogen produced can be a major contributor to early containment failure.

The integrated MELPROG/TRAC/VICTORIA code includes the TRAC in-vessel thermal-hydraulics module and the VICTORIA in-vessel fission-product and aerosol behavior module. A 2-D version of the established TRAC code is used to calculate the strong effects of in-vessel natural convection upon the core-melt progression process. VICTORIA is a theoretically-based mechanistic code, in contrast to the CORSOR data correlation for fission product release used in the STCP. Early versions of the integrated MELPROG/TRAC code and the VICTORIA module are now operational, with the completed versions scheduled for completion in FY 87. The basic data base used for the development of MELPROG has come from experiments largely concerned with the early rod-deformation phase of a severe accident. There are few data currently available for the later debris formation and melt progression phases. Such data for key elements of the code will be obtained from a new program of separate effects experiments and corollary analysis. Because of the large scale of and complex interactions in the in-vessel core-melt progression processes treated by MELPROG, it will not be feasible to perform an integral validation proof test of the code as a whole, although results of the TMI-2 core examination will be of value.

The FASTGRASS code is a mechanistic code developed to predict fission product release from fuel pellets. The code was originally developed to predict noble gas release during design basis accidents but was improved to also predict the release of volatile fission products during the early phase of a severe reactor accident. Specifically, the code can estimate release of noble gases, Cs, I and Te for accident conditions up to and including fuel liquefaction, but will not handle release during melt progression. At present, the FASTGRASS code is linked with SCDAP/RELAP and TRAP-MELT in the so-called SCDAP/RELAP package (see discussion above). The latter is used to analyze terminated accidents such as TMI-2 and experiments at PBF and LOFT, and to benchmark risk codes such as the STCP and MELCOR. Efforts in FY 86 and beyond will focus on code application, validation and maintenance. It is expected that a large quantity of data for FASTGRASS validation will become available in the near future from the out-of-pile separate effects experiments at ORNL and BCL, the in-pile separate effect experiments at ACRR and the in-pile integral experiments at NRU.

The TRAP-MELT code was developed to predict the transport behavior of fission products and aerosols in the reactor coolant system. It has a detailed mechanistic treatment of aerosol behavior in the RCS. The code was used in the

BMI-2104 calculations and is therefore part of the STCP. At present, the code is coupled with the SCDAP/RELAP package. As stated above, this package is being used to analyze the TMI-2 accident as well as experiments at PBF, NRU, and LOFT, and will also be used to benchmark MELCOR. Future activities on TRAP-MELT will be concentrated on code application, validation and maintenance. Results from the ORNL Aerosol Transport Tests and the EPRI LWR Aerosol Containment Experiments will provide the necessary data for TRAP-MELT validation.

The VICTORIA in-vessel fission-product aerosol behavior code has been developed as a module of the MELPROG code and also for stand alone use. VICTORIA is a detailed mechanistic code that uses first-principles modeling of fission-product behavior. Chemical interactions and rate-limiting mass transport processes during release and transport are considered. VICTORIA includes the mechanistic FASTGRASS fission-product release modeling and the detailed TRAP-MELT fission-product and aerosol transport methodology. VICTORIA treats fission-product release and transport during the core-melt progression phase of the accident up to vessel failure. VICTORIA will be validated by current data and by additional new data from the current program of separate-effects out-of-pile experiments at ORNL and BCL and in-pile experiments in ACRR.

It is not the intent of the NRC to use these detailed mechanistic codes directly in risk analyses, but rather to use selected results from them to help reduce the wide range of input selection possibilities currently available in the STCP and MELCOR. However, the mechanistic codes will be used directly for detailed analysis of the most difficult and risk significant cases encountered in risk analysis.

2.2.2 Ex-Vessel/Containment Loads Research

The Ex-Vessel/Containment Loads sub-area is primarily concerned with studies of ex-vessel core and containment interactions. It consists of core/concrete interaction experiments, direct heating of containment experiments, hydrogen behavior, and core-debris/water interaction experiments. (FIN's A1218; A1019; A1198; A1406; A3024; A1246; B0121; A1030). The major deliverables of this work are the CORCON code, the VANESA code and the CONTAIN code. CORCON is a major module of the STCP and is used to model the core/concrete interaction and containment/core energy exchange. The accuracy of this model is critical in the determination of the source term to the containment and to the loading of the containment. It provides the input parameters necessary for the calculation of the ex-vessel release of fission products and aerosols by the VANESA module of the STCP. Most of the FY 1986 and 1987 work in this area will be used to validate CORCON and VANESA.

Another major deliverable of this sub-area is the CONTAIN code. This code is a detailed best-estimate containment behavior code designed for in-depth studies of containment pressure and temperature histories during severe accidents. It is used mainly for assessing SASA studies and the adequacy of containment loading predictions used for containment event trees, as well as aerosol transport and deposition in multicompartment containment geometries. The code should be fully developed by the end of FY 1987 with further validation pending the receipt of the results of the experiments noted above.

The Hydrogen Behavior Program is directed towards providing information to quantify the threat to nuclear power plant containment structures, safety equipment, and the primary system posed by hydrogen combustion. This program consists of transport and combustion code development, flame acceleration and transition from deflagration to detonation (DDT) experimental and modeling efforts, detonation experiments, and experimental and theoretical efforts to determine the chemical and physical effects of hydrogen combustion on cesium iodide-containing aerosols. The major deliverables of this work include the HECTR and HMS-BURN computer codes. HECTR is a lumped volume parameter code for calculating the containment atmosphere pressure and temperature response. In FY 1986, HECTR version 1.5 will be documented and released. HECTR 1.5 will include improved correlations based on data from hydrogen combustion tests and incorporate models for fan coolers (large dry PWR analysis), containment leakage (SASA), and carbon monoxide/carbon dioxide generation. Efforts in FY 1986 will be directed towards code maintenance and in release of HECTR version 2.0. This final version of the code will include a diffusion flame model for the assessment of equipment survival of thermal loads produced by a standing diffusion flame. HMS-BURN is a three-dimensional finite difference transport and combustion code used to benchmark HECTR and to provide a limited number of calculations when more detailed information is needed with respect to transport and mixing or temperature, pressure and heat loads resulting from hydrogen combustion. In FY 1986, HMS-BURN will be used for posttest calculations for licensing review of the BWR Hydrogen Control Owner's Group (HCOG) Quarter Scale facility tests. This code should be fully developed by the end of FY 1987.

Tests of flame acceleration and transition to detonation should be completed in FY 1986. A report will be issued in FY 1986 indicating the effects of transverse venting on flame acceleration and DDT in an obstacle-free channel. In FY 1987, a final report will be published discussing the ability of the three-dimensional, finite difference code CONCHAS-SPRAY to accurately predict flame acceleration and DDT in nuclear plants. Upon completion of the final test series, it is expected that an adequate data base will exist to quantify the effects of flame acceleration. These data are needed to resolve flame acceleration issues for BWR MARK III and PWR ice condenser plants. Experiments in the Heated Detonation Tube (HDT), which will address the effects of temperature, initial pressure and density, and inert diluent on detonability, will be completed in FY 1986.

2.3 Containment Behavior Research

The research effort in this area is depicted in detail in Figure 2.5. Note that it is also divided into two major sub-areas, one concerned with actual containment structural integrity and the other with the behavior of essential equipment within the containment during severe accidents.

2.3.1 Containment Performance Research

The first sub-area consists of experiments on large models of containment structures and penetrations, as well as full size experiments on seal and gasket materials, valves, and electrical penetrations. The purpose of the experiments is to determine how well state-of-the-art analytical methods can predict the time and location of containment failure under severe accident loading scenarios. Note that no computer codes are being developed: existing large deformation structural codes are being utilized instead. Experiments on steel containment

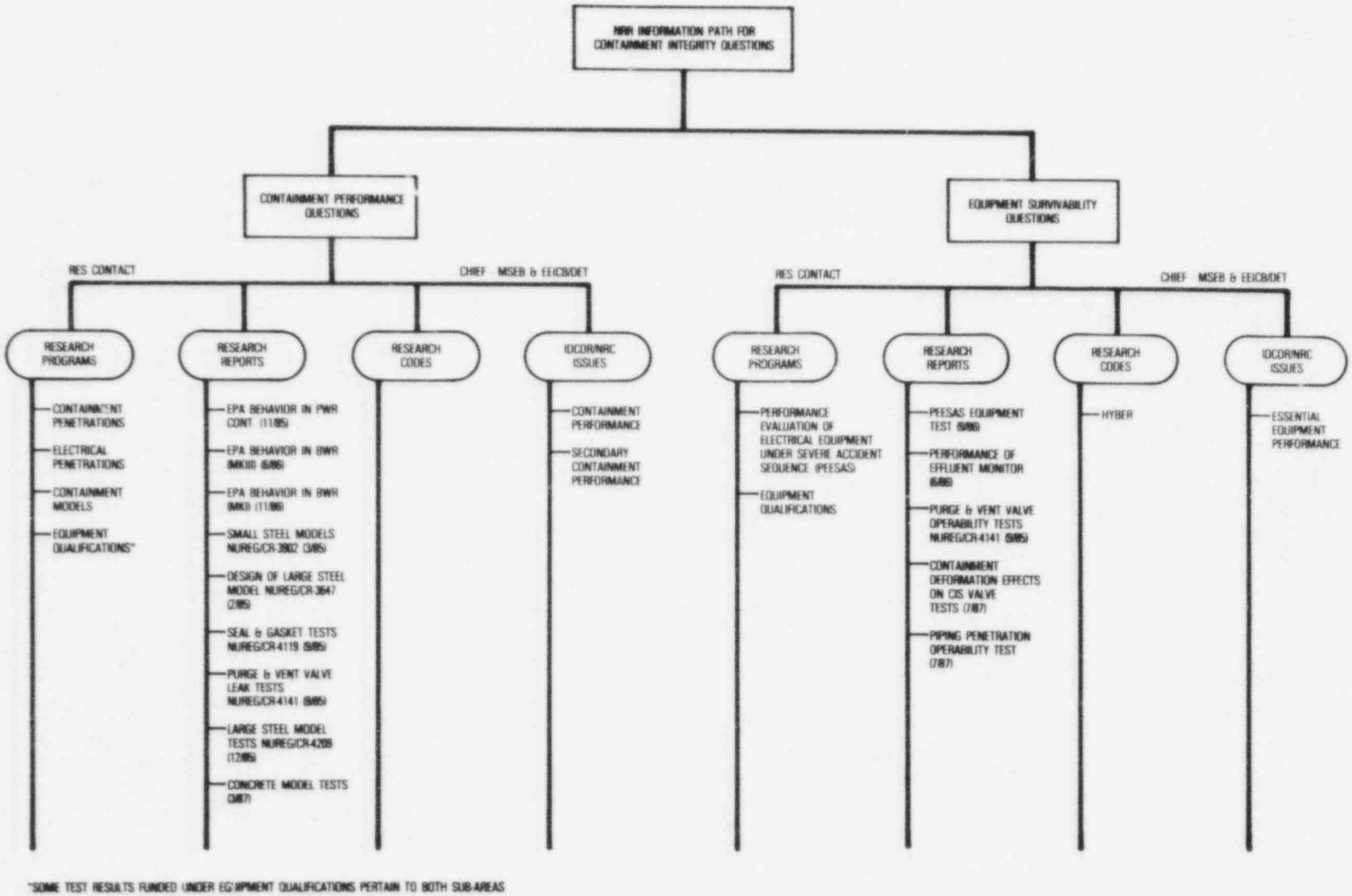


Figure 2.5 Containment behavior research

models were completed in FY 1985 and a concrete containment model is under construction and will be tested to failure in FY 1987. Seal and gasket material tests, aimed at ascertaining leakage potential under severe accident temperatures and pressures, will be completed in FY 1986. Separate effects tests on penetrations will be completed in FY 1987. Tests on electrical penetrations and valves under severe accident conditions will be completed in FY 1986 and FY 1987 respectively. The experiments in this program are designed to cover the wide range of containment and penetration designs used in US reactors. The end use of the data will be in judging the credibility of predictions of containment failure modes performed by licensees for particular plants.

2.3.2 Equipment Survivability Research

This sub-area of developmental research is conducted on instrumentation proposed for use during severe accidents such as liquid level detectors. Experiments are also conducted on the behavior of critical equipment components under severe accident ambient conditions (i.e. very high temperatures and pressure and complex atmospheric conditions such as large concentrations of aerosols, hydrogen, hydrocarbons, and steam).

In FY 1986 the research to be conducted on instrumentation will consist of determining the performance of existing plant instrumentation and electrical components under severe accident conditions. Instrumentation readings are to provide a basis for making accident management decisions and subsequent actions during a severe accident. The data from this research are to be used in evaluating the effectiveness of these actions and to help identify cost effective methods to reduce the consequences of severe accidents. The results are also intended to support accident management and emergency preparedness procedures.

The effort under equipment qualification is intended to address the operability of purge and vent valves (PVV) and other kinds of containment isolation valves (CIV) under severe accident conditions. The capability of these valves to open and/or close during increasing temperature and pressure will be determined by testing under such conditions. In addition, tests to determine the effects of containment response on CIV operability will be performed. All tests will be completed in FY 1987 and the results will provide the NRC with a basis for evaluating the performance of isolation valves under severe accident conditions.

Work in this general area is critical to reducing uncertainties in containment event tree determinations and therefore critical to the final determination of the source term and the associated risk.

2.4 Risk Profile Determination

In order to put all this work into the perspective of how risk profiles have and will continue to be obtained, Figure 2.6 illustrates a schematic of the path taken by the NRC staff in determining the risk profile for any LWR plant. This diagram shows the essential methodology needed and how that methodology is supported by the six sub-areas of research noted above. Note that if a risk profile of a given plant is not available from the current and future information base (i.e. NUREG-0956 & NUREG-1150), and is sufficiently different from established plant profiles, four major computational analyses must be accomplished. First, a determination of all accident sequences (including probability

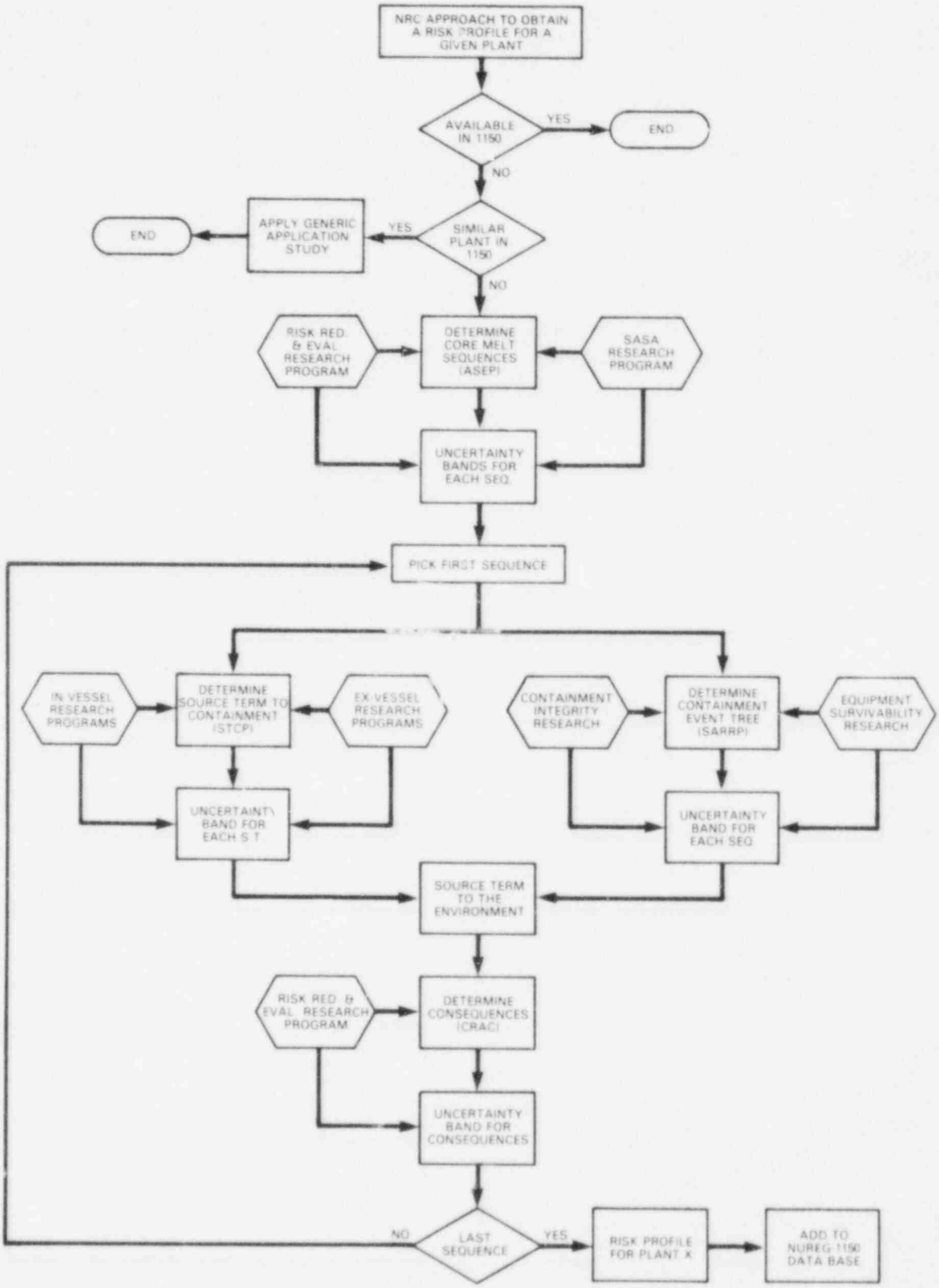


Figure 2.6 Determination of risk profiles

of occurrence and uncertainty analyses) leading to core melt is made using results from the ASEP research program and the SASA program. Secondly, for each sequence, a calculation of the source term to the containment using the STCP is made including the associated uncertainties. This information is used for the third calculation (probably done in parallel) for the containment event tree (i.e. containment failure type, timing, and probability). Given the source term to the environment with its associated uncertainty, the fourth major computation describing the consequences is made for each sequence. The risk profile is obtained by combining the consequences and probabilities for each sequence and the information for that plant is added to the NUREG-1150 data base. Note that supporting research programs feed the calculations by providing state-of-knowledge improvements in the calculational models used (the STCP and MELCOR) and in a continually improving module for determining the uncertainties in each of the four major calculations.

3 DETAILED SEVERE ACCIDENT RESEARCH PROGRAM

The Severe Accident Research Plan, NUREG-0900, was issued in January 1983. The plan is subdivided into thirteen program elements. In this report, the thirteen elements have been grouped into six topical areas for convenience. The Severe Accident Research Program (SARP) includes two different types of projects; safety assessment studies and phenomenological research.

Safety Assessment Studies

- Risk Reduction and Evaluation
- Severe Accident Sequence Analysis

Phenomenological Research

- In-Vessel Melt Progression and Fission Product Behavior
- Ex-Vessel Fission Product Behavior and Containment Loads
- Containment Performance
- Equipment Survivability

The phenomenological research efforts are oriented to the development and validation of computer codes. Because of the variety of plants, the number of severe accident sequences and the cost of large-scale experiments, it is not practical to undertake a set of "proof tests" that demonstrate the plants' behavior during severe accidents. Computer codes are required to extrapolate from small scale and separate effects tests in order to predict the progress of a particular type of severe accident sequence in a specific plant design. Section 4 of this report describes the different computer codes that are being written and validated in the Severe Accident Research Program.

3.1 <u>Risk Evaluation and Sequence Analysis Research</u>

3.1.1 Risk Reduction and Evaluation Research

3.1.1.1 Scope of Research

This topical area includes efforts being undertaken in Elements 1, 10, 11 and 12 of the Severe Accident Research Program as well as the Source Term Reassessment Study in Element 9. This topical area provides the integration for all of the results of SARP as required to address the fundamental regulatory questions identified in NUREG-1070. The products of this topical area provide the technical basis on which regulatory decisions are made.

In the Source Term Reassessment program an improved methodology for source term analyses has been demonstrated by application to six reference plants. The NRC document NUREG-0956 will provide the results of these analyses, an assessment of their uncertainties, an evaluation of the status of validation of the methodology used, an assessment of the impact of improved source term methodology on predicted risk and an assessment of regulatory implications. This study provides the technical basis for the Source Term Regulatory Implementation program in NRR.

3.1.1.2 Research Accomplishments to Date

Accident Sequence Evaluation Program (ASEP). ASEP is: (1) rebaselining the plant-specific PRAs performed earlier for Surry, Peach Bottom, and Sequoyah; (2) using the RMIEP results for developing the accident sequence likelihood information for LaSalle; and (3) using previous re.iews of the Zion PRA. This PRA information is being used to perform these plant-specific analyses. With the exception of Zion and LaSalle, the rebaselining efforts generally follow the tasks outlined in the IREP Procedures Guide.

For the reference plant analyses, a draft methodological guidance has been developed to maintain consistency on the application of the methodology, models, computer codes and data for each of the reference plant analyses. Plant visits to obtain plant-specific information have been completed. The information gathered includes plant and system configurations, dependency information, support system initiators, plant-specific data, visible common causes, recovery actions, and procedures and operations required for the accident sequence likelihood analysis. ASEP has completed the identification and description of the accident sequences through event tree modeling and has selected the dominant accident sequences for detailed investigations. Fault tree modeling has been completed and a failure rate data base has been established for each plant. The models have been computerized and linked for accident sequence quantification. For each dominant accident sequence identified for each reference plant, its core melt frequency has been estimated and its major contributors has been identified. As of January 1986, uncertainty and sensitivity analyses have been completed for Surry and Peach Bottom. These results provide the additional insights on the point estimate frequencies due to the lack of knowledge on modeling assumptions and data. Before the documentation of the reference plant analyses, a second plant visit has been scheduled to assure the plant-specific information and the assumptions made are correct. This confirmatory plant visit has already been made for the Peach Bottom analyses. Throughout the analyses, continuous quality control and high level peer review have been implemented to assure the program activities are technically accurate, consistent and traceable. The documentation of the six reference plant analyses will be completed by summer of 1986.

The results of the reference plant analyses will provide valuable insights to the ASEP generic analysis. The reference plant accident sequence likelihood information can be extended to develop insights regarding factors influencing accident sequence likelihood for all plants. This industry-wide ASEP information base can be valuable in supporting regulatory and licensing applications for plants that have no PRAs.

Source Term Reassessment Program. In July of 1985, the NRC published a report "Reassessment of the Technical Basis for Estimating Source Terms," NUREG-0956, describing an improved methodology for the analysis of source terms for severe accidents. The computer codes comprising this methodology have been coupled to form the Source Term Code Package (STCP) which is scheduled for public release in the spring of 1986. These methods in an interim version of the code are being used to provide improved risk estimates for the reference plants. These radionuclide release calculations for selected severe accident scenarios supplement the analyses in BMI-2104 using the same codes as the code package, but in their stand-alone forms. Accident scenarios will be analyzed to develop the risk profile for each plant by a binning process. In particular, each of the accident scenarios identified by ASEP (the Accident Sequence Evaluation Program) and the Severe Accident Risk Reduction/Risk Rebaselining Program (SARRP) is mapped to one of a set of source term bins. These bins describe the timing, quality and characteristics of release of fission products to the environment. Calculations are performed with the full STCP for source term bins which are highly individual. Extrapolations may be made to obtain source terms for other bins if the amount of extrapolation is reasonably small and if the physical basis for the extrapolation is sound. Key uncertainty parameters and their effects on the source terms are determined by code sensitivity calculations. Calculations performed for the reference plants will be checked for quality by BNL and published in technical reports. Several scenarios and important phenomenological issues will be independently studied by DAE/BNL in QUASAR (Quantification and Uncertainty Analysis of Source Terms for Severe Accidents in Light Water Reactors)

Severe Accident Rebaselining and Risk Reduction Program (SARRP). The overall objectives of SARRP are as follows:

(1) Incorporate insights gained from severe accident research toward a rebaselining of reactor risks and of the overall uncertainty.

(2) Investigate the use of generic plant categories as a means for generalizing the results of plant-specific risk analyses.

(3) Evaluate the risk reduction potential of proposed new safety features designed to reduce the frequencies and/or consequences of severe accidents.

(4) Perform cost and feasibility assessments for safety options that have promising risk reduction potential, and

(5) Provide technical support to NRR reviews of individual plant examinations, using as a foundation the results obtained above.

Initially, in SARRP, a data bank of risk parameters was developed for 8 plants (not necessarily the same plants as for NUREG-1150) that had been analyzed previously in PRAs. The effects on these risk parameters of about 20 safety options listed in NUREG-0900 were evaluated and incorporated into the data bank. Risk reduction calculations were performed for the individual safety options and for combinations of safety options. A procedure was developed for evaluating the financial risk (i.e , the mean cost per reactor year) from reactor accidents. Offsite, onsite, and total financial risks were evaluated for the 6 reference plants. This process has continuously upgraded the state of knowledge for the past several years.

A task force was organized (SARRP Phenomena Assessment Task Force) to estimate uncertainty bounds in the phenomenological parameters important to risk. This information was later used for NRC working groups on containment loading, containment performance, and fission product source terms. The results of this work are now being used as one element of the limited Latin hypercube uncertainty evaluation.

The values and impacts of providing filtered, vented containment systems in boiling water reactors were studied in detail for Mark I and Mark III containments. The risk reduction potentials of various design options and operational strategies were compared, explicitly accounting for potential system interactions and modes of degraded operation. Criteria necessary to achieve cost-effectiveness were identified.

The severe accident risk reduction and rebaselining program (SARRP) currently is evaluating six reference plants (Surry, Peach Bottom, Sequoyah, Grand Gulf, Zion and LaSalle). The work for Zion will draw heavily on past NRC evaluations. The following tasks will be performed in FY 1986:

- Continue to develop and document containment event trees for five reference plants.

- Develop containment event tree for Mark II plant (LaSalle),

- Complete risk profiles for all reference plants and evaluate risk reduction alternatives.

The containment event trees are a synthesis of systems and phenomenology (both analytical and experimental) information that is obtained from SASA, Battelle Source Term Code Package runs, past PRAs, generic safety studies, NUREG/CRs on phenomenology, FSARs, Architect-Engineering reports, operating procedures, plus information obtained from IDCOR. A computer code developed by SNL, EVENTRE, assembles the data, based on a logical structure determined by SARRP analysts, and calculates the release fractions from the containment by various pathways.

In addition, the SARRP program assembles the systems information from ASEP to provide the "risk profiles." The offsite consequence calculations are also calculated under the SARRP.

MELCOR Code Development and Validation. In 1982 the NRC initiated the development of a new severe accident computer code called MELCOR. The objectives of the effort were threefold:

(1) Development of a code for use in risk studies and staff reviews of such studies (the MELCOR code system) which: (a) appropriately models phenomena essential to the description to severe core damage accidents, (b) provides credible predictions of the progression and consequences of severe core damage accidents, (c) permits quantitative estimates of the sensitivities and uncertainties associated with those predictions to be made (and the principal sources of this uncertainty), and (d) has a structure that facilitates the incorporation of new or alternative phenomenological models.

(2) Identification of the statistical tools, application strategies, and supporting data required to estimate the sensitivities and uncertainties associated with MELCOR code predictions.

(3) Demonstration of the use of the MELCOR system of codes and its associated statistical tools in several case study applications.

The objectives of the MELCOR program will be met by developing and demonstrating by case study applications a new system of risk assessment codes, the MELCOR code system, and a set of associated statistical tools for the performance of sensitivity and uncertainty analyses with that code system.

By the end of CY 1985 the MELCOR program: (1) performed and documented assessments of the phenomena that determine the course and consequences of severe core degradation accidents; (2) developed and implemented a code structure that meets the objectives of the MELCOR program; (3) developed where necessary, coded, and tested models of the major phenomena that will be treated by the MELCOR code system; (4) compared statistical techniques that have been used to determine the sensitivities and selected a set of techniques appropriate for use with the MELCOR code system; (5) used the set of selected techniques to perform sensitivity studies of results obtained using the MAEROS, CRAC2, and MARCH codes, thereby beginning to develop the strategies and data required to conduct sensitivity and uncertainty studies using the MELCOR code system; (6) released MELCOR 1 (BWR version) to NRC contractors for testing, verification, and validation; and (7) progressed substantially toward the completion of the required PWR models.

3.1.1.3 Discussion of Outstanding Issues

One of the most difficult issues facing the NRC in the area of SARP plant safety and risk analysis is the extent to which a set of reference plants can be used to accurately portray the risk from all existing LWRs. As noted above, ASEP is attempting to combine plant design and risk data in order to develop generic plant categories. The degree of success in this work will largely dictate to what extent ASEP (and SARRP) results can be applied to other similar plants and other classes of plants.

The efforts of ASTPO address a number of phenomenological issues relating to the release and transport of fission products as discussed in Section 3.5. At a higher level the source term issues are:

(1) Have previous studies significantly over estimated the consequences of severe accidents?

(2) Are reactors inherently safer than previously recognized?

(3) What changes should be made in the regulations to account for an improved understanding of source terms?

Because of SARRP's integrating role in SARP, the issues to be addressed in this program are the four fundamental regulatory questions discussed on page 1-2 of this report. That is, SARRP will provide the technical basis for regulatory consideration of the need for additional severe accident protection and the most cost-beneficial means for meeting such needs. The technical information will first be provided for the six reference plants. Depending on the degree of success in the ASEP development of generic classes, general insights on the need for additional protection, and cost-benefit of methods for achievement for the remaining existing LWRs may also be provided by SARRP for regulatory consideration.

3.1.1.4 Planned Activities

Accident Sequence Evaluation Program (ASEP). ASEP will continue the current six reference plant analyses through most of FY 1986. A final NUREG/CR on the NRC reference plant accident sequence likelihood characterization will be produced in late 1986. A guideline report on the methodology used for the reference plant analyses will also be published in late 1986.

Two additional plants will be added as part of the NRC reference plants. They will be PWR plants, both CE and B&W vendor types. The plant-specific analyses for the B&W plant (exact plant not chosen at this writing) will be initiated in late FY 1986 and continue through FY 1987. The CE plant-specific analysis will be initiated in FY 1987. Both analyses will follow the methodology established for the current reference plant analyses.

ASEP will also continue its generic analysis and use the insights gained from the reference plant analyses to expand the refine the industry-wide accident sequence likelihood information base. Plant-specific information will be verified. Additional sequences and systems will be analyzed to provide a more complete information base. The information will be peer reviewed and published as technical reports. Recommendations will be made on the use of the information for licensing and regulatory applications. This industry-wide accident sequence likelihood information base, as well as the one for the reference plants, will be computerized by the SARA and PRISIM programs for NRC applications.

Source Term Reassessment Program. A key product of the Severe Accident Research Program is a reassessment of the severe accident source term technology. This reassessment was issued as Draft NUREG-0956 (for comment) in July 1985 and will be published in final form early 1986. NUREG-0956 describes NRC staff and contractor efforts to reassess and update the agency's analytical procedures for estimating accident source terms for nuclear power plants. The effort included development of a set of computer codes that is intended (a) to replace the methods of the Reactor Safety Study (WASH-1400) and (b) to be used in reassessing the use of TID-14844 assumptions (10 CFR Part 100). Both the Reactor Safety Study methods and the TID-14844 assumptions are currently used in many areas of regulatory practice. Improved analytical procedures are needed in some areas of regulation to resolve safety questions, to assess the adequacy of current regulatory practices, and to implement the Commission's Severe Accident Policy Statement. NUREG-0956 describes the development of these codes, the demonstration of the codes to calculate source terms for specific cases, the peer review of this work, some perspectives on the overall impact of new source terms on plant risk, the plans for related research projects, and finally the conclusions and recommendations resulting from the effort.

This source term reassessment leads the way into three closely related planned activities. First, NUREG-0956 recommends that the new source term analytical methods be used for regulatory purposes notwithstanding their limitations because they are so far advanced compared to the current regulatory methods. Attempts to use these new methods will require considerable research support. This support is expected to be in the form of additional calculations with the integrated Source Term Code Package and special analyses using detailed mechanistic codes to address difficult technical questions.

Second, NUREG-0956 identified major areas of uncertainty requiring additional research. This additional work will be in the form of continued experimental programs, code improvements in NRC's integrated codes and detailed mechanistic codes, and extended uncertainty studies such as the new Brookhaven study. The uncertainty studies are important because, even if uncertainties are reduced to acceptable levels, the uncertainties must be quantified to support many regulatory applications.

Third, in the process of discussing NUREG-0956 with industry groups (particularly IDCOR), a number of technical issues were identified on which the NRC and the industry groups have reached different conclusions. These issues must be resolved in order to effect regulatory changes and to establish confidence in the new analytical methods. Special efforts have been organized to address each of these prominent issues (about 18 in number). Resolution of some of these issues will require further experimental results from the Severe Accident Research Program.

Severe Accident Risk Rebaselining and Risk Reduction Program. The SARRP program is the focal point and integrating task for the Phase I and II results of SARP. The following SARRP activities and milestones are planned.

The methods of analysis used to predict fission product release and transport behavior are essentially the same as those presented in NUREG-0956, Reassessment of the Technical Bases for Estimating Source Terms. These computer codes have been assembled as a Source Term Code Package (STCP) to simplify the use of the codes and reduce the potential for input error by automating the interfaces between codes. In addition, a number of improvements in the models, or in the coupling between models, have been made.

Figure 3.1 illustrates the manner in which the codes are grouped in the Source Term Code Package. The MARCH 2, CORSOR, and CORCON-Mod 2 codes are now coupled. The CORSOR-M version of the CORSOR code, which uses an Arrhenius form for the empirical correlation, has been incorporated into MARCH. A consistent treatment can now be made of the release of fission products and the transport of sources of decay heat from the fuel. Similarly, CORCON-Mod 2 is now used in the code package to predict the thermal-hydraulic loads on containment resulting from core-concrete interactions and as input to the VANESA code to calculate fission product release. In BMI-2104 these processes were treated in an inconsistent manner. Potentially significant changes also resulted from the intimate coupling of the MERGE and TRAP-MELT codes in the code package.

Architect-engineering studies of the feasibility of retro-fitting safety options into existing plants, the potentially negative impacts, and the associated costs will continue, with emphasis being given to additional safety options and reference plants beyond those considered during FY 84.

The risk for the six reference plants treated in the ASTPO Studies and and the risk reduction potentials of the various safety options listed in NUREG-0900 will be reevaluated. Several types of risk measures will be considered, including (a) core melt frequency, (b) population dose, (c) early fatalities, (d) latent cancer fatalities, (e) off-site cost, (f) on-site cost, and (g) mortality probability for the most exposed individual. Results will be displayed both for annual risk and for the consequences given a core melt. All displays will include estimates of uncertainty and/or sensitivity. The results from the

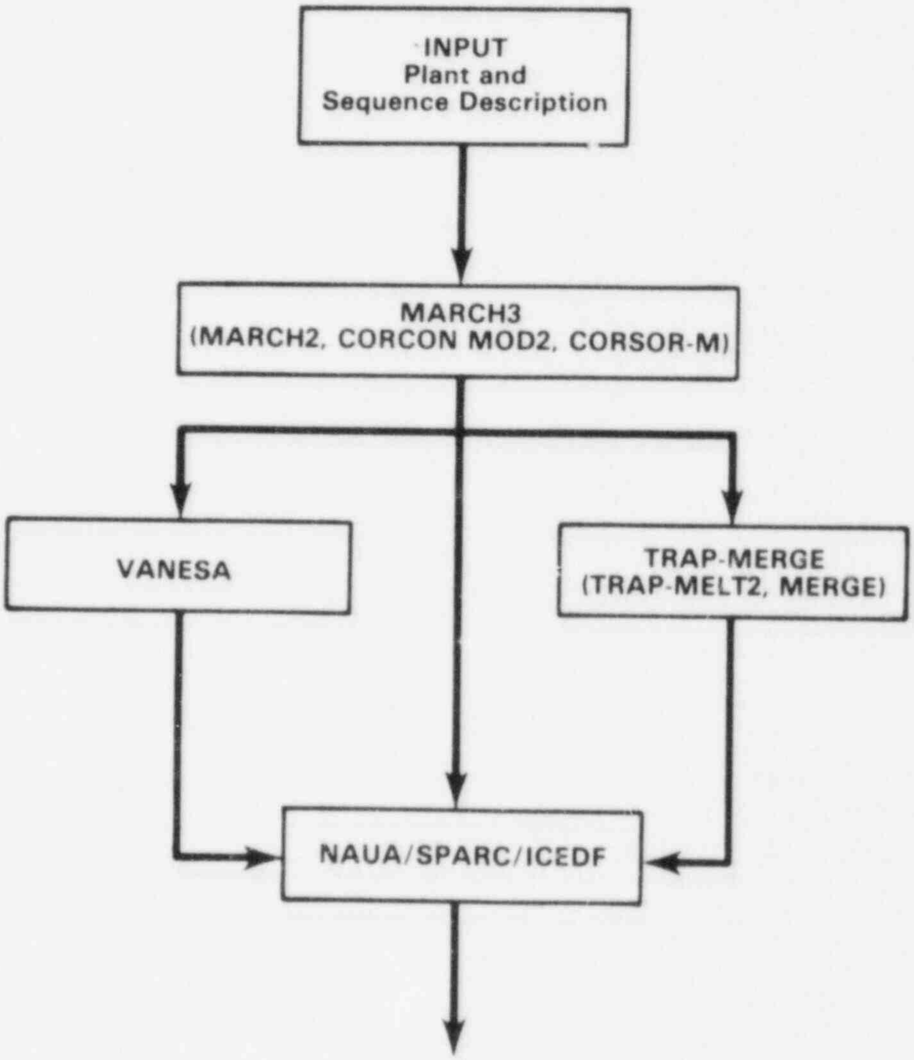


Figure 3.1 Components of the source term code package

risk reduction evaluation will be combined with cost information to produce comparisons of cost versus benefit.

As the draft reports on the SARRP six-plant analyses are completed, a second report will be initiated which describes in detail the methods by which the SARRP risk evaluations and cost-benefit studies were performed. The purpose of this report is twofold: to clearly specify how the analyses of the six plants were made, permitting more complete review; and to establish a set of procedures by which similar studies could be performed by other organizations. The latter would be of particular value to licensees potentially required to perform risk and cost-benefit analyses. Such a requirement is one possible aspect of the severe accident implementation plan discussed in the proposed severe accident policy statement. As Figure 3.2 indicates, this documentation would be provided in draft form by March 1986, with a final report planned for July 1986.

In addition to the work described above for SARRP, considerations of more generic risk and cost-benefit are possible. The degree to which this will occur depends heavily on the success of ASEP in developing a manageable set of generic plant classes (as described previously). Such generic analyses are not intended to provide precise risk estimates of the plant classes or individual plants. Rather, the purpose of such studies would be to provide a rough ranking of plant classes according to risk, and to identify what possible plant modifications would likely be most cost-beneficial for such classes. Such information could be valuable in helping to orient and bound the plant-specific severe accident studies now recommended in the proposed severe accident policy statement.

MELCOR Code Development, Validation, and Maintenance. MELCOR was designed to replace the MARCH-based risk codes and the related Source Term Code Package (STCP). MELCOR first became operational in early 1985, an interim version was provided to BNL in mid 1985 for independent assessment, a completed BWR version MELCOR-1.0 was released at the end of 1985, and versions with PWR capabilities as well are scheduled for release in mid 1986 (MELCOR-1.5, adding large dry containment capability) and at the end of 1986 (MELCOR-2.0, adding ice condenser containment capability).

Code validation for MELCOR has been initiated and will run in parallel with development of the fully capable (BWR and PWR) MELCOR-2.0. Code validation is being performed by the MELCOR developers at SNL, and an independent validation effort referred to as benchmarking will be conducted at BNL. The validation efforts will compare MELCOR calculations with calculations from detailed mechanistic codes as well as with data to establish confidence in the MELCOR code or to lead to code improvements if needed.

Code maintenance, which will continue, will provide for required changes, correction of errors, documentation, configuration control, user support, and peer review. Based on NRC's experience with peer review of the BMI-2104 suite of codes, code maintenance activities may be extensive.

3.1.2 Severe Accident Sequence Analysis (SASA) Research

3.1.2.1 Scope of Research

This topical area (Elements 2 and 3 of NUREG-0900) covers the response of the plant to severe accident conditions not only in terms of the behavior of hardware

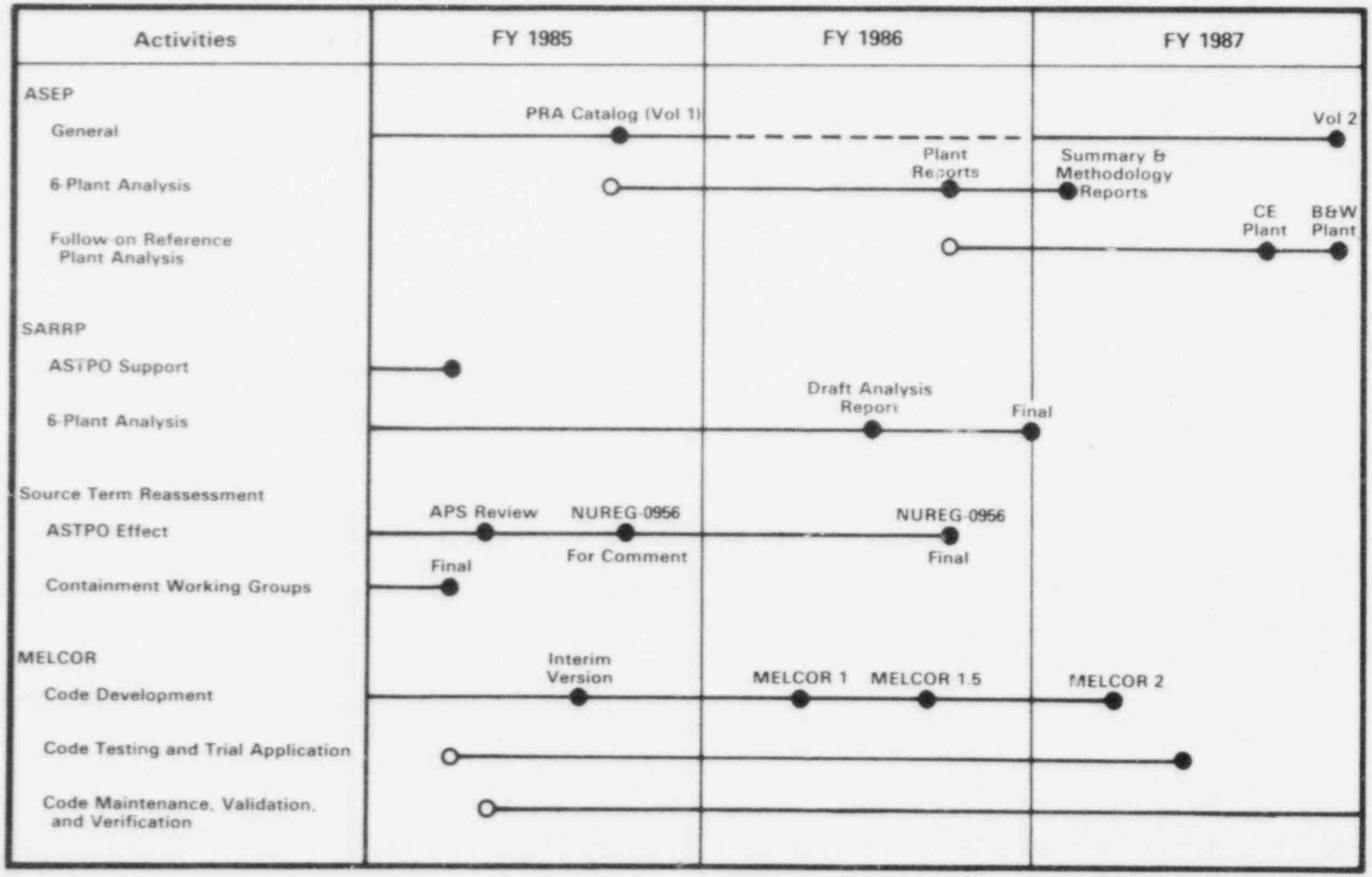


Figure 3.2 Planned activities for risk reduction and evaluation research

but also the performance of the operating staff. In the Severe Accident Sequence Analysis (SASA) program detailed investigations are performed of plant behavior in severe accidents. The risk-dominant sequences for each plant analyzed are selected from appropriate probabilistic risk assessment (PRA) programs such as the Accident Sequence Evaluation Program (ASEP), the Interim Reliability Evaluation Program (IREP), and the Reactor Safety Study Methodology Application Program (RSSMAP). These risk-dominant sequences provide the framework for analyses of complex accident sequences in which critical equipment performance and operator actions are simulated. SASA analyses differ from typical PRA analyses of severe accident sequences in that they employ the best available mechanistic computer codes and greater effort is expended in determining the minimum safety features required to prevent core meltdown, the influence of the plant control system, and the effects of alternative operator actions on the accident scenario. Accident management studies are also being performed to determine how emergency procedures can be improved to reduce the likelihood of off-normal conditions degrading to the level of severe fuel damage or in the event of core meltdown to provide protection to the containment.

3.1.2.2 Research Accomplishments to Date

SASA studies which have been completed include: Small break LOCAs, loss of AC power, large LOCAs, interfacing system LOCAs, loss of feedwater (LOFW) transients, BWR loss of decay heat removal, BWR loss of injection, BWR anticipated transient without scram (ATWS), and multi-plant common mode failures.

The SASA program has interacted closely with IDCOR and similar activities in France, Germany, and the United Kingdom. In addition to documentation of the severe accident studies, presentations have been made to I&E personnel on severe accident sequences for both BWR and PWR plants for their emergency response facilities training appraisal program.

The completion of these analyses has provided valuable insights into the response of various plant designs to scenarios in categories of concern to NRC licensing and research. For example:

(1) The loss of AC power analyses have provided plant response, equipment performance, and event sequence timing information in support of the NRC's effort to resolve the Station Blackout Unresolved Safety Issue (USI A-44).

(2) The desirability of an automatic shift of the high pressure coolant injection (HPCI) pump suction on high sensed pressure suppression pool (PSP) level without the opportunity for reversal by the operator has been examined. A design change is under review by TVA to avoid this problem.

(3) The study of risk dominant sequences at Browns Ferry has determined that current control logic may need some modification to allow operator insertion of partially inserted rods.

(4) The reactor building fire protection sprays were shown to be effective in reducing fission product transport through the secondary containment of BWR plants. This has not been considered in PRAs.

(5) The SASA program enhanced the understanding of the capability of existing PWRs to remove decay heat using "Feed & Bleed" following loss of all secondary cooling. This will assist in the resolution of USI A-45.

(6) The loss of decay heat removal study for BWRs has determined that the effect of CRD hydraulic system and standby liquid control (SLC system) should be factored into the ASEP program and operator training.

(7) Sequence analyses completed to date have developed an expanding data base of great value to other programs. The data base is being used to develop operator action event trees that can be used to define appropriate operator actions for a variety of scenarios.

(8) An analysis for the potential for local detonation in a large dry containment has shown that the hydrogen concentration provides a detonable mixture only at a single position in the most constrained location. This potential for equipment damage appears minimal but is being evaluated.

Alternative accident management strategies have been examined as part of a number of the SASA studies to determine not only the most likely scenario for the accident but also to identify the actions which would lead to the most favorable outcome. Studies of control room personnel and other plant staff involved in managing severe accident sequences were performed in FY 1984. The TC sequence (ATWS involving closure of all mainstream isolation valves) in Browns Ferry 1 was analyzed from a human factors perspective to identify the procedures, man-machine interfaces and training which could significantly affect the sequence's outcome. Assessing the reliability of critical operator actions and evaluating the displays and alarms utilized by the crew enhanced our understanding of how the crew's response to that sequence might be optimized. Similar studies for a broader range of sequences should be performed.

3.1.2.3 Discussion of Outstanding Issues

The Commission has divided the consideration of severe accidents (April 18, 1984 version of NUREG-1070) into the two categories of (1) existing plants and (2) future designs. In evaluating the need to reduce the hazard to the public from severe accidents in existing plants, the Severe Accident Research Program must consider regulatory questions identified in the introduction to this section. The SASA and accident management studies address the safety of existing plants with respect to severe accidents and evaluate methods for increasing the level of protection for severe accidents.

In SASA, deterministic analyses are made of severe accident consequences using the best available computational techniques. The SASA and associated accident management studies also provide input into determining which operational changes could be most cost-effective in reducing the risk of severe accidents. The results of these studies provide input to the integrating task Severe Accident Rebaselining and Risk Reduction Program, which addresses the safety of existing plants from both a deterministic consequence viewpoint and from a risk viewpoint, and which compares the cost-benefit trade-offs of various design modifications with the cost-benefit trade-offs of operational changes.

Perhaps the most important lesson learned from the TMI-2 accident was that the operators were not adequately prepared to recognize the symptoms of severe accidents and to cope with them appropriately. Through the analyses being performed in the SASA program, a catalog of severe accident behavior is being developed for a variety of accident sequences in different plant designs. These realistic analyses of accident behavior are essential to the development of improved guidelines for the operator in responding to severe accidents.

3.1.2.4 Planned Activities

A schedule of planned activities is provided in Figure 3.3. The accident sequences to be analyzed based on findings of the ASEP program as dominant sequences include: the loss of off-site and on-site AC power plus the loss of auxiliary feedwater (TMLB'), the small break LOCA with the loss of high pressure injection system (S2D) for PWRs, and the anticipated transient without scram (ATWS) for BWRs. Specific plants under study include:

Bellefonte, a B&W Plant	(FY 86,87)
Seabrook, a W plant	(FY 86,87)
Surry, a W plant	(FY 86)
Browns Ferry, a BWR MK I	(FY 86)
Limerick, a BWR Mark II	(FY 86, 87)
Grand Gulf, a BWR Mark III	(FY 86, 87)

For the Bellefonte and Seabrook plants an investigation has been started for the TMLB' sequence to determine potential system failure points and their timing during high pressure core damage conditions. Also more detailed modeling of the core and upper plenum regions will be conducted to evaluate natural circulation flow patterns during this transient.

The SASA study of accident sequences involves a base case study in which no operator action is involved. This determines how long it takes for the accident to get to core uncovery or to a core melt situation. The study then involves the operator actions for accident management. This study examines the various actions that an operator can take to prevent or mitigate the severe accident from getting into the core melt situation.

The SASA program is oriented to plant specific analyses because this is the only way to conduct a meaningful study that includes the effect of operator actions. Also, the event timing is most susceptible to plant specific considerations. Nevertheless, almost all of the major findings and conclusions of each study are applicable on a generic basis to classes of plants. Sequence events trees (SETs) are developed which provide a logical representation of the systems that are challenged by the accident and potential operator actions that could be taken to mitigate the consequences of the accident. The SETs provide the basis for accident management studies in FY 86 and beyond.

In FY 1985 a framework was completed for considering the NRC's role in accident management and how accident management relates to and is distinguished from the well defined terms of risk management, emergency management and safety management. This included a review of existing regulations, their relevance to accident management, and the industry's responses to these regulatory positions. A review of the French "U" procedures also is planned. These procedures are oriented toward ultimate efforts to prevent core melt in PWR's and toward the management of containment integrity given a severely damaged core. The review

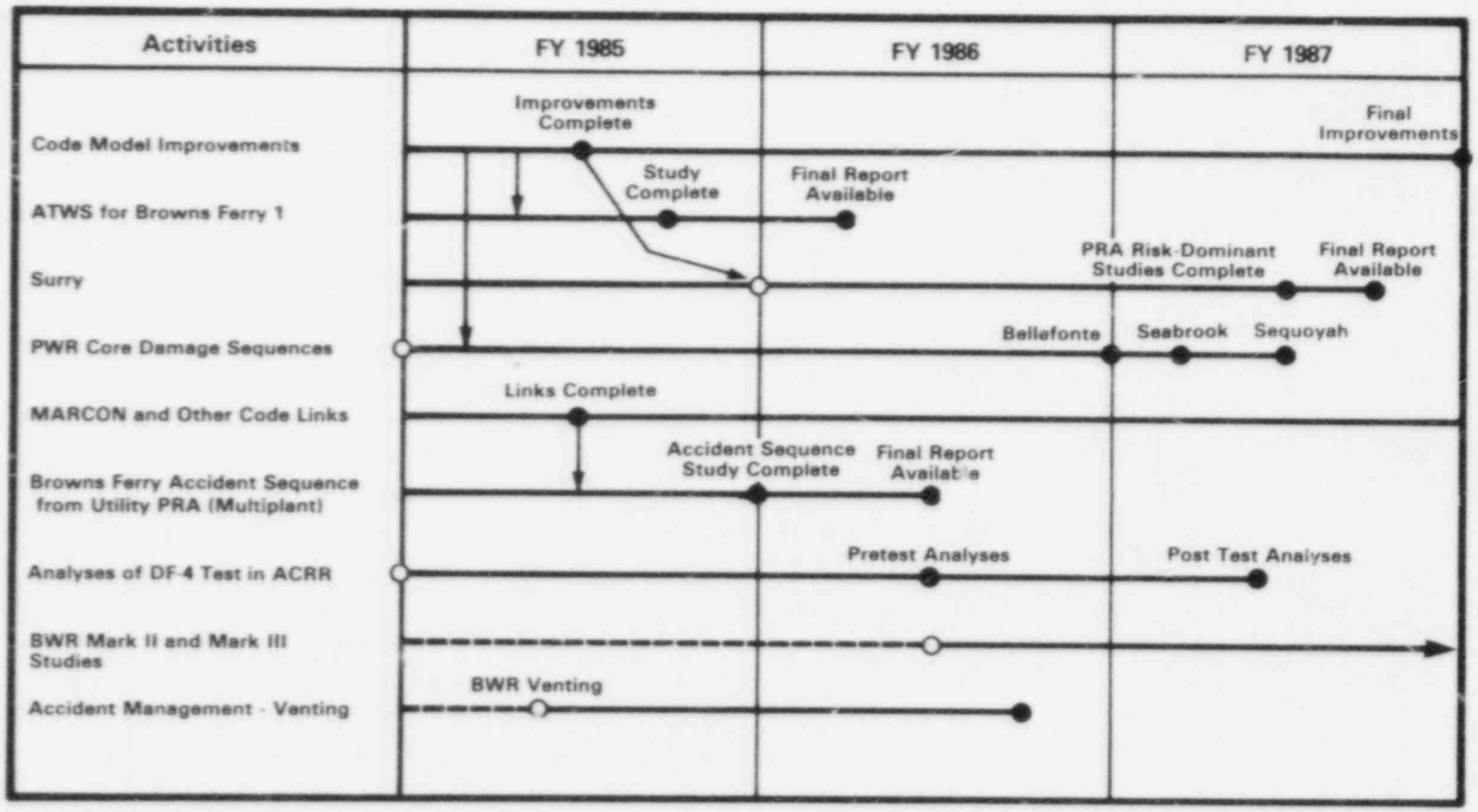


Figure 3.3 Planned activities for severe accident sequence analysis research

in progress will assess the relevance of such procedures and judge the values and impacts of extending that concept to U.S. nuclear operations.

Accident management as affected by "operator actions" vs. "no action" has been an important constituent of SASA studies. A more particular accident management need, the evaluation of the effectiveness of early venting of the wetwell air-space at Peach Bottom to eliminate or significantly reduce the source term from early containment failures, began in late FY 198[illegible]. Evaluation is underway of strategies available to manage the containment and its associated systems to mitigate the release of fission products to the public in core meltdown accidents. The plants under study include: Bellefonte with large dry containment, (FY 86, 87) and Surry with a sub-atmospheric containment (FY 86). Radiological consequences associated with containment failure are part of this study. The accident sequences are selected in cooperation with the ASEP program. The development of methodology for management of PWR containment during severe accidents as part of the overall goal to protect public health and safety will begin in FY 1987.

3.2 Source Term and Containment Loads Research

3.2.1 In-Vessel Research

3.2.1.1 In-Vessel Melt Progression Research

3.2.1.1.1 Scope of Research

In-Vessel Melt Progression (Original SARP Element 4) deals with the progression of a severe accident from initial core uncovery through core heatup, cladding oxidation and hydrogen generation; fuel liquefaction, melting, and relocation, melt attack on the reactor internal structure and the vessel; and vessel failure, the mode of vessel failure, and the initial conditions for melt entry into the reactor cavity and the containment. Fission-product release and attenuation and aerosol generation and transport occur throughout this in-vessel melt-progression sequence. Both are highly dependent upon the core temperature distribution and the physical and chemical state of the core which vary with time. Except where the experiments overlap with melt-progression research, the fission-product research is treated separately later in this Section. Hydrogen generation occurs primarily during the in-vessel melt progression process and is measured and modeled in the melt progression research. Melt progression determines the initial conditions, including the melt mass and temperature distribution, for the melt entry into the reactor cavity and the containment for use in Section 3.2.2, "Ex-vessel/Containment Loads Research."

A data base and validated analytical models for the governing processes in the complex in-vessel melt-progression sequences are needed for the assessment of severe-accident consequences and uncertainties. An integrated research program with a number of elements is required to obtain the wide range of needed information. The Severe Fuel Damage (SFD) and Source Term (ST) research program includes integral multiple-effect in-pile tests for multiple-effect interactions at large scale in the PBF and the NRU test reactors; separate-effect phenomenological experiments, both in-pile in the ACRR test reactor and out-of-pile, for model development and model validation testing and to cover the relevant accident parameter range; and analytical model and code development and validation. Two complementary mechanistic severe-accident fuel behavior codes are being developed as part of this program: the Severe Core Damage Analysis Package (SCDAP), and the Melt Progression Model (MELPROG). The validated models and codes, not the experimental data directly, are the tools that are used in the assessment of severe accident consequences and uncertainties. Therefore, the codes and their validation data base are the major outputs of the research program.

The results of the "In-Vessel" research program will provide basic information for assessing the consequences and reducing the uncertainties associated with many of the most important severe-accident issues. These include the source-term issues of in-vessel fission-product release, attenuation, and transport, and the melt progression issues of hydrogen generation and the conditions of melt entry into the reactor cavity and the containment that are the primary determinants of the threat to the integrity of the containment. The results of the research are to be embodied in the mechanistic MELPROG and SCDAP codes that are then used directly in severe-accident safety assessment for the most difficult risk-significant cases and for benchmarking the simplified and faster-running risk analysis codes. Finally, the real significance of the results of this research will be a reduction and quantification of the uncertainties associated with the assessment of severe-accident source-terms and containment-failure probabilities.

3.2.1.1.2 Research Accomplishments To Date

A major part of the Severe Fuel Damage research program is the now completed series of four large integral 32 rod SFD tests in the PBF test reactor under core uncovery accident conditions up to 2500K. These tests use either fresh or high burnup fuel that is trace irradiated by five day operation at full power before the test transient to build up a short-lived fission-product inventory in the test fuel, with a pretest shutdown for build up of the proper cesium/iodine ratio. Fission-product release, hydrogen generation, and temperature distributions are measured during the test transient. The resultant fuel damage conditions are determined by neutron radiography and tomography and by post-irradiation examination (PIE).

The four SFD tests in the PBF have now been completed, but much of the analysis of the results and the Post Irradiation Examination (PIE) for the last two tests with high-burnup fuel along with final reports on all four tests remain to be completed. The facility, which had been dedicated to full time NRC use, has been returned to Department of Energy.

The results of the four PBF SFD tests have produced a substantial base for determining and modeling the governing phenomena under core-uncovery accident conditions. Important integral data have been obtained on rapid-oxidation heating to 2500K, with the resultant generation of large quantities of hydrogen, fuel liquefaction (fuel dissolution in molten unoxidized metallic zircaloy) and relocation downward, damaged fuel characterization, and fission-product release, transport, and deposition. Preliminary reports have been issued on the results of all four tests.

The NRU reactor at Chalk River can accommodate integral coolant boildown tests with full-length fuel bundles. Such tests are important for validation testing at full length of the SCDAP and MELPROG codes, particularly for cladding oxidation and the resultant hydrogen generation where length scaling from the shorter PBF and ACRR data have uncertainties. The initial FLHT (Full Length High Temperature) test to 2200 K with an artificially slow boildown to produce high oxidation has been performed. This test gave significant data for assessing the SCDAP oxidation and fuel damage models under these conditions.

A series of small-scale integral experiments in the ACRR test reactor is providing information on the governing mechanisms involved in core melt progression, and the initial three DF (Debris Formation) experiments have been performed. These experiments use visual diagnostics (cinematography) to give time-continuous data on the melt progression processes and surface temperatures, as well as direct measurement of the hydrogen generation. The data from these in-pile separate-effect experiments are particularly important for model assessment and improvement in the mechanistic MELPROG and SCDAP codes. The first two of the four planned fresh fuel DF experiments gave significant data on the development of fuel liquefaction, relocation, and potential blockage formation as well as hydrogen generation during relocation. It was found that a dense tin aerosol was formed from the molten zircaloy. The recently performed DF-3 experiment with a fuel bundle that included a PWR silver-indium-cadium control rod is giving data on the effects of these materials on core melt progression.

Laboratory separate-effects experiments are providing information on zircaloy oxidation rates and the viscosity of liquified fuel. Most of the program's out-of-pile data on melt progression comes from our German partners in the SFD international program at KfK, including data on the thermodynamics and kinetics of the zirconium-uranium-oxygen system and results of the KfK pioneering integral experiments on fuel rod oxidation and fuel liquefaction and relocation.

Development has been completed on the Mod 1 version of the mechanistic fuel damage code SCDAP and its integration with the RELAP5 thermal/hydraulics code. The Mod 0 version of the complementary mechanistic melt-progression code MELPROG has also been completed. SCDAP, in detailed rod geometry, treats the state of the core during the earlier stages of core-uncovery transients and is particularly useful for analysis of accident recovery by core reflooding as at TMI-2. SCDAP/MOD1 has been assessed against PBF data, and has been the principal tool used in the TMI-2 and LOFT FP-2 analyses. SCDAP uses the models of the FASTGRASS mechanistic fission product release code and the TRAP-MELT fission-product and aerosol transport code.

The mechanistic MELPROG code treats the in-vessel progression of core melt, liquefaction, and relocation in core-uncovery accidents through attack on the reactor internal structure and the vessel, including the mode of vessel failure and the initial conditions for melt entry into the reactor cavity and the containment. MELPROG has been linked to a 2-D version of the TRAC thermal/hydraulics code, and is being used to assess the potential significant effects of in-vessel natural convection on core melt progression. MELPROG also includes a new mechanistic fission-product and aerosol behavior module VICTORIA that includes the TRAP-MELT fission-product and aerosol transport model and the FASTGRASS model. The initial results on the state of the TMI-2 core are becoming available and are being used to benchmark SCDAP. The data available for assessing the core melt progression models in MELPROG are very sparse, although some PBF, ACRR, and NRU results are applicable along with results of the German integral fuel damage experiments at KfK. Useful but limited data are also coming from the TMI-2 core examination. A special set of experiments and analysis for assessment and validation of the MELPROG code has been planned that will start in FY 86.

The mechanistic SCDAP and MELPROG codes will themselves be used to benchmark the new code MELCOR, and to assess codes such as STCP (MARCH module) and MAAP that are currently used in risk studies.

3.2.1.1.3 Discussion of Outstanding Issues

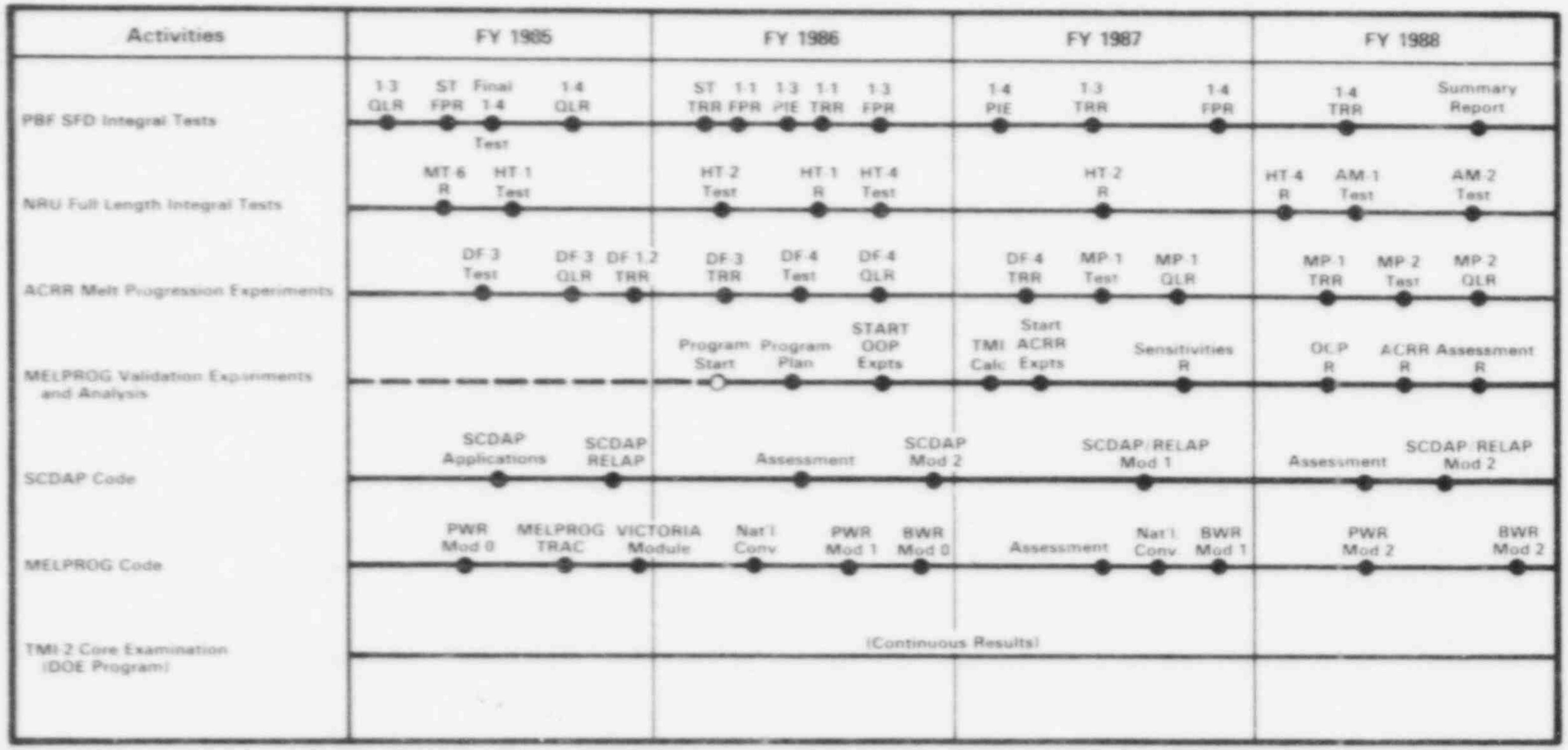
A major part of the uncertainty in the assessment of the radiological consequences of severe accidents arises from lack of information on in-vessel melt-progression processes. Of first importance are the mass of hydrogen generated, mass of core melt, in-vessel thermal-hydraulic behavior, the temperature distribution of liquified or molten fuel and the solid core debris, and the time and mode of vessel failure with melt ejection into the containment. These quantities are determined by the in-vessel melt-progression process, and their magnitudes largely determine the fission-product release and the ex-vessel core-melt threat to containment integrity including the time of containment failure.

Current (BMI-2104) methodology and also the IDCOR MAAP code make the essentially arbitrary assumption that the core collapses and the molten mass is deposited into the lower head of the vessel when a given fraction of the core mass reaches the melting temperature. This assumption can produce large uncertainties in the source term and the risk. The recent QUEST source-term uncertainty study showed that, at least for the Surry TMLB' sequence studied, the large existing source-term uncertainty is about equally divided between uncertainties in fission-product release and in-vessel core melt progression, and that the current uncertainty in the suspended aerosol activity in the containment is about a factor of 100. The melt-progression research described here is an integrated program of research designed to furnish a data base and validated analytical models to reduce substantially the current large uncertainties in the assessment of core melt progression.

At the NRC-IDCOR exchange meeting, there was as much as a factor of four difference between the IDCOR hydrogen generation, as calculated by MAAP, and the NRC hydrogen generation calculated with MARCH. MAAP assumes that hydrogen production is terminated when the cladding reaches a slumping temperature (usually 2300 K) and, in the BWR version, that fuel melting leads to complete blockage of the steam-flow. Both these questionable assumptions will be checked soon by data from the research program, particularly from the ACRR melt progression experiments. This factor of four difference in hydrogen generation has a major impact upon the probability of early containment failure from over pressure or hydrogen combustion, as treated in Section 3.2.2 of this report. The melt-progression research will also furnish a basis for modeling the core-melt mass, composition, and temperature distribution during slumping and melt progression into the lower plenum in place of the arbitrary assumptions currently made in MARCH and MAAP.

3.2.1.1.4 Planned Activities

Planned activities are illustrated in Figure 3.4. The major code activity during the FY 85 to FY 87 period is improvement of the models in the developing SCDAP and MELPROG codes, and assessment and validation of these models with the data that becomes available. These two codes have different capabilities and different applications. SCDAP, in detailed rod geometry, treats the state of the core during the earlier stages of core-uncovery transients and is particularly useful for analysis of accident recovery by core reflooding as at TMI-2. MELPROG treats the in-vessel progression of core melt, liquefaction, and relocation in



Note: QLR, Quick Look Report; TRR, Test Results Report; OOP, Out of Pile; PIE, Hot Cell Exam & Radiography Report; FPR, Fission Product Report; R, Report

Figure 3.4 Planned activities for in-vessel melt progression research core-uncovery accidents through attack on the reactor internal structure and the vessel, including the mode of vessel failure and the initial conditions for melt entry into the reactor cavity and the containment. Modeling to allow analysis of BWRs as well as PWRs will be added to both codes. Both will also have fission-product release and transport modules added that incorporate the models of the TRAP-MELT transport code. A special fission-product behavior module, VICTORIA, that will accommodate results of the current fission-product release research, is being added to MELPROG. Both codes will also be linked to appropriate thermal-hydraulic codes to incorporate into the analysis a very important aspect of in-vessel melt-progression behavior. Because of the internal structure of the codes, and for other reasons, SCDAP will be linked to the RELAP5 thermal-hydraulic code, and MELPROG will be linked with a 2-D version of TRAC to include the potentially significant effects of in-vessel natural convection on core melt progression. By FY 88, these codes in their MOD-2 versions will provide advanced tools for the mechanistic analysis of in-vessel melt progression behavior in severe LWR accidents. The governing models in these mechanistic codes will have undergone validation testing by data generated in the research program. Particularly for MELPROG, however, large-scale integral validation will not be possible, although useful integral data will be obtained from the TMI-2 core and vessel examination. The validation program for MELPROG will use small-scale experiments starting in FY 87. These codes can be used in direct detailed analysis of the more difficult and risk-significant accident sequences, analysis of experiments, and for benchmarking the advanced risk-assessment code MELCOR. At this time, the need for further analytical or experimental work to further reduce the uncertainties in the assessment of severe-accident consequences and risk will be examined.

The final PBF SFD test, SFD 1-4, with high burnup fuel, Ag-In-Cd control rods, and on-line aerosol diagnostics, was performed in FY 85. Analysis of results, Post Irradiation Examination (PIE), and preparation of reports will continue through FY 86 and FY 87.

The PBF results furnish a significant integral (multiple interactions) data base on in-vessel behavior under core uncovery accident conditions. These results include in-vessel fission-product release, chemical form, transport and deposition, and aerosol generation, with high-burnup fuel and control rod materials. Also included are melt-progression data on hydrogen generation and the state of the core during melt progression up to the approximately 2500K temperature at the end of the rapid steam oxidation transient.

Two NRU full-length coolant boildown tests to 2500K are to be performed in FY 86. These tests are primarily for validation testing at full length the modeling of the oxidation heating transient in SCDAP and MELPROG with its corollary hydrogen generation. This modeling is currently based on data from PBF and ACRR with short fuel bundles. The first test, FLHT-2, will use a coolant boildown transient to 2500 K for about 20 minutes to determine effects on fuel damage of extended time at high temperatures. A full-length high-temperature melt-progression and source-term model validation test with high-burnup fuel is planned for late FY 87. These full-length NRU tests will provide definitive integral data on oxidation and hydrogen generation for resolution of this issue. Future use of NRU is seen for accident management testing of recovered accidents with core reflooding.

The last of the ACRR fresh-fuel melt progression experiments, DF-4, will be performed in FY 86. This experiment will contain a BWR boron carbide control blade in the fuel bundle to provide data on the effects of such materials on core melt progression, aerosol generation, and on the chemical environment in the core which is important for fission product behavior. This experiment will furnish unique data for use in the BWR versions of SCDAP and MELPROG. In FY 87 and FY 88, two small integral Melt Progression (MP) experiments will be performed in ACRR to provide data on processes during the later stages of core-melt progression, including attack on the reactor vessel.

Special experiments will start in FY 86 for validation testing of the key models in the MELPROG code. Sensitivity studies with MELPROG will determine the phenomena that require new experimental data, and appropriate small in-pile or out-of-pile experiments will be performed. Phenomena that determine the maximum fuel temperatures reached during core-melt progression appear to be among the most important.

Throughout this period, data from the out-of-pile experiments on melt progression phenomena will become available, both from NRC work on zircaloy oxidation rates and the viscosity of liquified fuel, and from the work of our German SFD program partners on the thermodynamics and kinetics of the zirconium-uranium-oxygen system and from out-of-pile integral experiments on fuel rod oxidation and fuel liquefaction and relocation. Work performed in the new CORA facility at KfK in close cooperation with NRC in the joint international Severe Fuel Damage and Source Term research program will provide important data for assessment and validation of the models in SCDAP and MELPROG.

Data from the TMI-2 core examination will also become available throughout this period, and these data will be used to test the models in SCDAP and MELPROG.

3.2.1.2 Fission Product Behavior Research

3.2.1.2.1 Scope of Research

The fission product behavior research (Element 9 of NUREG-0900) is directed towards obtaining an experimental data base under a wide range of severe accident conditions. The information is necessary to judge the adequacy of Probabilistic Risk Assessments, to prepare Environmental Impact Statements, to determine whether changes in plant equipment and operations would be cost beneficial, and to provide a basis for improving the treatment of source terms in existing regulations. This data base, which will be used for model improvement and/or code validation is necessary in order to provide realistic source term estimates for severe accidents and to understand their uncertainties. The technical areas covered under this topic are:

1. In-vessel fission product and aerosol release,
2. Fission product and aerosol transport in the reactor coolant system,
3. Ex-vessel fission product and aerosol release,
4. Containment transport and attenuation of fission products.

These four technical areas are in chronological sequence in a severe accident, and thus cover the behavior of fission products from initial transient to in-vessel melt progression, from vessel failure to ex-vessel core-concrete interaction, and finally to containment failure. A directly related topic is the performance of Engineering Safety Features in attenuating fission products. However, this technical area and items 3 and 4 above will not be discussed here. A discussion of these topics will be given in the next section under Ex-Vessel/Containment Loads Research.

3.2.1.2.2 Research Accomplishments to Date

In-Vessel Fission Product and Aerosol Release - The final in-pile severe fuel damage test at PBF was conducted in FY 85. A total of four such tests have been conducted, two with trace irradiated fresh fuel and two with high burn-up fuel. Results analyzed so far have indicated that the release rates from the fresh and the high burn-up fuel are significantly different. Compared to the release rates in the CORSOR code, release rates from the PBF tests were at least one order of magnitude lower. This shows that the release rates are not simply a function of temperature, as assumed in CORSOR. In contrast, out-of-pile fission product release experiments being conducted at the Battelle Columbus Laboratory indicate that release rates measured from this program are comparable to those from CORSOR.

The difference in the release rates measured from the in-pile and out-of-pile experiments may be explained by factors that are thought to be important but have not been taken into account because of a lack of information. Changes in fuel morphology with burn-up and system pressure are two of these factors.

The NRC/IDCOR technical issues on tellurium (Te) retention by zircaloy cladding and on control material release are being resolved. Separate effects experiments at Battelle Columbus have shown that two Te retaining mechanisms are operative for the range of severe accident conditions of interest. At low temperatures, the formation of zirconium telluride was found to be the Te withholding mechanism, whereas, at high temperatures, the formation of tin telluride was favored. Tin is an alloying agent in the zircaloy cladding.

For the release of Ag-In-Cd control rod materials, results from the Oak Ridge Core Melt Experiments and from experiments at U.K. Winfrith and FRG Karlsruhe indicated that the amount of Ag aerosol release was relatively small. Cd was the only control material which was released in significant quantities.

The FASTGRASS code is a mechanistic code which was developed to predict fission product release from fuel for conditions up to and including fuel liquefaction. Recent improvements in the code include the incorporation of release models for Ba and Sr in addition to release models for I, Cs and Te. The code has also been assessed against all four PBF tests and most of the Oak Ridge HI series of tests. Comparable values were obtained for predicted and measured fission product release rates. FASTGRASS has also been incorporated in the SCDAP mechanistic core damage code designed for use in recovered accidents like TMI-2.
The VICTORIA in-vessel fission product and aerosol behavior code has been developed as a module of the MELPROG mechanistic in-vessel core-melt progression code. VICTORIA is a detailed mechanistic code that uses a first principles treatment of fission product behavior. Chemical transformation during release and transport and rate-limiting mass transport processes are considered. VICTORIA includes the mechanistic FASTGRASS fission product release modeling and the detailed TRAP-MELT fission product and aerosol transport methodology. In addition, it considers fission product release during the melt progression phase of the severe accident.

Fission Product and Aerosol Transport in the Reactor Coolant System - Scoping experiments to study the effect of radiation on fission product behavior during transport in the reactor coolant system have been conducted. Preliminary results indicated significant CsI decomposition in the presence of radiation and structural materials. Cs was found to be bound to the experimental apparatus while volatile iodide was generated.

The MARVIKEN-V program on the simultaneous transport of aerosols and volatile fission products was completed. Assessment of the TRAP-MELT RCS fission product and aerosol transport code with the MARVIKEN data has been carried out. Improvements in the TRAP-MELT code were also made. A turbulent impaction model has been added to the code to better predict aerosol deposition along pipe bends. TRAP-MELT has also been successfully linked with MERGE to accommodate calculations which require feedback between thermal/hydraulic parameters and fission product deposition models.

3.2.1.2.3 Discussion of Outstanding Issues

In-vessel Release - In addition to temperature, gas flow rates, gas composition, fuel burn up, pressure, and heating rate can play important roles in fission product release. Experimental results from the severe fuel damage tests at PBF yield fission product release rates for the volatiles at least an order of magnitude lower than the CORSOR rates or the rates obtained from the out-of-pile experiments at ORNL and BCL. Fuel burn-up and system pressure are two of the factors which have not been studied previously but are believed to have contributed to the lower releases in PBF. Additionally, fission product release rates for temperatures beyond 2400 K have not been determined. These measurements are necessary to model fission product release during melt progression processes.

Model development for the FASTGRASS fission product release code is essentially completed, but the Ba and Sr models need to be validated against relevant experimental results; namely, higher temperature data where the release of Ba and Sr are significant.

The effect of boron carbide control rods on fission product chemistry needs to be assessed. B_4C can be oxidized by steam to form a variety of products which subsequently react with CsI. The result is the decomposition of CsI and the evolution of volatile iodides which are more difficult to retain in the RCS and containment.

Transport in Reactor Coolant System - The competitive deposition of fission products on suspended aerosols and on structural material surfaces has not been investigated. Volatile fission products may escape the RCS by depositing on aerosols rather than on walls. The kinetics of the fission product/wall interaction for some of the volatiles have been determined experimentally, but those for the fission product-aerosol interaction have not been measured.

Although preliminary results on the effect of radiation on CsI chemistry have been obtained, additional tests are necessary to study the effect over a wider range of experimental conditions. The effect of radiation on Te deposition chemistry also needs to be examined.

The revaporization of previously deposited fission products from the RCS structural material surfaces has been a technical issue between NRC and IDCOR for some time. IDCOR believes the issue to be unimportant because the heat generated by the decay of the deposited fission products will be efficiently conducted away to the containment and consequently, will not be available to revaporize the deposited materials. NRC believes the importance of this issue lies in the timing of revaporization. Source terms for delayed containment failure accidents are expected to be relatively small. If revaporization occurs at this time, however, the source term will be increased and the consequences increased. An integrated analysis is required to assess the impact of this issue.

Resuspension of deposited aerosols in the RCS and containment is also an issue of concern. During failure of the reactor pressure vessel or containment, significant turbulence may be generated such that the fission products and aerosols deposited in the upper plenum and containment may be resuspended. There is currently little data in this area to model the process.

The TRAP-MELT fission product and aerosol transport code has not been validated at high velocity conditions. These conditions are relevant to the containment bypass accident sequence (Event V).

3.2.1.2.4 Planned Activities

In-Vessel Release - A schedule of programmatic activities is provided in Figure 3.5. As mentioned previously, release data at high pressures, higher temperatures (>2400 K), oxidizing and reducing environments, and with fuels of various burn-ups have not been determined. The programs addressing these needs are the out-of-pile experiments at ORNL and BCL, the in-pile separate effect experiments in ACRR and the integral experiments in PBF and NRU. The PBF severe fuel damage tests have been completed but sample and data analyses will continue. Investigation of the effects of temperature, burn-up and other parameters at ambient pressures will be the focus of the ORNL tests, while release at high pressures will dominate the BCL experimental matrix. Two separate-effect fission-product release experiments in ACRR will provide confirmatory data under in-core accident conditions, including particularly the effects of radiation and pressure. Along with the much improved new laboratory results, these experiments will provide a definitive answer the reality of the large and significant difference between existing in-pile and out-of-pile release data. An NRU experiment will provide integral confirmation of the in-pile and out-of-pile separate effect experiments. VICTORIA validation will be carried out as results from the above programs become available.

Experiments are being conducted at the ORNL Core Melt Facility to study the extent of boron carbide control rod oxidation by steam and the subsequent reaction with CsI in the presence of core materials. Confirmatory data will also be obtained in-pile at the ACRR Facility (DF-4 test).

Transport in Reactor Coolant System - The High Temperature Fission Product Chemistry program at SNL will continue to obtain kinetic data on the interaction between volatile fission products and aerosols. Investigation of the effect of radiation on fission product retention in the RCS will also be part of the work scope of this program. Integral calculations to assess the impact of fission product revaporization on severe accident source terms are being planned and would likely fall under the scope of this program.

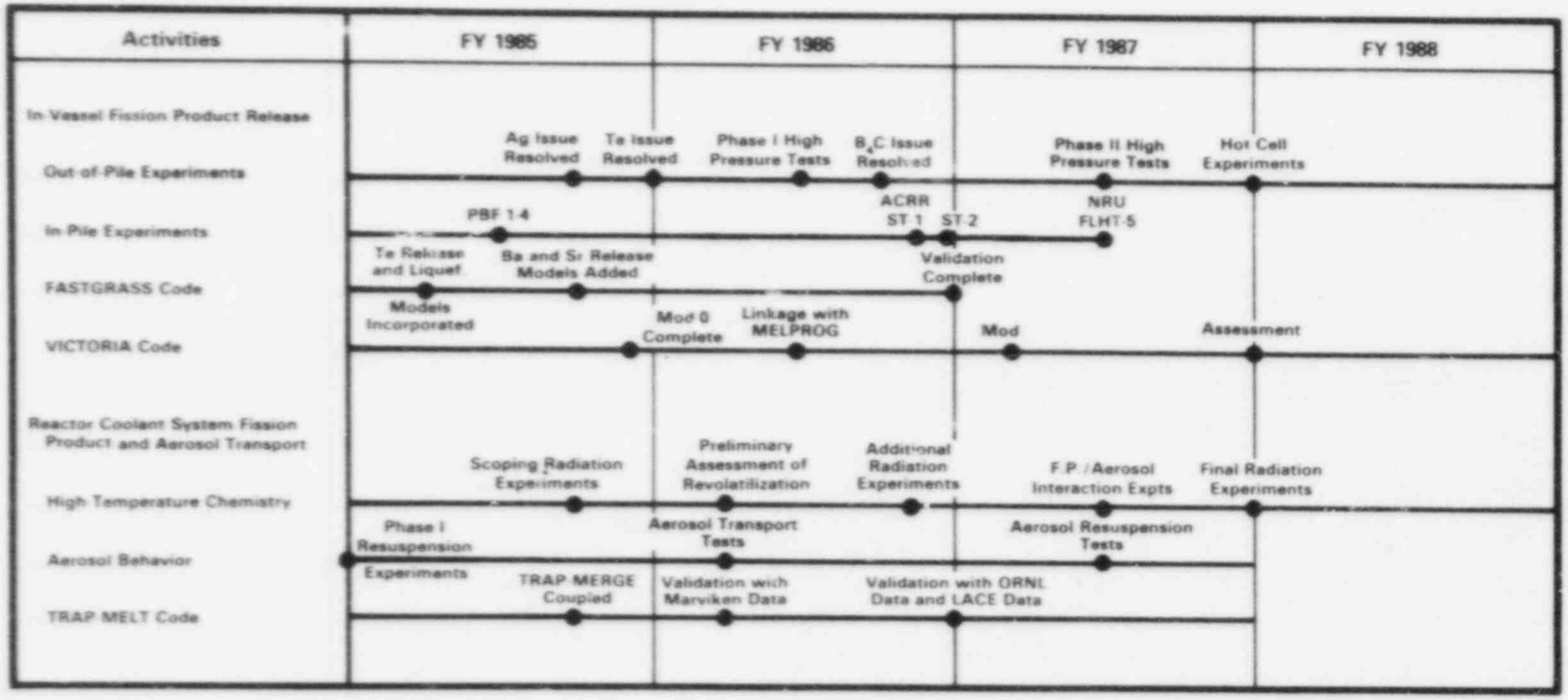


Figure 3.5 Planned activities for fission product behavior research

The last of the ORNL Aerosol Transport Tests will be conducted in the first half of FY 86. The program will be subsequently redirected to focus its efforts on the study of fission product and aerosol resuspension.

TRAP-MELT validation under high velocity conditions will be carried out by BCL with data obtained from EPRI's LWR Aerosol Containment Experiments (Tests LA1 and LA3).

3.2.2 Ex-vessel/Containment Loads Research

3.2.2.1 Containment Loads Research

3.2.2.1.1 Scope of Research

This part of SARP (Elements 5, 6 and 7 NUREG-0900) includes investigation of the processes which may challenge the integrity of the containment systems in a severe reactor accident. These challenges are:

1. Containment overpressurization because of:

   a. Steam generation resulting from interaction of the hot core with water,

   b. Direct heating of the containment atmosphere resulting from chemical and thermal interactions of the dispersed melt particles in the containment atmosphere,

   c. Production of noncondensible gases resulting from concrete ablation and cladding oxidation,

   d. Combustion of combustible gases (H_2 and CO) generated during concrete ablation and cladding oxidation,

   e. Heating of the atmosphere by decay of airborne fission products.

2. Melt-structure interactions such as basemat penetration, or thermal attack on containment structural components, e.g. reactor pedestal.

3. Degradation of engineered safety features, e.g. plugging of the containment cooling system or the containment spray system by aerosols.

4. Overheating of structures or structure penetrations.

5. Missile production as the result of steam explosions or vessel failure.

6. Transport and spatial distribution of hydrogen; various hydrogen burning modes including ordinary deflagrations, diffusion flames, flame acceleration, transition from deflagration to detonation, and global and local hydrogen detonations.

This research includes experimental and analytical programs aimed at development of tools with which the threat to the containment and the fission product source term can be predicted for given accident scenarios and representative containment systems.

3.2.2.1.2 Research Accomplishments to Date

Core Melt Technology Experiments - A series of experiments has been performed at Sandia National Laboratories (SNL) to study the behavior of molten-core materials released from the reactor vessel under high or low pressure, with or without the presence of water. The earlier tests used molten steel to simulate molten core materials. The more recent tests use inductively heated molten corium and show drastically different heat transfer characteristics because of early crust formation. Quantities of the test melts ranged from 10-80 kg for the high pressure tests, and up to 200 kg for the low pressure ones as compared to several tons expected for reactor accidents. Some wet tests were conducted with a water filled cavity and some with the addition of water after the melt was released from the reactor vessel.

High pressure ejection tests were conducted to investigate the melt-dispersal hypothesis predicted in the Zion Probabilistic Safety Study. Preliminary results not only confirmed that fuel debris would be swept from the reactor cavity but also will cause direct heating of the containment atmosphere that could threaten containment integrity.

Core/Coolant Interaction Experiments - SNL has conducted experimental research on the molten core-coolant interaction (MCCI) using masses of core simulant material ranging from a single droplet up to 20 kg. During the past two years 25 tests were conducted. These tests evaluated ten independent variables and provided a valuable data base to assist in the development of models for this highly uncertain physical phenomenon. Of the 25 tests, 13 (52 percent) resulted in steam explosions, 8 (32 percent) resulted in eruptions while the remaining 4 (16 percent) resulted in benign film boiling. Although models have been developed to predict the energetics of steam explosions, the ability to extrapolate the model to full-scale based on the existing data base is subject to significant uncertainty.

Containment Analysis - The computer code CONTAIN 1.0 is the pivotal point of the analytical program intended to develop mathematical tools to predict and quantify the abnormal loads imposed on containment systems under severe accident conditions. CONTAIN calculates best estimate predictions for the temperatures, pressures, thermodynamic and compositional details of the atmosphere as well as aerosol and fission-product behavior throughout a coupled network of compartments in the containment system. CONTAIN 1.0 is available for use and has been tested successfully against experiments both here and in Europe. The recent addition of an implicit numeric solver has improved running time by a factor of 100.

The computer code CORCON MOD2 is a mechanistic, best-estimate code available for direct application or for benchmarking of faster-running, risk analysis codes such as MARCH, or MELCOR. It treats molten-core concrete interactions in the reactor cavity, generation of water vapor, CO_2 and combustible gases, and crust formation and freezing. CORCON has been used in support of the Source-Term Reassessment Study and the Containment Loads Working Group effort. CORCON is being used for the planning and analysis of experiments at the SNL Large Melt Facility as well as for the large-scale test facility, BETA, in West Germany.

Thermal-Hydraulic Experiments - This program (at BNL) includes analysis and experiments. Models for heat transfer in gas bubble mixed debris pools were developed, experimentally substantiated, and incorporated into the CORCON code. This information contributed significantly to the Containment Loads Working Group study, the Source-Term Reassessment project, and the joint NRC-IDCOR safety issue reviews. The experimental data and modeling have been used in the development of MARCH 2 and the advanced risk-assessment code, MELCOR. New insights concerning energetic interactions between water and molten metals have stimulated review and some reorientation of the large-scale steam explosion program at SNL.

Hydrogen Behavior and Control - A hydrogen compendium (NUREG/CR-1561) was written to provide a resource describing hydrogen behavior during accidents in LWRs. A LWR hydrogen manual (NUREG/CR-2726) which provides general guidance in developing plant specific procedures for handling hydrogen during normal and off-normal conditions. Analytical methods have been developed for analyzing hydrogen and/or steam transport in containment and these methods have been successfully applied to experiments conducted by EPRI in the Containment Safety Test Facility at HEDL, experiments at Battelle Frankfurt sponsored by the BMFT (NUREG/CR-2764 and NUREG/CR-3463), intermediate scale facilities at SNL, and the large-scale facility at the Nevada test site.

Experimental data on premixed hydrogen deflagrations in air and steam has been accumulated in recent years as a result of research sponsored by NRC (NUREG/CR-4136 and 4138), EPRI, and others. The NRC effort (NUREG/CR-3721 and 3273) has provided data needed for model and code development. The large tests at the Nevada Test Site have provided data at a sufficient scale to aid in the validation of the hydrogen combustion codes. A major accomplishment over the past two years has been the development of the HECTR code (NUREG/CR-3913) for the analysis of hydrogen combustion events in various reactor containment types. The HECTR code has been applied to the analysis of selected accident scenarios for an ice condenser plant (NUREG/CR-3912) and as part of a preliminary assessment of the adequacy of the deliberate ignition system for the BWR Mark III Grand Gulf plant (NUREG/CR-2530). The models in the code serve as the basis for the hydrogen burn models in the containment systems code CONTAIN and the advanced PRA code MELCOR.

Our understanding of the conditions under which diffusion flames might occur and the stability of these flames has improved. The NRC has sponsored some experimental work on the burning of hydrogen/steam mixtures flowing at high velocities (NUREG/CR-3638). The EPRI program at NTS has provided some data on the burning of hydrogen/steam mixtures during continuous injection. The Hydrogen Control Owners Group (HCOG) have completed experiments at 1/20 scale and are currently performing tests in a 1/4-scale facility representative of a Mark III containment. It is anticipated that these results will enhance our understanding of combustion behavior in a Mark III containment and may provide a sufficient experimental basis for diffusion flame modeling in reactor accidents. Because of the complexities in modeling diffusion flames due to geometric considerations, fluid dynamics, and other parameters, two separate modeling efforts have been pursued. One is the development of the detailed three dimensional finite difference code HMS-BURN for benchmark analyses, and the second is a simplified one dimensional model for the HECTR code.

Hydrogen has a high potential for flame acceleration and transition from a deflagration to a detonation. Research performed in the U.S., Canada, Germany, and Norway over the past two years has vastly enhanced our understanding of flame acceleration and the potential for the transition to detonation. Recent experiments in the FLAME facility at Sandia, experiments at Battelle-Frankfurt and at the Fraunhofer-Gesellschaft indicate a much higher potential for the transition to detonation; additionally they indicate that scaling is possible based on theoretical considerations. The analytical efforts in support of these experiments has included the existing code (CONCHAS-SPRAY) and other models in an attempt to understand the effects of obstacles and turbulence on flame acceleration with some success.

Significant progress has also been made on understanding the likelihood and potential for hydrogen detonations. Experimental work sponsored by the NRC has resulted in establishing lower values for the concentration limits for a hydrogen detonation in air as less than 13.5 percent hydrogen in air. Analytical work on hydrogen detonations has included an assessment of the containment response of Zion and Sequoyah (NUREG/CR-2385) and selected areas in the Grand Gulf containment.

In the area of hydrogen burn mitigation studies, several options have been reviewed (NUREG/CR-1762, 2767, and 2865) and, aside from a deliberate ignition system and a passive catalytic igniter, most have been eliminated because of engineering considerations, cost, or other negative effects. The research program has contributed to the assessment of deliberate ignition systems for degraded core accidents for ice condenser and Mark III containments. The program is currently studying the effects of condensing steam on igniter operation.

3.2.2.1.3 Discussion of Outstanding Issues

A number of NRC/RES activities during the past year have contributed significantly to a heightened appreciation, evaluation, and refocusing of the Severe Accident Research Program. Analysis of data and a critical review of the computer codes used in the Source-Term Reassessment Study surfaced several safety areas needing further intense research. Results of the QUEST sensitivity study accented these conclusions. Corresponding guidance for future research was also derived from the Containment Loads Working Group activities as well as the mutual exchange between NRC and IDCOR staffs on the complete spectrum of severe accident safety issues. Key areas requiring further research for resolution include:

Transient Phenomena Following Vessel Failure - The dispersal of debris at the time of vessel failure is poorly understood; there was consensus on this issue at the NRC-IDCOR discussions. The nature of dispersal depends not only on the vessel pressure and temperature, but also on the geometric configuration of the reactor cavity and is therefore plant dependent. Debris bed coolability and potential overpressurization by direct heating of the atmosphere depend on the dispersal process. Further experimental work is necessary to clarify these transients and to provide data to support the modeling and validation of the concomitant analytical tools.

Steam Explosion Behavior - Because of the potentially high consequences that could be associated with an in-vessel steam explosion that would result in containment failure, interest in steam explosions has primarily focused on the likelihood of this combination of events. Actually, steam explosions that do not lead to containment failure but otherwise alter the progression of the accident could also be important. The subject of steam explosions was reviewed extensively by the Steam Explosion Review Group (SERG). The findings of that review group were published in June 1985 as NUREG-1116. The major conclusions reached by that group are:

1. The occurrence of steam explosions of sufficient energetics which could lead to direct containment failure (i.e., the WASH-1400 alpha-mode failure) has a low probability. This conclusion is reached despite the expression of differing opinions on modeling of basic steam explosion sequence phenomenology.

2. The SERG members disagreed with the assumed distributions for key phenomena used in NUREG/CR-3369 "An Uncertainty Study of PWR Steam Explosions" dated May 1984 and nearly all disagreed with the SNL conclusion that "Indeed the results (for the conditional probability of containment failure given a core melt) span the range from 0 to 1."

3. A consensus was reached among SERG members on the need for continuing steam explosion research. They supported intermediate scale experiments (20 Kg to 100 Kg) and investigations of stratified contact modes but could not reach a consensus on the need for large scale (up to 2000 Kg melt) experiments. The program discussed in later sections follows these recommendations.

Molten Core Technology - Larger-scale tests on core-concrete interactions with prototypic materials are needed for validation of the CORCON code. Accompanying efforts to model and experimentally validate tools (VANESA) for predicting ex-vessel fission-product release and aerosol generation are necessary. Uncertainties in the modeling of core-concrete interactions affect both the prediction of loads on the containment as well as the source term of fission products released to the containment.

Containment Loading Analysis and Code Development and Application - Refinement and validation of the CONTAIN code are required to permit its application to specific issues such as aerosol deposition and decay heating, transport of fission products via liquid pathways, condensation of water on aerosols during slow containment depressurization, ESF recovery, and high velociy gas entrainment. Completion of the ESF modeling is essential to provide adequate analysis capability for BWR systems.

Hydrogen Behavior - Hydrogen combustion is a significant contributor to the risk of early containment failure for Mark III and ice condenser plants and can significantly influence containment integrity in sub-atmospheric and large dry PWRs when coupled with containment pressurization from steam.

With regard to hydrogen behavior during core melt accidents, there are significant differences between IDCOR and NRC staff positions on hydrogen issues.

Outstanding issues related to hydrogen behavior which affect reactor safety can be roughly grouped into three interrelated categories (1) hydrogen transport, (2) hydrogen combustion phenomena, and (3) hydrogen mitigation.

1. In the area of hydrogen transport the principal issue is the question of hydrogen mixing in the absence of strong forced or natural convection forces or in the presence of a temperature inversion. This is a potential problem for the subatmospheric and large dry PWRs which are

not required to provide hydrogen control for severe accidents. The consequence of poor mixing could lead to the formation of volumes with detonable mixtures.

2. In the area of hydrogen combustion phenomena there are outstanding technical issues related to the understanding of hydrogen deflagrations, diffusion burning, flame acceleration and detonations.

   In the area of diffusion flames there are a number of unresolved issues: these include autoignition and the likelihood and consequences of momentum-dominated flames.

   The technical issues on flame acceleration requiring some resolution include, effects of containment structures and geometry on burn rates, the effects of ESFs on flame acceleration and the effects of flame acceleration on ESFs, effects of gas composition and ignition requirements, the magnitude of dynamic loads from flame acceleration and lastly, the potential for transition to a detonation.

   The outstanding technical issue associated with detonations is an assessment of the potential and consequences of local detonations. For plants with no deliberate ignition system (e.g., large-dry and sub-atmospheric containments) the potential and consequences of global detonations need research. This includes assessing the effects of large local dynamic loads on seals, penetrations, the containment shell, ESFs and safety related equipment and evaluating the potential to generate missiles. Other issues include improving our understanding of the effects of temperature and gas composition (H_2/CO/CO_2/Steam), turbulence and ignition requirements on detonation limits.

3. In the area of hydrogen mitigation the outstanding technical issues relate to the effectiveness and risk reduction benefit of a deliberate ignition system for core melt accidents and degraded core accidents involving loss of all A/C power, where currently installed and proposed deliberate ignition systems would not work. Additionally the mitigation research needs to address current and proposed emergency procedures (e.g., should igniters be activated after a station blackout accident when the hydrogen concentration could be high in containment) to provide some guidance to the regulatory staff. Research needs to be completed on assessing the efficacy of igniters in the presence of condensing steam. The effectiveness and risk reduction benefits of a hydrogen control system for large dry containments and possible mitigation strategies for detonations should be addressed. For severe accidents, preliminary results indicate igniter performance in the presence of aerosols is not reduced. However, these scoping experiments do indicate that aerosol chemistry can be changed as the result of a hydrogen burn, which might result in an increased fission product source term.

3.2.2.1.4 Planned Activities

Further tests are planned at SNL to augment the data base for the long-term behavior of molten core debris collected in a reactor cavity; they include Sustained Urania-Concrete Interaction (SURC) tests and Hot Solid Concrete Interaction tests. All tests are scheduled to be completed in FY 86; but analyses of the test data and development of analytical models will be extended into FY 87. Sandia's Large Melt Facility (LMF) will be used to produce molten corium needed for these tests. However, a method will have to be developed to simulate decay heat in the core material. Projected activities are shown in Figure 3.6.

High-pressure ejection tests in the SURTSEY facility will begin early in 1986 and will continue into 1987. These tests seek to quantify the influence of melt properties on debris dispersal characteristics (particle size distribution, amount dispersed, etc.) The main issue addressed by the high pressure ejection tests is rapid heating of the containment atmosphere. Dynamic effects of the pressure wave on containment integrity will be studied in tests conducted in a simulated containment. The tests will begin in the second quarter of FY 86 and extend into FY 87. If these tests confirm the hypothesis that the results are highly sensitive to plant configuration, the study will be extended to a number of representative plant configurations in FY 87.

Molten Core - Coolant Interaction (MCCI) - The experimental research activities for the MCCI Program are based on the following:

Designed experiments will investigate the effects of test conditions on the rate at which energy is transferred from the fuel to the coolant, the efficiency of this conversion (conversion ratio), and the fraction of the molten corium which actually participates in the explosive interaction. It has been postulated that flow diverters and other structural members in the lower plenum of a PWR will change the character of the interaction. This hypothesis will be examined by using perforated plates just below the water level in the tests. Other tests will examine the importance of contact mode (melt into water or water into melt), vessel pressure at the time of contact, and the need for an external pressure source to trigger the explosion. This program will complete the present phase during FY 1987. At that time it will be reviewed to identify residual problems and to justify further work.

To extend the analytical capability of CONTAIN 1.0 and permit its application to the resolution of outstanding issues, the following tasks will be pursued: (a) improve the numerical algorithms in the intercompartment flow model to increase the code efficiency and permit the evaluation of multi-dimensional effects, e.g., containment atmosphere stratification, hydrogen combustion, and ESFs (ice condensers); (b) develop and refine new and/or missing models to ensure the comprehensive capability of CONTAIN 1.0 so that all existing reactor systems (PWR and BWR) can be analyzed; (c) complete and distribute the CONTAIN 1.0 reference manual (to supplement the user's manual) so that the mathematical details, the phenomenological models, and the numerical algorithms used for their solution are readily available to users.

Further research in the area of molten-core concrete interaction modelling will include the following: (a) in collaboration with the SURC and Hot Solid programs, develop and validate reliable freezing models and incorporate these into CORCON MOD2; (b) incorporate the VANESA fission product release model into CORCON MOD2 and test against experimental data from core-concrete tests doped with simulated fission products; (c) a compressed version of CORCON must be written, tested, verified, and installed in the MELCOR code; (d) validation of CORCON MOD2 should be continued; the code will be applied in the planning and execution of the related experimental programs.

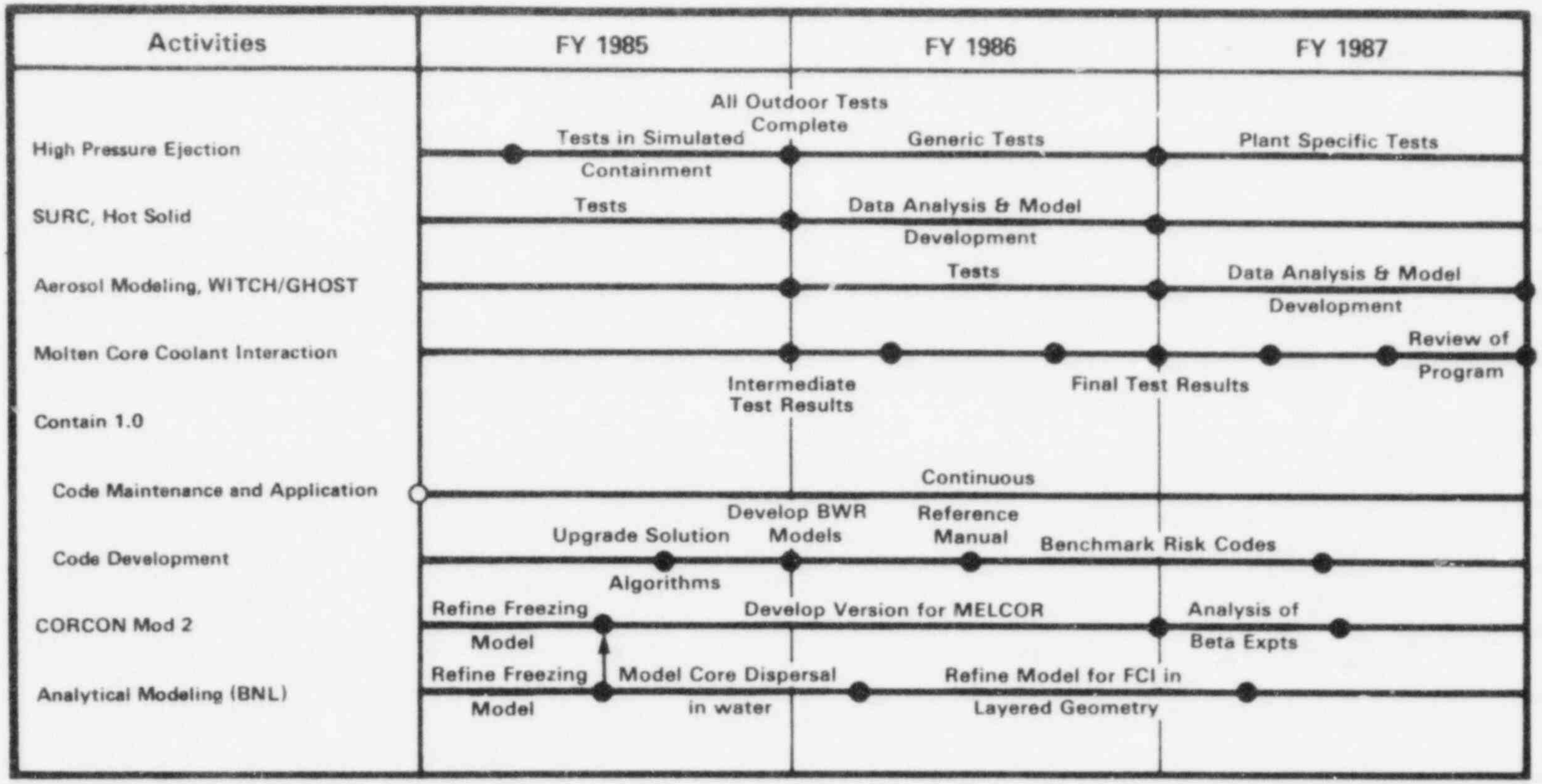


Figure 3.6 Planned activities for fuel-structure interaction and containment analysis research

Small-scale simulant experiments together with concomitant model development will continue at BNL to clarify unresolved issues related to steam generation rates, debris bed formation and coolability, and energetic fuel-coolant interactions. These include: (a) study the mechanisms that determine the dispersal of hot particulate (or molten) debris as it enters the water from a failed vessel, and determine the steam generation rates that accompany such processes; (b) investigate pool surface boiling phenomena over a range of anticipated accident conditions; (c) study crust formation and other freezing mechanisms, develop models, and incorporate this improved understanding into the CORCON code; and (d) investigate the mechanisms that influence the initiation and propagation of fuel-coolant interactions that occur between pools of molten simulants and overlying layers of water.

Projected activities for all the preceding are shown in Figure 3.6.

Hydrogen Behavior - Analysis of hydrogen burning in a typical large dry and a sub-atmospheric containment will be done in FY86 using the HECTR and HMS-BURN codes. A scoping analytical study of the IDCOR postulate of hydrogen/CO/CO2 burning for an ice condenser plant will be completed in FY86.

With the completion of the HCOG program an assessment will be made of the need for any additional experimental work on diffusion flame phenomena. Analytical work will focus on the incorporation of a diffusion flame model into HECTR in FY86 and the application of the code to Grand Gulf and the HCOG experiments. Additionally, HMS-BURN will be used to perform a number of benchmarking calculations for selected Grand Gulf accident sequences and HCOG experiments, in FY86. Results from the work on deflagrations and diffusion flame studies, along with the results from the other areas below, will be used in resolving USI A-48.

Flame acceleration and the transition to detonation experiments in the FLAME facility will be completed in FY 86. These tests will include the study of venting, obstacles, turbulence and an attempt to simulate a few specific plant geometries. Limited comparisons will be made for the codes under development with selected experiments FY 86. The plan is to develop enough understanding of the phenomena that, with engineering judgment, assessments can be made for selected accident sequences in selected plants.

Experimental work on detonations will be completed in FY 86. This work is very important in providing a building block for an assessment of local detonations in specific plant/containment types. An analytical and engineering assessment of the potential for local and global detonations will be completed in FY 86.

Mitigation studies on the efficacy of the deliberate ignition system and of a passive catalytic igniter will be completed in FY 86 with a final report and recommendation on the catalytic igniter. Also in FY 86 a final report will be issued on the effect of sprays and fans on igniter operation.

Projected activities are shown in Figure 3.7.

3.2.2.2 Ex-Vessel Fission Product Behavior Research

3.2.2.2.1 Scope of Research

In-vessel Fission Product Behavior Research and the overall scope of Fission Product Behavior Research are discussed in Section 3.2.1.2. This section

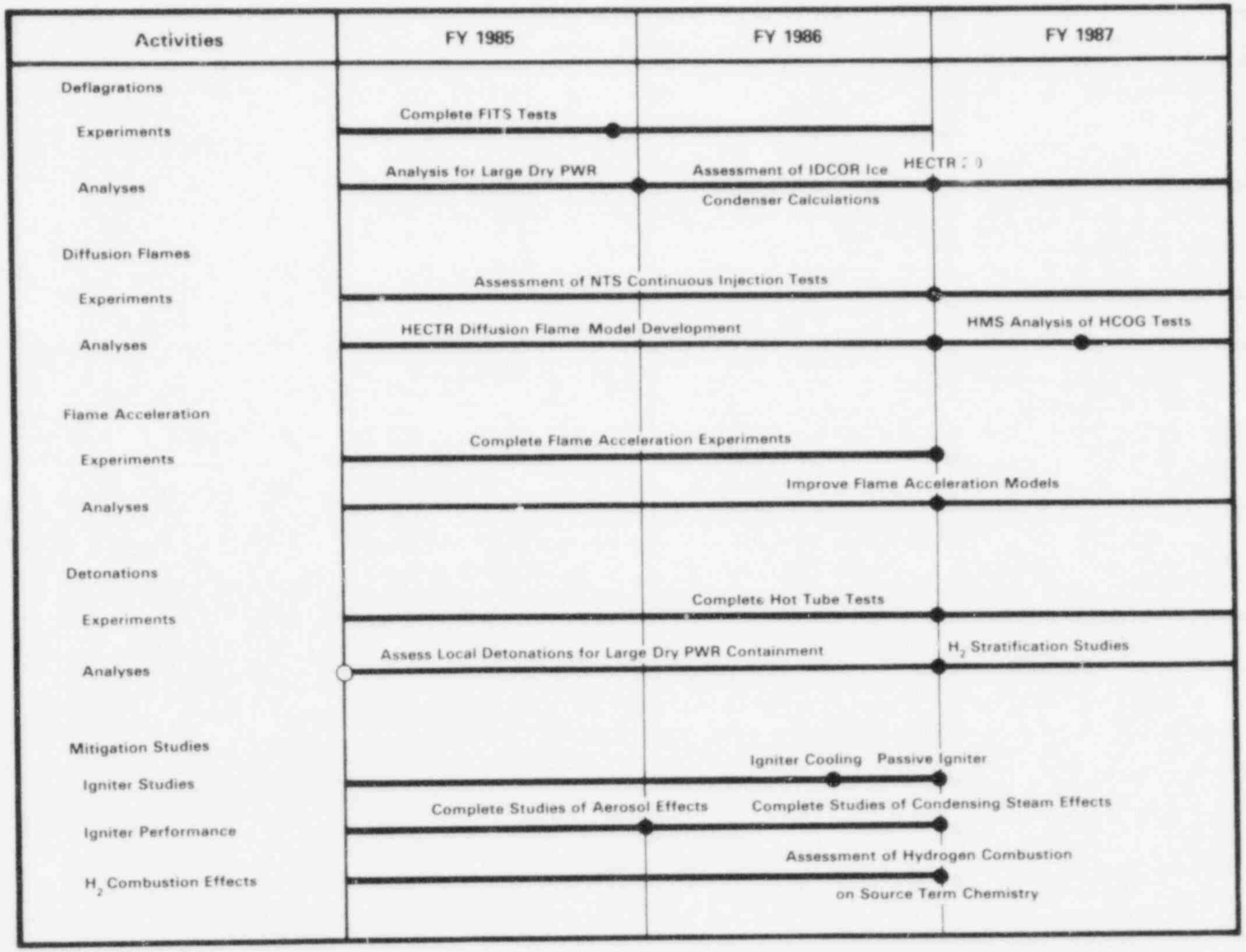


Figure 3.7 Planned activities for hydrogen behavior research

addresses ex-vessel fission product and aerosol release, transport of fission products and aerosols in containment, and attenuation of fission products by aerosol settlement and Engineered Safety Features.

3.2.2.2.2 Research Accomplishments to Date

Ex-Vessel Fission Product and Aerosol Release - The CORCON code models the interactions between core debris and concrete in the reactor cavity. MOD2 of the code was released in August 1984 and was improved as a result of close NRC-FRG cooperation at the large-scale BETA facility at KfK, Karlsruhe. CORCON predicts basemat penetration, evolution of combustible and noncondensible gases, and thermal energy which, in turn, impact containment loading. The VANESA code, which requires CORCON output parameters as input data, models the release of fission products from the core debris to the atmosphere. Two mechanisms contribute to the generation of aerosols at the pool surface which, in turn, transport the fission products into the containment and thus determine the radiological source term. The two mechanisms are (1) condensation of vapors in gas bubbles as they leave the pool surface, and (2) fragmentation of the bulk material when the bubbles burst at the surface. Preliminary experimental validation has begun through comparison of VANESA calculations with data from the TURC & SWISS tests.

Fission Product and Aerosol Transport in the Containment - Single and two-component aerosol behavior were studied in the Nuclear Safety Pilot Plant (NSPP). Results showed that iron oxide and uranium oxide aerosols agglomerate well and that steam enhances gravitational settling of single as well as two-component aerosols except those that include concrete aerosols. Concrete aerosols do not appear to agglomerate with iron oxide. Steam seems to spheroidize the concrete agglomerates but it has little apparent effect on the settling of these aerosols. The reasons for this behavior are under review.

Containment Iodine Behavior - Experiments have been performed at ORNL to provide data to develop models for the TRENDS code to predict containment iodine behavior. Results from these separate effect experiments indicate significant generation of molecular and organic iodines from an aqueous solution of iodides and impurities at various pH conditions and in the presence of a radiation field. The results are applicable to containment sump and suppression pool conditions.

Engineering Safety Features - To model the effectiveness of engineer safety features on the retention of fission products and aerosols, the SPARC and ICEDF codes were developed to estimate fission product and aerosol removal in BWR suppression pools and ice condenser pressure suppression system respectively. FY 85 efforts results in modifications of both codes to better predict particle growth in the presence of condensible vapors while considering supersaturated environments. For the SPARC code, new correlations for hydrodynamic behavior of bubble swarms as well as a model for hydrodynamic entrance effect were also added which subsequently provide improved model/data comparisons.

3.2.2.2.3 Discussion of Outstanding Issues

Ex-Vessel Fission Product and Aerosol Release - The influence of chemical effects within the molten core debris influence the volatility, and thus the magnitude of fission product aerosol release due to vapor condensation as predicted by the VANESA code. The level of confidence attributable to CORCON/VANESA predictions is sensitive to such chemical effects and therefore significant uncertainties will persist until the experimental data base is expanded and factored into the model development program.

Containment Transport - Qualitative information obtained to date indicates that steam transforms chain-like aerosols aggregates into spherical forms. Aerosol shape governs the aerodynamic behavior of the aerosols and, consequently, their removal rates in the containment. The sensitivity of the source term calculations to uncertainties in the aerosol shapes for containment failure cases were demonstrated in the recent QUEST calculation. The results support the need for additional effort to obtain quantitative information on the aerosol shape factors as a function of system humidity and steam condensation conditions. Other directly related issues include the functional relationships among aerosol fallout behavior and the set of independent variables of interest, and changes in measured responses (e.g., mass concentration, aerodynamic size distribution, fallout time, microscopy), due to quantitative changes in initial conditions.

Containment Iodine Behavior - As mentioned earlier, functional dependencies of volatile iodine generation on pH, concentration and other parameters have been obtained for most of the process occurring in the containment and suppression pool. However, the possibility of volatile iodine evolution from the decay of TE^{132} or other fission product isotopes still need to be examined. Furthermore, model development activities to estimate the iodine source term in the containment have not been completed.

Engineered Safety Features - The SPARC and ICEDF codes were developed to predict fission product retention in suppression pools and ice condensers, respectively. At present, differences exist in the suppression pool decontamination factors calculated by the IDCOR/GE model and by SPARC. It is agreed that there is a need to validate these codes and the EPRI code SUPRA. The ICEDF code is the first ice condenser code developed to predict aerosol retention. Separate effects and integral experiments are needed to validate some of the basic model assumptions and the overall integrated code.

3.2.2.2.4 Planned Activities

A schedule of activities is provided in Figure 3.8.

Ex-vessel Release - The validation of VANESA is continuing in several programs which are discussed under the topic of "Ex-vessel Core Behavior and Containment Loads." Of particular interest are tests which involve UO_2/ZrO_2 melts, doped with fission product elements, and in containment with concretes. Separate effects tests and tests with an overlying pool of water are also being planned.

Containment Transport - One limited series of experiments is planned to obtain quantitative information on phenomena which affect the characteristics and behavior of LWR aerosols.

These experiments will be conducted in the Aerosol Moisture Interaction Test Vessel (AMIT) at ORNL using a single aerosol component (FE_2O_3). The aerosol behavior questions will be answered for the independent variables: relative humidity, aerosol concentrations, and re-suspension velocities.

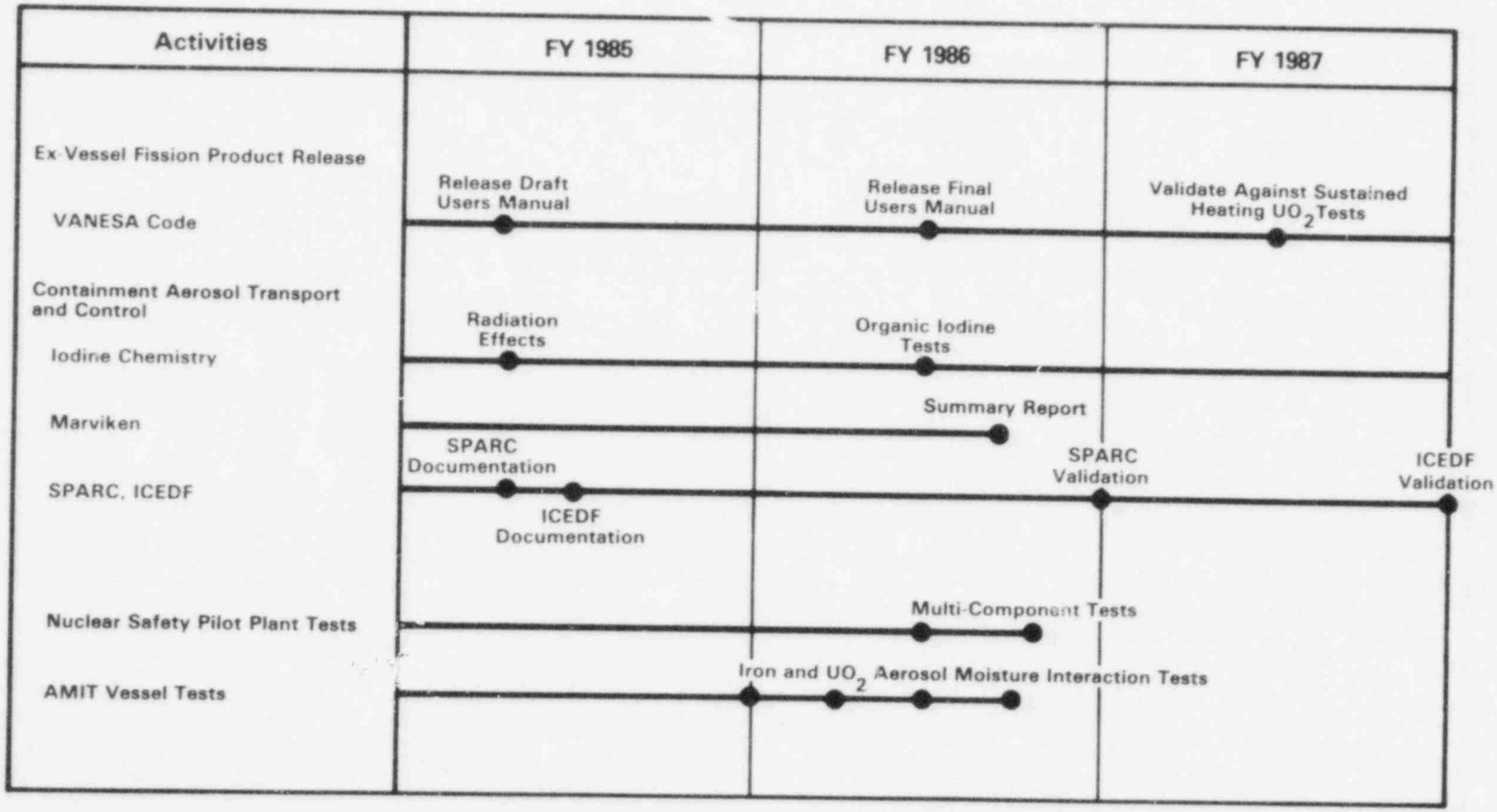


Figure 3.8 Planned activities for ex-vessel fission product and aerosol research

Containment Iodine Behavior - The iodine chemistry program at ORNL will continue to address analytically and/or experimentally the issue of volatile iodine generation from the decay of TE^{132}, and also to incorporate models in the TRENDS code for the prediction of iodine concentration in the containment atmosphere.

Engineering Safety Features - Since code development activities on SPARC and ICEDF are coming to an end, code validation will be the major focus in this program in FY 86 and 87. Validation of the SPARC code continues assuming that adequate experimental data has been released by others. For the validation of ICEDF, engineering scale unit cell tests designed to simulate a mixture of steam, air and aerosols passing through four ice basket columns are planned.

3.3 Containment Behavior Research

3.3.1 Containment Performance Research

3.3.1.1 Scope of Research

The major source of risk to the public from the operation of nuclear power plants stems from accident scenarios that lead to a containment failure. The regulatory concern is that the failure modes and associated load levels for containment structures cannot be predicted with any real confidence by the methods used for design. This is especially so if the contemplated failure mode is localized leakage. Both assessments of the risk posed by loads outside the design basis and estimates of the effectiveness of proposed mitigative steps require an ability to predict the way in which a containment will fail.

Research on containment failure modes is based on tne observation that excessive leakage can occur, basically, from four sources:

1. Failure of the shell, either the containment shell itself, in the case of steel containments, or the liner, in the case of concrete containments;

2. Leakage at large penetrations as a result of inelastic deformations, and/or degradation of seals and gaskets.

3. Leakage at electrical penetrations due to degradation of materials under the high temperatures associated with accident scenarios; and

4. Leakage through valves due to pressure and temperature effects.

3.3.1.2 Research Accomplishments to Date

Efforts to date on containment shells and major penetrations include: the development of a pressure testing facility for containment models, the testing of small steel models and a large steel model. A series of full-scale tests of electrical penetrations has been started. An existing facility has been modified for these tests. Also tests on seal and gasket materials used in valves and electrical penetrations are near completion. Pressure tests have been performed in the valve test facility on three purge valves. Summaries of these activities are provided below.

Model Tests - Four 1/32 size steel models of three configurations were built and tested at Sandia National Laboratories. The first configuration, of which there were two models, was termed a clean shell. Geometrically the clean shell is a right circular cylinder with one end welded to a hemispherical steel dome and the opposite end welded to a thick base ring which in turn is bolted to a rigid testing fixture. The diameter of the cylinder was about 43 inches (1.1 meters) and had a height of 65 inches (1.65 meters) including the dome. The thickness of the cylinder and dome material was about 0.045 inch (1.15mm). The basemat was not modeled. The second configuration was a ring-stiffened containment which used the clean shell geometry with the addition of ten stiffening rings brazed to the cylinder wall. The third configuration was a penetration model which is also based on the clean shell geometry. Three penetrations are included in this model, representing two personnel locks and an equipment hatch. The containment models were pressurized incrementally with nitrogen gas. Strain and displacement data were recorded at each pressure increment. Due to the inherent dangers of pneumatic pressurization, testing was performed remotely in an isolated area. The steel containment models were instrumented with high elongation strain gauges, several displacement gauges, pressure transducers and thermocouples to acquire data during the testing of the models. A coordinate determination system, which uses theodolites and the principles of triangulation, was also used to measure large displacements.

For the geometries investigated, stiffening rings attached to the containment wall increased the pressure at which the majority of the wall yields and the ultimate strength of the vessel. Treating the stiffening rings implicitly, by increasing the thickness of the cylinder wall by a volume equal to the increase in volume represented by the stiffening rings i.e., smearing of the rings, was found to be a reasonable analytical procedure to predict ultimate capacity.

The penetrations in the penetration model did not have a thickened shell area around the penetration sleeves and no gaskets or seals were included in the penetrations. For these simplistic geometries, the presence of penetrations does not significantly decrease the capability of the containment. The tendency of ductile steels to flow plastically mitigates the effect of discontinuities such as penetrations. In the absence of severe flaws, estimating failure pressure using an equivalent plastic strain criterion does an adequate job for the simple geometries investigated with these analyses.

Finally, a 1:8-scale steel containment model was tested to determine its response to pressure levels exceeding the design basis. Extensive structural analyses of the model were performed prior to the test. A number of penetrations were present in this experimental model, including operable equipment hatches with single "O" ring seals, personnel lock representations, and a constrained pipe. The model was built to ASME code specifications with a design pressure of 40 psig. An extensive structural data base was generated during the high pressure test of the 1:8-scale steel containment model, which was conducted November 15-17, 1984. Data were recorded at twenty-one different pressure levels up to and including 190 psig, which is 4.75 times the design pressure. The model ruptured after the pressure in the model was increased to 195 psig. No significant leakage was detected up to this point, although the measured displacements around the equipment hatch indicated that leakage was imminent. The membrane strains, which denote strains in the cylinder away from the effects of penetrations, were between 2.5 percent and 3 percent at 190 psig.

Leakage Tests on Seal and Gasket Materials - Because of the different types of major penetrations that exist in LWR nuclear power plants and because of the large number of designs that exist for a given type of penetration, a comprehensive survey was conducted on 48 U.S. plants to determine these variations for all of the major penetrations. The survey which was performed by Argonne National Laboratory (ANL) includes all containment types, materials, penetration designs, and all types of seals and gaskets. Based on this survey data, penetrations which are most susceptible to leakage were identified. To better evaluate the relative leakage potential for the different penetrations, a number of figure of merit analyses were made. These figure of merit analyses are based on the structural behavior of the penetration-containment system and on the geometry and material variations of the seals and gaskets that are used in the various penetrations. A comparative figure of merit analysis was performed on different types of penetration designs to come up with a list of penetrations which are most susceptible to leakage at beyond design conditions, and hence, may need to be tested to determine their leakage behavior.

Also, since the seal and gasket materials used in the major penetrations are expected to be a major cause of leakage of the penetrations under severe accident conditions, an extensive literature survey of the behavior and the capability of the different seal materials under different environments was conducted. This survey indicated that the performance and the capability of all of the commonly used seal materials are affected by (i) temperature, particularly beyond 400 degrees F, (ii) aging due to radiation beyond a dose level of about 10^6 - 10^7 rad, (iii) the combination of radiation and temperature and (iv) the steam environment (mainly for silicone rubber). However, very little information was found on their behavior under severe accident environments of interest. Hence, a plan has been put in place to test some of the more commonly used seal and gasket materials for severe accident conditions in order to understand their leakage behavior. Utilizing the information from the survey of the plants and the survey of the seal and gasket materials, a test matrix for testing seals and gaskets has been developed. Some of the common geometries that will be tested are: O-ring, tongue and groove, double dog-ear, double gumdrop and inflatable seals. All these geometries are commonly used in large mechanical type penetrations (e.g., equipment hatch, personnel air lock). The O-ring type is also commonly used in EPAs. Some of the materials which will be tested are: silicone rubber, ethylene-propylene type rubbers (EPR/EPDM), Neoprene and Viton. The test matrix for the seals and gaskets will evaluate the effects of radiation aging, temperature, linear scaling (i.e., effects of the length of the seal), cross-section, rotation between seal mating surfaces and gaps between mating surfaces. Full-size cross sections of the seals will be used in these tests.

The experiments on seals and gaskets were initiated in FY 84, continued into FY 85, and will be completed in FY 86. Tests are conducted at both Sandia National Laboratories and at Idaho National Engineering Laboratory. At Sandia, tests have been performed on unaged silicone rubber 1/4 inch O-ring seals in both steam and air environments. Both ages and unaged EPM/EPDM 1/4 inch O-ring seals have been tested in steam and air environments. The remaining seal and gasket tests will evaluate the effects of different seal geometries, size of the seal cross section, materials, different aging scenarios, and the ability of seal materials to fill gaps. A fixture for testing inflatable seals is being designed and work has begun on a test plan.

At INEL neoprene double gumdrop and double tongue and groove seals have been tested at flange rotations of three, six, and twelve degrees. Future tests will evaluate different seal materials and geometries at the same three flange rotations.

Electrical Penetration Assembly Tests - The first of a series of Electrical Penetration Assembly EPAs with the highest potential for leakage are (i) those with organic seals and gaskets (ii) those with elastomer O-rings on header plates and (iii) those designed for low pressure capability. Regarding the availability of the EPAs, of the identified 18 suppliers, only 3 suppliers are active today from whom actual full size EPAs similar to those supplied to the existing nuclear power plants can be obtained. These are D.G. O'Brien, Westinghouse, and Conax.

The EPAs will be tested to determine their leakage under severe accident environments typical of both PWRs and BWRs. These severe accident profiles are based on calculations performed under the Severe Accident Sequence Analysis (SASA) program, also funded by the NRC. Where such information is not available, other sources including data from probabilistic risk analysis (PRA) studies of plants were used. The test matrix is:

EPA	PLANT	SEVERE ACCIDENT ENVIRONMENT
Conax	BWR - Mark I	700 F, 135 psia
Westinghouse	BWR - Mark III	400 F, 75 psia
D.G. O'Brien	PWR	360 F, 155 psia

The expected end product of the program is to obtain penetration degradation information and measured leakage data through the EPAs if leakage occurs, measured thermal gradient along with EPAs, and an evaluation of electrical degradation (insulation resistance), if any, with time. The EPAs will be aged for the equivalent of a 40-year life and exposed to radiation simulating that of severe accident prior to the severe accident steam tests.

A full scale qualified D.G. O'Brien electrical penetration assembly (EPA) including low and medium voltage and instrument and control modules was tested in a PWR containment simulated severe accident environment for 10 days at 360 F and 155 psia steam conditions. The glass to metal electrical seals and flange seals exhibited zero leakage during the test indicating that containment integrity thru the EPAs would be maintained. However, the low voltage and instrument/control electrical circuits developed low resistance to ground after 2 days exposure to severe accident conditions. Operator response actions relying on these circuits for information signals or equipment actuation would thus become questionable after the second day into the accident. The EPA electrical connections, on post test examination, were found to have been wet and the elastomeric seals were found to have degraded permitting the moisture ingress at the pin connectors.

Leakage Through Valves - The Energy Technology Engineering Center (ETEC) completed elevated temperature leak testing on two 8 inch and one 24 inch butterfly valves typical of containment purge and vent valves.

The 24-inch valve passed the design basis elevated temperature leak test without leaking during or after the test. During the severe accident portion of the elevated temperature leak test, leakage was observed with the valve pressurized at 90 psig. The pressure was applied to the shaft upstream side of the disc. The packing was also observed to be leaking. The leakage started shortly after stabilization at 350 F. The pressure was increased to 120 psig and the packing leakage increased to the point that it was necessary to lower the pressure to continue the test. The 350 F hold period with the shaft pressurized upstream was completed at 90 psig. The opposite side of the valve was then pressurized and the pressure increased to 120 psig. During a four hour hold period at 350 F, random leakage was observed; however, it was not constant. After the valve cooled down, ambient leak tests were performed. With pressure applied to the shaft upstream side of the disc, seat leakage averaged 38 standard cubic feet per hour (SCFH) at 50 psig and 320 SCFH at 125 psig. With pressure applied to the opposite face of the disc, leakage was zero at all pressures.

The pressure-temperature combinations were also applied to the two 8" valves for shaft side and nonshaft side disk orientations. One of the valves experienced leakage at most of the test points considered while the second 8" valve experienced no leakage either at design basis conditions or at severe accident conditions for all disk orientations. For the first 8" valve, leakages were significantly higher with pressures on the nonshaft side than for the shaft side orientation. Specifically, at design basis conditions (60 psig and 285°F) the leakage began at 58 cm^3/min and increased to 84 cm^3/min at 60 psig and 310°F. The leakage diminished to 32 cm^3/min when the pressure was increased to 90 psig with steady temperature of 310°F. No leakage was observed for pressures on shaft side orientations. The past elevated temperature test showed the leakage to decrease from 52 SCFH (at 50 psig) to 36 SCFH (at 125 psig) with pressure on the nonshaft side orientation. For the shaft side orientation the leakage remained nearly constant at 20 cm^3/min when the pressure was increased from 50 to 125 psig.

The capability of typical containment isolation valves to function and to prevent excessive leakage when subjected to loads resulting from containment wall displacement due to severe accident is important to the containment performance program. The loads transmitted from the piping penetrations through the piping to these valves may prevent the valves from isolating containment during an accident. Testing using typical piping, supports and valves will be completed in FY 87 to obtain information for upgrading the model for estimating containment leakage.

3.3.1.3 Discussion of Outstanding Issues

the amount of load that can be sustained by a containment structure before the rate of leakage becomes unacceptable. State-of-the-art methods cannot reliably predict whether leakage will begin around penetrations or in the membrane region of the shell. If, as is thought, leakage at penetrations is critical, the effects of aging on gasket performance will be of significance. The technical problems involve developing an ability to predict deformations for the wide variety of containment designs, relating deformations of containment structures to leak behavior, and determining the sensitivity of predictions to uncertainties about actual containment structures and the loads associated with accident scenarios.

Since 1980, attempts have been made to estimate the capacity of containments. Previously, design practice had concentrated on assuring low leakage under design basis accident conditions through a combination of elastic analyses and performance tests. The first generation of attempts to predict containment performance under severe accident conditions clearly indicated that minor modifications to design methods will not suffice to predict failure modes.

The approaches taken involved using inelastic analyses of an axi-symmetric shell model to predict deformations at pressures beyond design level. Then inferences about leakage at penetrations and shell failure were attempted based on calculations of, respectively, deflections and strains. This, although a reasonable first approximation, does not really reflect what will happen near failure if, as is generally suspected, failure will be due to leakage around major penetrations. The fundamental weakness is that the inelastic analyses neglect the existence of penetrations and, thus, their effect on local deformations.

A study of containment response, sponsored by IDCOR, concluded, based on engineering judgment, that containment failure would likely be due to slow growth in leakage. However, no calculations were available to substantiate the conclusion. Further studies, by the Containment Performance Working Group, reached the following conclusions:

- The potential for significant leakage before reaching currently reported containment threshold pressures appears to be greater for BWRs than PWRs.
- Leakage before reaching threshold pressures can also occur with PWRs, but such leakage is much more plant specific.
- It is judged that leakage before gross failure will always occur. However the demonstration of such leakage for some containments will require investigations at pressures above currently reported threshold pressures (large containment deformations).
- Failure of non-metallic seals in containment penetrations (primarily equipment hatches, drywell heads and purge valves) are the most significant potential sources of containment leakage.
- Although generic studies of containment types are useful in identifying sources of containment leakage, final conclusions may need to be plant specific.
- Current efforts rely on analysis and engineering judgment. Additional test data are needed to better quantify the leak tightness of containment penetrations when subjected to severe accident conditions.
- Based on the results to date, both analytical and experimental studies should continue to better quantify containment leakage during severe accident conditions.

3.3.1.4 Planned Activities

Most of the activities planned through FY 86 relate to failure modes of containments after severe accidents initiated by an internal event. These activities are summarized below and depicted in Figure 3.9. Planning for experiments

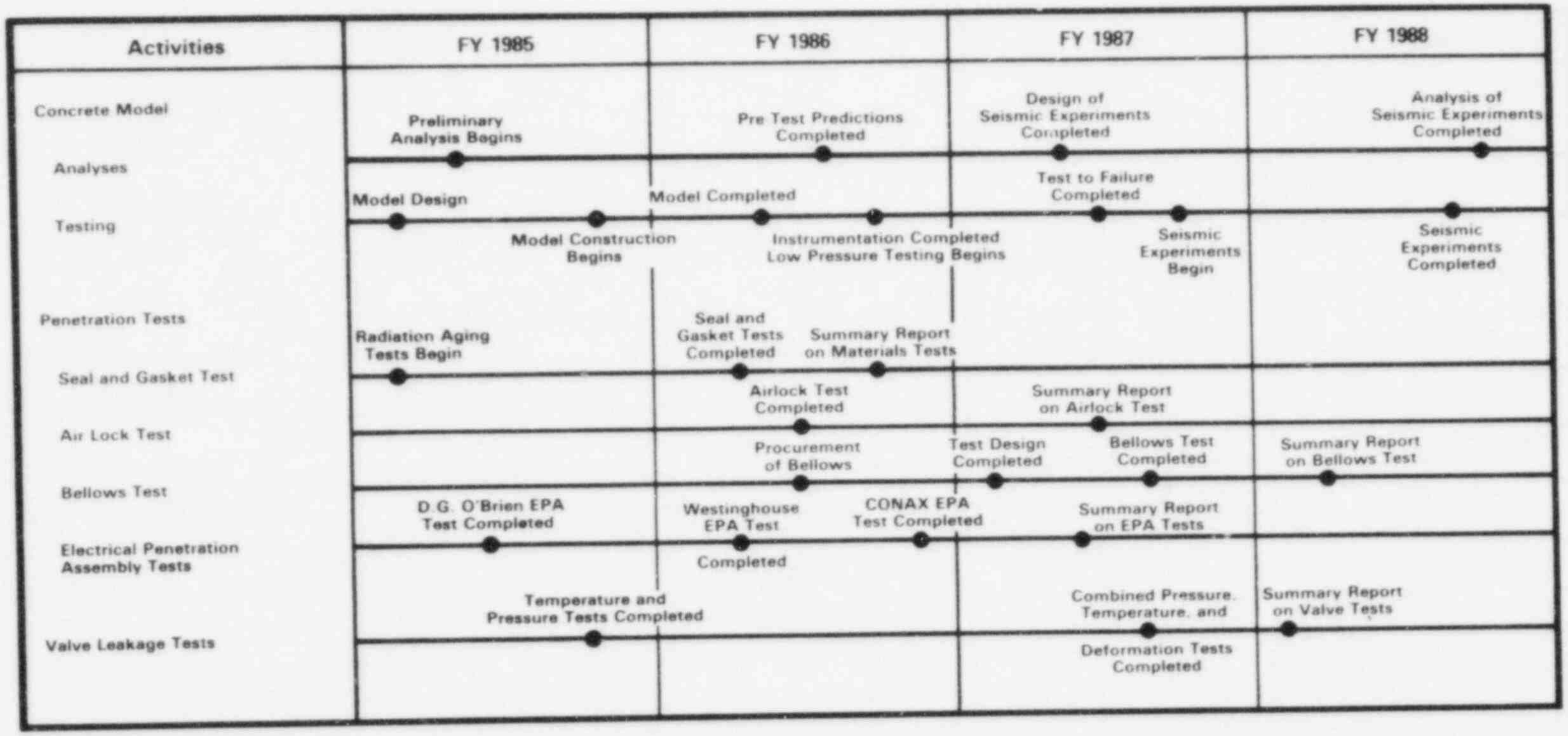


Figure 3.9 Planned activities for containment failure modes research

necessary to determine when containment capacity could be degraded in an earthquake initiated accident will begin in FY 86, and experiments will begin in FY 87.

Model Tests

During FY 1985, construction began on a 1/6 size model of a reinforced concrete containment structure to further study the structural and leakage behavior of nuclear containment buildings during severe accidents. The concrete model will be approximately 35 feet high and 24 feet in diameter. The cylindrical wall will be approximately 9 inches thick. The conceptual design calls for #3 reinforcing bars for its main reinforcing and a 1/16 inch thick steel liner. Other features of the model include operating equipment hatches, personnel lock representations, constrained and unconstrained piping penetrations, and thickened liner sections around penetrations. An important part of the project will be the support tests. These tests are designed to confirm that a high quality model can be built and also to provide a data base of the properties of the various materials used in the model. The current plans call for the model to be ready for testing around the middle of FY 86.

The proposed design pressure for the concrete model is 46 psig. This value represents the average design pressure from a survey of 17 reinforced concrete containments. The design details for the model will be representative of those found in actual containments. The seismic reinforcing will be geometrically scaled from that of actual containments.

In order to test as many configurations as possible, two equipment hatch penetrations and three hatch covers are planned for the model. On one of the equipment hatch penetrations, there will be a pressure seated hatch cover. On the other penetration, there will be a pressure unseated hatch cover on the exterior and on the penetration sleeve and a pressure seated hatch cover on the interior end of the sleeve. The pressure seated cover will be closed if leakage from the pressure unseated cover becomes excessive.

Each equipment hatch cover will have a different type of seal. Seals being considered include double tongue and groove, double dog ear, double O-ring, and gum drop gaskets. There will be two personnel lock representations in the model. The possibility of having sealing surfaces in the locks is being investigated. Two pipe penetrations will be partially constrained from radial movement. These penetrations will be scaled from typical main steam line dimensions.

Penetration and Materials Tests

Tests of both pressure-seating and pressure-unseating equipment hatches, at 1/6 scale, will be a part of the concrete model tests described above. In addition, a full scale size personnel air lock, with pressure seating seals, will be tested to failure in 1986 under pressure and temperature conditions. This test will be a key element in assessing the applicability of the seal and gasket materials test data for predictions of performance in actual penetrations. Finally, a bellows connection will be tested to failure under pressure and temperature conditions in FY 1987.

Electrical Penetration Assembly Tests

The Westinghouse EPA has been received and aged. It will be tested in a BWR Mark III environment during December 1985-January 1986. The test of the Conax EPA in a BWR Mark I environment is scheduled to begin in June 1986.

Valve Leakage Tests

Test of Containment Isolation System (CIS) valves will be performed under combined pressure and seismic conditions in FY 1986. Butterfly valves, small globe valves and gate valves will be tested. Tests with the combined effects of high pressure, temperature and the loadings due to containment displacements will be completed in FY 1987.

3.3.2 Equipment Survivability Research

3.3.2.1 Scope of Research

Equipment survivability during severe accidents should be maintained and monitored with the aid of specific instrumentation. Tests will be conducted on a typical effluent monitoring system piping design to determine critical transport parameters under severe accident conditions.

Other efforts will focus on determining the effects of containment displacement due to severe accident pressures and temperatures on the operability of typical containment isolation system valves. The results of this effort will provide the NRC with the basis for evaluating the performance of isolation valves under severe accident conditions.

Finally, an important effort is underway to assess the ability of safety related equipment in containment to survive and function under hydrogen combustion conditions. This work includes the development of thermal correlations for predicting equipment behavior and response to deflagration type hydrogen burns in ice condenser PWR containments and large dry PWR containments, and to standing flames in Mark III BWR containments.

3.3.2.2 Research Accomplishments to Date

Testing plans have been written for a typical effluent monitoring system piping design. Efforts to date on containment isolation valve performance under severe accidents consisted of designing the test apparatus for use in the tests that will begin in FY 1986. The apparatus will be used to simulate the loading resulting from containment displacement (due to internal pressure and temperature). The displacement causes loads to be transmitted to the valves through the attached piping.

In the hydrogen burn area, work on assessing the survivability of safety related equipment started with the development of the HYBER code (NUREG/CR-3853 and 3779) for calculating the thermal response of equipment to a global hydrogen burn. Hydrogen burn experiments were performed at SNL to develop models for the code (NUREG/CR-3521), and calculated temperature profiles have been compared with simulated heat fluxes for specific accident sequences (NUREG/CR-3776). The code gave reasonably good results. The tests at NTS provide some large scale data on the performance of actual safety related equipment. An analysis of the NTS premixed combustion test data has been made (NUREG/CR-4138). A similar analysis for the continuous injection combustion test data is underway. The HECTR code predictions for global temperatures and pressures gave reasonable agreement with the experiments. However, the loss of heat flux gauges and thermocouples limited the use of the data for validating equipment thermal response models in the HYBER code. Simulation of the thermal response of cables, a pressure transmitter and solenoid valve in a test at the Central Reactor Test Facility (CRTF) has been made using radiant thermal heat fluxes that duplicate those from the 13.5 volume percent hydrogen premixed burn test in NTS. Sensitivity tests have been made at CRTF with the same components exposed to a simulated heat flux three times that for a 13.5 volume percent hydrogen burn. The cable jackets burned as at NTS but the insulation was undamaged and all components performed functionally during the tests (NUREG/CR-4146 and 4324). Analysis of equipment performance in an ice condenser containment has been done for selected accident sequences (NUREG/CR-3954). The calculations predict equipment temperatures that exceed typical qualification temperatures for certain sequences where the ESF systems performance are degraded. Information from the HCOG quarter scale tests will be used in evaluating the survivability of equipment in a Mark III containment and in assessing code predictions for a standing flame.

3.3.2.3 Outstanding Issues

The outstanding issues in this area relate to survivability of safety related equipment in a Mark III containment from diffusion flames, the survivability of equipment in an ice condenser plant for certain core melt sequences and the survivability of equipment and penetrations in sub-atmospheric and smaller volume large-dry PWR's under severe accident conditions. The issue related to assessing the effects of dynamic loads on equipment from flame acceleration or local detonations for specific containment types has been held in abeyance pending the results of on-going research described in Section 3.3.1.

3.3.2.4 Planned Activities

Activity will focus on determining the performance of typical plant instrumentation under severe accidents. Testing is planned for FY 1986 for parametric testing of an effluent monitoring sample line system to determine the degree of plate-out and the time response of an effluent monitoring system under severe accident conditions.

In addition to the effort identified above, the operability of purge and vent valves (PVV) and other kinds of containment isolation valves (CIV) will be determined when subjected to severe accident conditions. The capability of these valves to open and/or close during increasing temperature and pressure will be determined by testing under such conditions. The loading associated with these tests will be applied by displacing the containment wall to correspond to the effects of severe accident pressures and temperatures. The resulting valve loads are applied through the attached piping. All tests will be completed in FY 1987.

In the area of hydrogen burns, research will focus in FY 1986 on follow up experiments to the NTS program using the new SNL Component Test Facility to evaluate the thermal response and survival of equipment. The data and reports developed in this program will be further reviewed and evaluated in FY 1986. The convective heat transfer model in HYBER and HECTR will be improved in FY 1986 using this test data. The assessment of equipment survivability in a Mark III containment will begin in FY 1986 and be completed in FY 1987. Analytical work on the response of equipment to a hydrogen burn from a severe accident for an ice condenser plant was largely completed in FY 1985. Work on equipment survival in sub-atmospheric or large dry containments will be completed in FY 1986 and a decision on the need for any change to the Hydrogen Rule will be made. A schedule of milestones for this research is given in Figure 3.10.

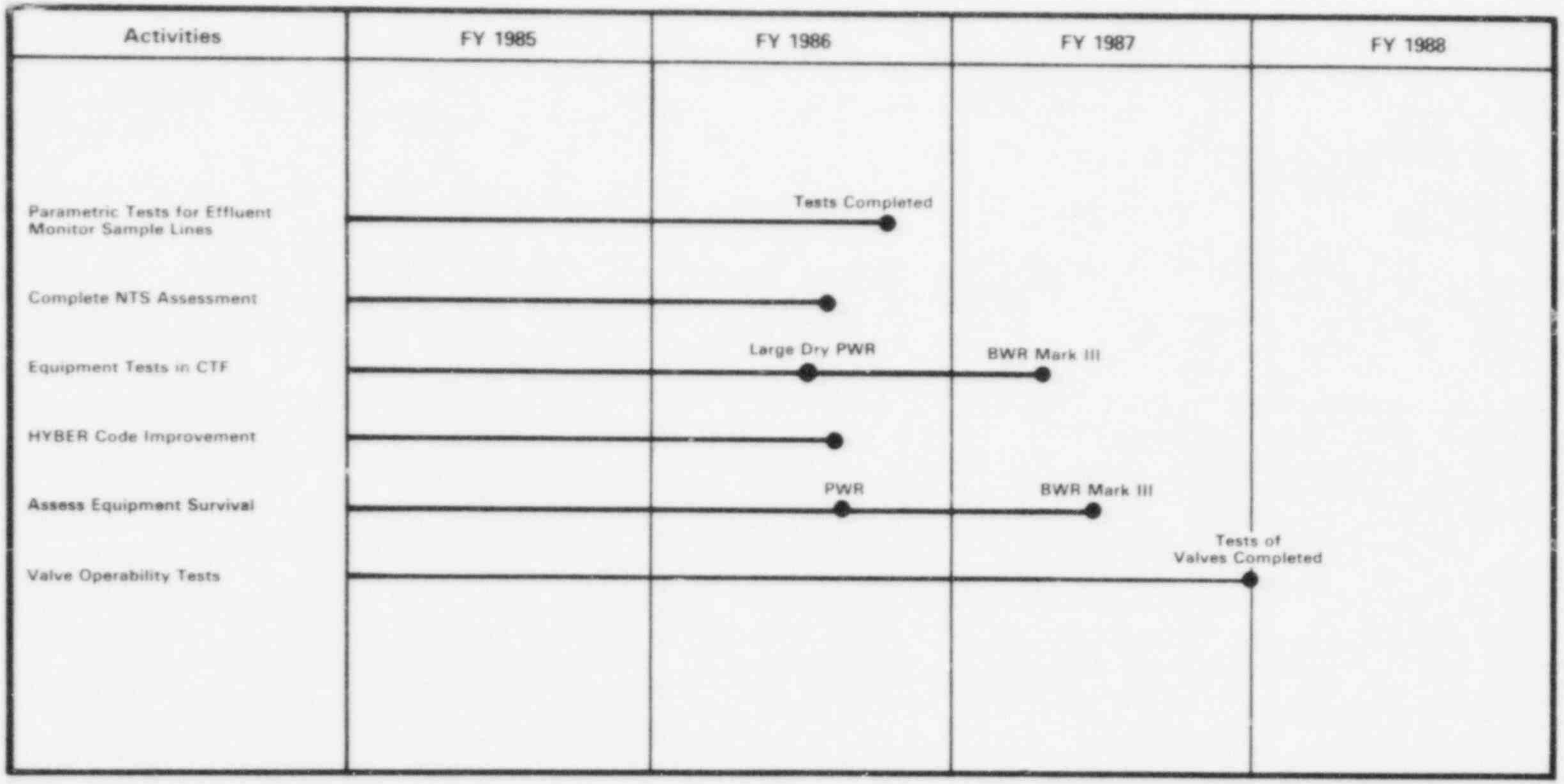


Figure 3.10 Planned activities for equipment performance and survivability research

4 SEVERE ACCIDENT COMPUTER CODES

In every program element described above, there is an effort to reduce experimental observations and theoretical considerations to mathematical models. Because these mathematical models are often complex, it is not practical to do hand calculations with them so the mathematical models are programmed into computer codes. In principle, it would be possible to take the many individual codes -- the so-called detailed mechanistic codes -- and apply them to certain accident sequences to calculate source terms and risks. In practice this is not usually attempted because of (a) absence of important feedback between codes, (b) excessive running time of the detailed codes, and (c) difficulty of managing code interfaces.

4.1 Two-Tier Code Strategy

As a consequence of the need to do calculations covering a broad range of phenomena, a two-tier code strategy has evolved for the analysis of severe accidents. One tier in this strategy consists of an integrated code that is designed to cover the full range of accident phenomena, to incorporate proper feedback, and to be fast running. There are, in fact, two such integrated codes at this time: one that is being used for current source term and risk analyses, and one that is being developed as a replacement that has been designed to correct limitations of the present code. A broad-scope integrated code is also needed to study uncertainties because of its ability to propagate variations through the full range of phenomena.

The other tier in this strategy consists of a number of detailed mechanistic codes that result from the above program elements. The detailed mechanistic codes are often developed in close connection with experimental programs. These codes of limited scope find direct application in planning and interpreting experimental results (e.g., the extensive use of SCDAP in the PBF and LOFT programs). They are used to investigate highly specialized problems for which the integrated codes are inadequate (e.g., the area of natural circulation in the upper core and plenum regions). These codes, or the scientific principles within them, are used in developing the integrated codes (MELCOR development is dependent on the mechanistic code programs), and the detailed mechanistic codes are also used for benchmarking the integrated codes (see Section 4.2.1). In short, the science of severe accident phenomena resides in the mechanistic codes.

Figure 4.1 illustrates the two-tier code strategy and identifies by name many of the computer codes used in the Severe Accident Research Program. Relationships between the detailed mechanistic codes have been shown in Figure 4.1 (namely, FASTGRASS is a part of SCDAP, VICTORIA is a part of MELPROG, and CORCON, VANESA, MAEROS, SPARC, ICEDF, and HECTR are parts of CONTAIN). Relationships between the detailed mechanistic codes and the integrated codes are not shown in Figure 4.1, but such relationships exist. In particular, TRAP-MELT, CORCON, VANESA, NAUA, SPARC, and ICEDF along with several other codes, which would not be characterized as detailed mechanistic codes (namely, MARCH, MERGE, and CORSOR), make up the BMI-2104 suite of codes. This suite of codes, with some modifications, is now called the Source Term Code Package (STCP). The fully

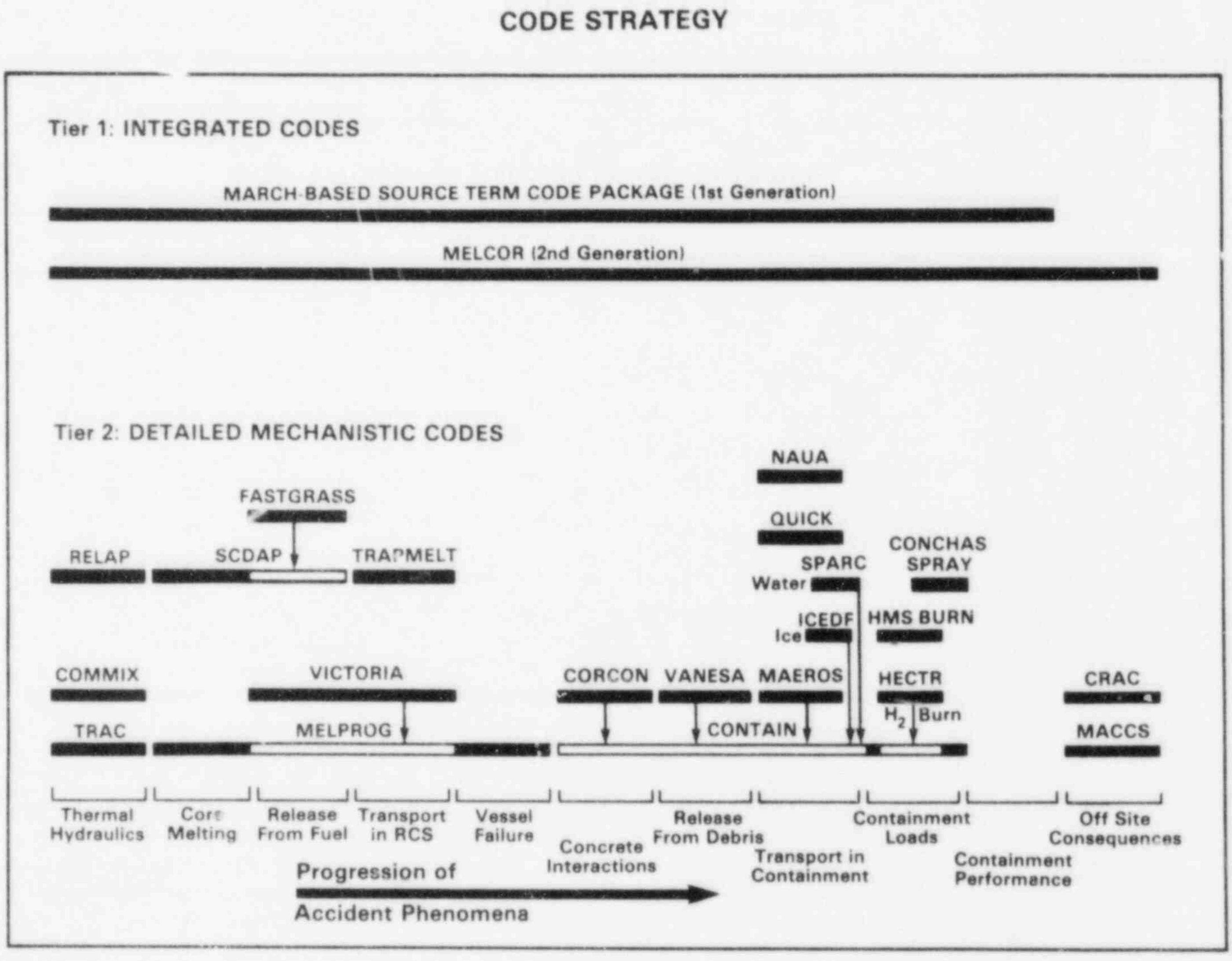


Figure 4.1 Two-tier code strategy showing approximately the accident phenomena covered by each code

integrated MELCOR code does not usually incorporate complete as-programmed mechanistic codes, but it does include a large amount of the modeling from MELPROG and CONTAIN. Furthermore, the MACCS off-site consequences model, which was developed as part of the MELCOR program, has been found to be useful as a stand-alone code and is thus shown as a detailed mechanistic code.

All of the codes used in the Severe Accident Research Program in a prominent way are described below in Section 4.3. There are a few areas of code development where the scope of development efforts overlap. In each case there are technical reasons why parallel efforts are maintained. The brief code descriptions in Section 4.3 provide some indication of these reasons. Additionally, the competition between groups and the technical critique provided by one group on the other's work can be quite valuable. There are costs associated with this duplication as well, so the areas of overlap are closely monitored to ensure overall value to the code development effort.

4.2 Verification and Validation

The following definitions of verification and validation have been adopted for this program. Verification is a quality assurance activity that involves checking a computer code to make sure that its mathematical equations have been programmed accurately and that the code runs as intended. Validation is the process of comparing computer code predictions with experimental observations or fundamental laws of physics to make sure that the mathematical models in the code are scientifically valid. Verification and validation efforts exist for all of the severe accident codes, but the extent and formality of these activities vary as described below.

4.2.1 Code Verification

The integrated code STCP is currently being used to provide many numerical results on which licensing decisions will be based, and MELCOR will likewise provide for licensing decisions in the future. As a result of discovering both coding and input errors during the review of similar BMI-2104 results, formal verification programs were initiated for both STCP and MELCOR. These verification programs are independent inasmuch as verification is being performed by Brookhaven National Laboratory whereas STCP and MELCOR were developed at different laboratories (Battelle and Sandia, respectively). In addition to checking for errors, these programs also check code portability, user friendliness, running time and traceability of modifications. Formal procedures based on industry standards and NRC licensing practices are being developed at BNL and will be described in early FY 86 reports from these programs.

The detailed mechanistic codes are more developmental in nature and tend to change or be modified frequently in the search for improved understanding of severe accident phenomena. Consequently, independent and formal verification programs do not usually exist for the detailed mechanistic codes. Instead, reliance is placed on individual laboratory practices, which often incorporate a system of double checking modeling changes that are made to an existing version of a code.

4.2.2 Code Validation

All codes in the Severe Accident Research Program have been validated to some extent inasmuch as the codes are based on experimental data and/or fundamental physical principles. In some cases, however, only subparts of a code have been validated leaving the code untested as a whole. In other cases a code may have been compared with data during its development, but it may be lacking a comparison with independent data. Furthermore, the validation that has been performed has not always been presented in readily available forms. Consequently, the American Physical Society's study group on radionuclide release from severe accidents strongly recommended additional validation of the NRC's severe accident computer codes.

To provide for improved code validation, a coordinated validation plan is being prepared for the Severe Accident Research Program. (See Figure 4.2). The first step in preparing the validation plan is to identify past and near-term future experimental results that are suitable for code validation. NRC staff members and contractors, who are responsible for experimental programs, are identifying such data. Contacts are being made with the Electric Power Research Institute, the Department of Energy, and with NRC's foreign partners in the Severe Accident Research Program to insure the inclusion of all relevant sources of information. Data from the TMI-2 examinations sponsored by DOE are also included. Table 4.1 is a preliminary list of major experimental programs that will provide data for code validation.

Step 2 of the validation plan involves scheduling specific calculations with each of the detailed mechanistic codes for comparisons with relevant data. Preliminary schedules have been established for SCDAP, TRAP-MELT, MELPROG, and the CONTAIN series of codes. Efforts are underway to complete a validation matrix for the detailed mechanistic codes.

Step 3 is referred to as benchmarking for the integrated codes, STCP and MELCOR. For the most part, Step 3 will consist of comparing STCP and MELCOR calculations with detailed mechanistic code calculations. The cases for comparison, when possible, will be a subset of the same cases selected for data comparisons with the detailed mechanistic codes. This selection process will thus produce code-to-code comparisons as well as code-to-data comparisons for the integrated codes. Step 3 will also include comparisons of STCP and MELCOR calculations with IDCOR and EPRI codes, which may be used in licensing, and with codes developed by the SARP foreign partners.

A note on code validation is appropriate here. No experimental facility has been constructed or proposed that could provide data over the whole range of phenomena covered by the integrated severe accident codes. The TMI-2 accident, for example, did not proceed to vessel failure and thus provides no data relevant to a large number of the severe accident phenomena of interest. Even for the phenomena that occurred, data will be difficult to obtain, uncertain, and incomplete. Therefore, under the best circumstances, integrated codes could not be fully validated, but could only be validated in a piecemeal manner to the extent that experimental programs cover broad (but not complete) ranges of phenomena. In this regard it should be mentioned that large tests that provide the best validation possibilities are expensive, and that for reasons of economy the PBF and LOFT programs have been terminated.

4.3 Descriptions of Current Codes

The following paragraphs will briefly describe what each of the severe accident codes calculates and indicate why it is maintained as an active code in the research program. The codes are discussed alphabetically. Numerous references

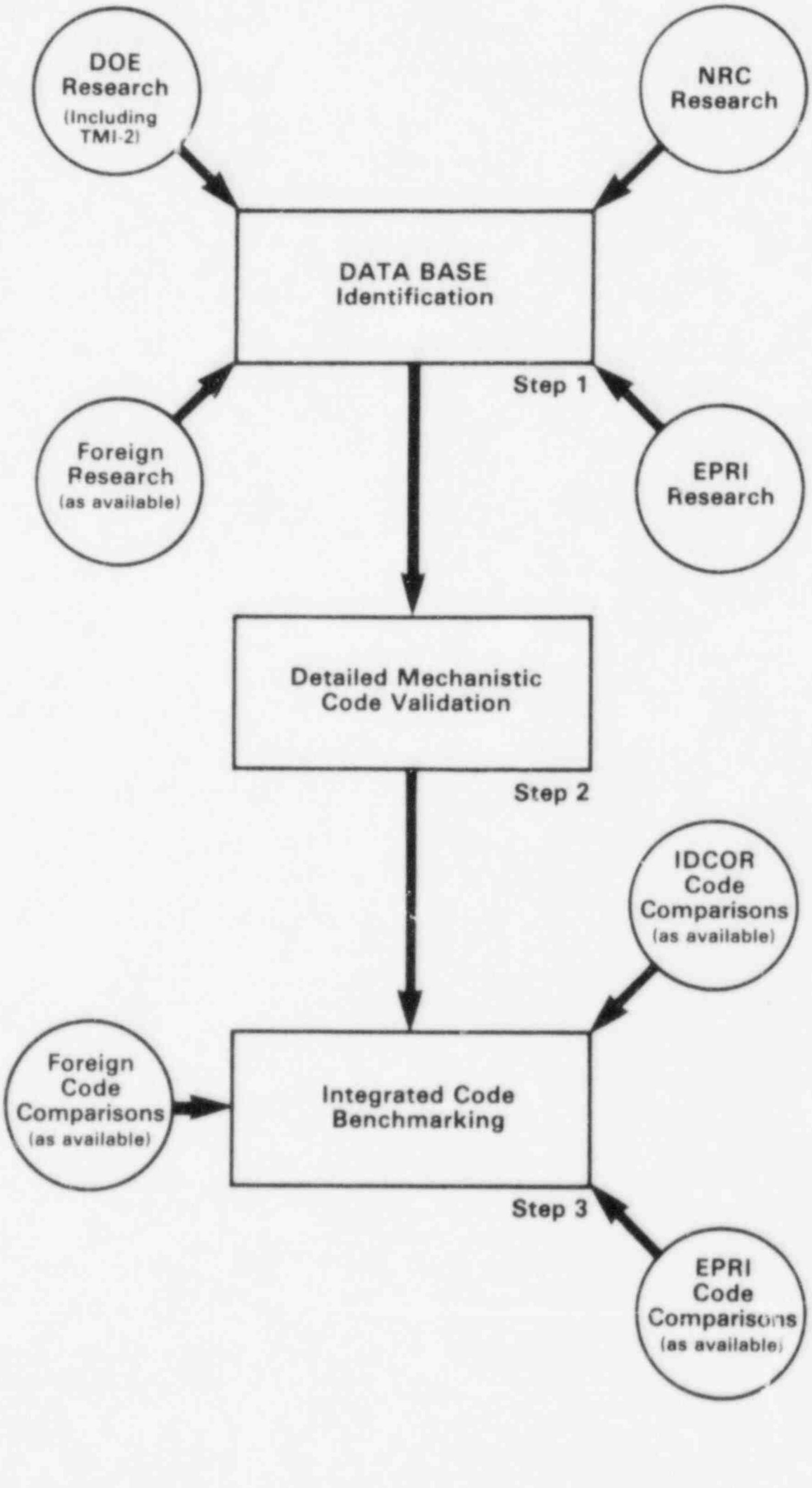


Figure 4.2 Three steps of the code validation plan

Table 4.1 Current programs producing data to be used for code validation

Test Identification	Lab.	Phenomena
PBF ST, 1-1, 1-3, 1-4	INEL	Core Melting, Release from Fuel
LOFT FP-2	INEL	Thermal Hydraulics, Core Melting, Release from Fuel, Transport in RCS.
ACRR DF-1 thru DF-4	SNL	Core Melting
ACRR DCC-1 thru DCC-3	SNL	Core Melting
NRU FLHT-1 thru FLHT-5	AECL	Core Melting
A103 thru A104	ORNL	Transport in RCS
LACE LA1, LA6	HEDL	Transport in Containment
MARVIKEN 1, 2A, 2B, 4, 7	Sweden	Transport in Containment
NSPP	ORNL	Aerosol Transport
SWISS, SURC, TURC	SNL	Concrete Interactions, Release from Debris
BETA	Germany	Concrete Interactions
Hydrogen Tests	NTS	Containment Loads
HDR Tests	Germany	Containment Loads
VGES Tests	SNL	Containment Loads

to these codes are made in other parts of this report in connection with the discussion of SARP program elements and individual experimental programs. Since the reader may wish to refer back to the comments that accompany these references, an index is included as Table 4.2.

BWR-LTAS (Developer: ORNL, Sponsor: CSRB/DAE)

The BWR-LTAS code (Ref. 4.1) calculates the thermal-hydraulic behavior of the reactor vessel, the reactor coolant system, the primary containment, and other reactor systems in a BWR. The code was developed in the Severe Accident Sequence Analysis (SASA) program for the detailed study of specific accident sequences at Browns Ferry Unit One. The primary use of the code has been to estimate the effects of operator actions on the timing and course of events during the part of the sequence leading up to, but not including, severe fuel damage. BWR-LTAS is not a user-friendly tool, but for the experienced users in the SASA program it offers savings in time and money over conventional codes.

COMMIX (Developer: ANL, Sponsor: CSRB/DAE)

The COMMIX code (Ref. 4.2) was developed to study heat removal from liquid metal fast breeder reactors. It is designed to analyze transient or steady-state fluid flow with heat transfer in three dimensions. Special features such as the porous-media treatment allow the user to model the whole interior of a reactor vessel in one calculation. COMMIX is a mature code and is widely used by the LMFBR community. The code is being applied to two special LWR problems: (1) three-dimensional effects in the upper plenum of Zion, where its special features allow a good representation of structures, and (2) mixing of water and steam in the main steam line break in Catawba. COMMIX was selected as the tool for these analyses as it can handle a very fine calculational mesh at reasonable cost.

CONCHAS-SPRAY (Developer: LANL, Sponsor CSRB/DAE)

CONCHAS-SPRAY (Ref. 4.3) provides a finite-difference approximation to the Navier-Stokes equations for two-dimensional fluid flow. The code is used to simulate two different processes that affect hydrogen control in nuclear reactors. The first is the acceleration of flames in confined geometries in the transition from burning to detonation. Detonation calculations are made with a mature code called CSQ (Ref. 4.4) that is being used, but not further developed, in the Severe Accident Research Program. The second process simulated by CONCHAS-SPRAY is the generation of air flows due to activation of a water spray system in a reactor containment. CONCHAS-SPRAY has also been used to study the effect of water sprays on hydrogen ignitors.

CONTAIN (Developer: SNL, Sponsor: CSRB/DAE)

The CONTAIN computer code (Ref. 4.5) is an integrated analysis tool for the physical, chemical, and radiological conditions inside a containment building

Table 4.2 Index to comments on the severe accident computer codes

Chapter	1	2	3	4	5	App
BWR-LTAS	-	-	-	7,18	-	-
COBRA	-	-	-	-	7	3
COMMIX	-	-	-	7,18	7	3
CONCHAS-SPRAY	-	14	29	7,18	-	-
CONTAIN	-	13	27,28,30,32	3,9,18	9,10,11	-
CORCON	5,6	13	7,27,28,30,32, 34,36,37	9,18	9,11	5
CORRAL	5	-	-	-	10	-
CORSOR	5	12	7,22,23	9,18	-	2
CRAC	-	8	5	9,18	10	-
EVENTRE	-	-	4	-	-	-
FASTGRASS	-	10,12,13	17,22,23,	10,18	-	-
HECTR	-	14	28,34	10,18	10,11	-
HMS-BURN	-	14	28,34	10,18	-	-
HYBER	-	-	47,48	11,18	-	-
ICEDF	5	-	36,37,39	11,18	-	-
MAAP(IDCOR code)	-	-	5,18	-	-	6
MACCS	-	6,8	-	11,18	-	-
MAEROS	-	-	5	11,14, 18	11	-
MARCH	5	-	5,7,9,18,27,28	12,19	10	4
MARCON	-	-	-	12,19	-	-
MELCOR	6	4,6,8,10 12,13,18	4,5,10,19,20, 27,28,32	13,19	10,11	-
MELPROG	-	10,12,13	16,17,18,20, 21,22	13,14, 19	7,11	2,4
MELRPI	-	-	-	12	-	-
MERGE	5	-	7,23	-	-	6
NAUA	5	-	-	13,19	-	-
QUICK	-	-	-	14,19	-	-
RELAP	-	9,10,12, 13	17,20	14,19	7,11	-
SCDAP	5	9,10,12, 13	18,20,21,22 16,17	1,4 14,19	8	6
SPARC	5	-	36,37,39	1,15,19	-	-
STCP	-	4,6,8,10 12,13,18	7,9,18	1,3,4, 15,19	-	
SUPRA(EPRI code)	-	-	37	-	-	-
TRAC	-	10,12	17,20	15,19	11	3
TRAP-MELT	5	9,10,12 13	7,17,20,22,23,24,26	15,19	5	3,6
TRENDS	-	-	36,39	16,19	-	-
VANESA	5	13	30,32,36	16,19	9,10,11	5
VICTORIA	-	10,12,13	22,24	16,19	-	2

following the release of radioactive material from the reactor coolant system. Interactions among thermal-hydraulic phenomena, aerosol behavior, and fission-product behavior are taken into account. Both light-water reactors and liquid-metal reactors can be modeled with CONTAIN. The code includes atmospheric models for steam and air thermodynamics, intercell flows, condensation and evaporation on structures and aerosols, aerosol behavior, hydrogen burning, sodium and atmosphere chemistry, sodium spray fires, and sodium pool fires. It also includes models for reactor cavity phenomena such as core-concrete interactions, coolant-pool boiling, and sodium-concrete interactions. Heat conduction in structures, fission product decay and transport, radioactive heating, and the thermal-hydraulic and fission product decontamination aspects of engineered safety features are also modeled.

CONTAIN uses four self-standing codes as modules (subroutines). Three of the codes (CORCON, VANESA, and HECTR) are still undergoing development. The fourth code (MAEROS) is a fully developed aerosol transport code.

CORCON (Developer: SNL, Sponsor: CSRB/DAE)

CORCON (Ref. 4.6) describes the interaction between high-temperature (usually liquified or molten) core debris and concrete in the cavity beneath the reactor vessel. Given initial debris composition and temperature, CORCON calculates the rate of erosion of the concrete, the amounts of liberated decomposition gases, temperatures within the debris, and heat transfer to surrounding structures. CORCON gas generation rates contribute directly to containment pressure loads, and local temperatures and gas sparging rates are very important input for the aerosol-generation code, VANESA. Mod 1 of CORCON was used in the important BMI-2104 calculations, Mod 2 (current version) has been programmed into the MARCH portion of the Source Term Code Package (STCP), CORCON is a major part of the integrated containment behavior code CONTAIN, and later versions of CORCON or its derivatives will be used in the MELCOR risk code.

CORSOR (Developer: BCL, Sponsor: FSRB/DAE)

The CORSOR code (Ref. 4.7) calculates the in-vessel release rates from the core of fission products (volatile and refractory species), structural materials (zircaloy and stainless steel), fuel (UO_2), and control rod materials (silver, indium, and cadmium). CORSOR is based on a 1981 empirical model that depends on temperature and empirical constants for each material, and it is used to calculate in-vessel releases in the Source Term Code Package. CORSOR is a small, simple code that does not have the capability to handle fuel burnup and structural effects or surface chemistry and kinetic effects. No developmental work is being done on CORSOR, but it containues to be used because the FASTGRASS code does not treat all of the materials of interest (only the volatile fission products) and VICTORIA is not yet fully developed. VICTORIA, or some simplified version of it, will provide in-vessel release calculations for MELCOR and will supersede the CORSOR code.

CRAC (Developer: SNL, Sponsor: RRB/DRAO)

Given source terms for individual accident sequences, CRAC (Ref. 4.8) calculates off-site radiation doses, health effects, and property damage. Using site-specific meteorological data, CRAC can calculate consequences for specific weather sequences or it can use one of several sampling methods to average the consequences over the yearly variation of weather conditions.

CRAC bases its calculations on a single, uniform release and a highly simplified model to account for extended duration. It has an evacuation model based on average rather than actual site characteristics, and it uses 1970 census data. CRAC was not written in standard ANSI Fortran and it is not amenable to sensitivity studies. A much improved consequence code called MACCS has recently been completed, and the CRAC code will be retired in 1987.

FASTGRASS (Developer: ANL, Sponsor: FSRB/DAE)

The FASTGRASS code (Ref. 4.9) calculates the in-vessel release of fission gases from fuel pellets. To the extent that iodine, cesium, and tellurium are gaseous at fuel temperatures during an accident, FASTGRASS calculates the release of iodine, cesium, and tellurium as well as krypton and xenon. FASTGRASS calculates fission gas transport to grain surfaces, bubble formation, bubble growth, bubble transport, venting to interlinked porosity, and the effects of fuel liquifaction and fragmentation.

FASTGRASS has done a good job of explaining fuel-burnup and structural effects in the experimental results from the Power Burst Facility; other codes cannot do this. However, FASTGRASS does not address low volatility materials and other core-debris materials that are important in core-melt severe accident sequences. Therefore, FASTGRASS is maintained as a detailed mechanistic code that is particularly suitable for arrested accident sequences. The less sophisticated CORSOR code is used in the integrated code STCP and the VICTORIA code, or some simplification of it, will used in MELCOR.

HECTR (Developer: SNL, Sponsor: CSRB/DAE)

The HECTR code (Ref. 4.10) is designed to calculate the transport and combustion of hydrogen and the related transient response of the containment. The code can calculate local concentrations for steam, nitrogen, oxygen, hydrogen, carbon monoxide, and carbon dioxide, and HECTR can provide information about the environment of the containment walls and equipment. HECTR was developed with emphasis on hydrogen combustion, but can also solve a limited set of other types of containment problems. HECTR is a relatively fast-running, lumped-volume code that is being benchmarked against the detailed HMS-BURN code. HECTR models have been incorporated into both the CONTAIN and MELCOR codes.

HMS BURN (Developer: LANL, Sponsor: CSRB/DAE)

HMS BURN (Ref. 4.11) calculates local hydrogen and oxygen concentrations within various compartments of the containment, and it calculates any resulting hydrogen burning. HMS BURN thus provides containment pressure loads due to hydrogen combustion, and it also provides local temperatures, heat-flux loads, and pressure conditions for structures and equipment within the containment. It is a three-dimensional, time-dependent, finite-difference code that requires large computational times relative to a lumped parameter code like HECTR. HMS BURN has a theoretical basis, it has been compared favorably with experimental data, and it is being used to benchmark the faster-running HECTR code.

HYBER (Developer: SNL, Sponsor: CSRB/DAE)

HYBER (Ref. 4.12) calculates combustion of hydrogen and carbon monoxide to provide gas temperatures and pressures in the atmosphere and surface temperatures of a single-volume compartment. It is a small, user-friendly code that has been designed to run on an IBM Personal Computer. HYBER is used principally to calculate the effects of burning on equipment located in a reactor containment. Equipment surface temperatures are obtained by assuming that the equipment surface is part of the control volume surface. HYBER can account for effects of water sprays, ice condensers, and leaks in the control volume.

ICEDF (Developer: PNL, Sponsor: CEB/DET)

Some PWRs are equipped with ice condenser containment systems. As steam bearing aerosols and fission products pass through the ice compartments, some aerosols are retained. ICEDF (Ref. 4.13) describes the retention of aerosol particles on ice compartment surfaces by accounting for a number of well known aerosol physics mechanisms.

ICEDF is used in the Source Term Code Package (STCP) and its predecessor, the BMI-2104 suite of codes. Models similar to the those in ICEDF have also been incorporated in MELCOR and CONTAIN.

MACCS (Developer: SNL, Sponsor: RRB/DRAO)

Given source terms for individual accident sequences, MACCS (Ref. 4.14) calculates off-site radiation doses, health effects, and property damage. Using site-specific meteorological data, MACCS can calculate consequences for specific weather sequences or it can use one of several sampling methods to average the consequences over the yearly variation of weather conditions.

MACCS bases its calculations on multiple releases, each of finite duration. This allows for more realistic treatment of extended releases expected for some accident sequences. MACCS incorporates 1980 census figures and has available an improved health-effects model and file. MACCS, which was developed as part of the MELCOR integrated code package, is portable, written in standard ANSI Fortran, designed to accommodate sensitivity calculations, and has many other improvements over the CRAC code it will replace. MACCS is currently being used in the NRC's six-plant risk rebaselining study. As part of its future development, MACCS will handle such effects as wind shifts during the release period and also will consider the actual road networks for plants being analyzed.

MAEROS (Developer: SNL, Sponsor: CSRB/DAE)

MAEROS (Ref. 4.5) calculates the agglomeration and deposition of aerosol particles within the containment environment. MAEROS has two features that make it more advanced than other aerosol codes. First, MAEROS keeps track of particle size distributions and particle composition distributions, whereas some aerosol codes like NAUA assign the average composition to all size particles. Second, MAEROS treats agglomeration by combining particle sizes in a manner that is mathematically correct, albiet more time consuming than the simplified approach used in NAUA and QUICK. MAEROS is used in the integrated containment behavior code CONTAIN, and the principal features of MAEROS are used in the MELCOR risk code.

MARCH (Developer: BCL, Sponsor: RRB/DRAO)

This large code (Ref. 4.15) calculates temperatures, pressures, and fluid flow rates at many locations in the reactor coolant system and containment for severe accident conditions. While MARCH covers a very broad range of phenomena, it does not in general cover original fission product inventories or fission product transport and retention. Therefore, MARCH calculations are supplemented with other code calculations in order to provide source term and risk analyses. MARCH plus ORIGEN, CORSOR, MERGE, TRAP-MELT, CORCON, VANESA, NAUA, SPARC, and ICEDF form the so-called BMI-2104 suite of codes. The BMI-2104 suite of codes is considered to be the original "integrated code" in the NRC's two-tier approach, although the constituent codes in the suite were only loosely coupled. A more recent, better integrated version of this code suite is now called the Source Term Code Package (STCP).

The MARCH code alone has been used in many NRC probabilistic risk assessments, and it has also been used outside the NRC for other risk assessments. The MARCH-based BMI-2104 suite of codes, as the name implies, was used for the source term reassessment analyses reported in BMI-2104. The related Source Term Code Package (STCP) is currently being used in NRC's six-plant risk rebaselining study (NUREG-1150) that will form the basis for NRR's implementation of the Severe Accident Policy Statement (NUREG-1070) and related regulatory changes. The MARCH-based risk and source term codes will eventually be replaced by MELCOR, which is now at an advanced stage of development.

MARCON (Developer: ORNL, Sponsor: CSRB/DAE)

MARCON (Ref. 4.16) is a version of MARCH that incorporates the CORCON code for core-concrete interactions rather than utilizing the original MARCH subroutine INTER for this analysis. MARCON also includes a number of BWR model innovations, some of which were developed by Rensselaer Polytechnic Institute in a version of MARCH called MELRPI.

MARCON is used in the Severe Accident Sequence Analysis (SASA) program to investigate prominent accident sequences and changes in sequences that might result when modeling assumptions are changed. This is accomplished by using MARCON as a MARCH-based vehicle for making numerous modeling changes. The use of CORCON in MARCH, as first accomplished in MARCON, is now a standard feature of current versions of MARCH as used in the Source Term Code Package (STCP). BWR options similar to those in MARCON have also been incorporated in recent versions of MARCH and STCP.

MELCOR (Developer: SNL, Sponsor: RRB/DRAO)

MELCOR (Ref. 4.17) is under development as a replacement for the MARCH-based source term and risk assessment codes. MELCOR is very comprehensive; it calculates fission product inventories, thermal hydraulics in the vessel and containment, fission product transport and retention, containment behavior, and off-site consequences. It was designed from the beginning to be modular and fully integrated. Its modular structure facilities the incorporation of new or modified phenomeno-logical models to accommodate new research results. MELCOR's structure was also designed to facilitate sensitivity and uncertainty studies, and special Monte Carlo techniques have been developed for this purpose.

Preliminary runs with the full MELCOR code were first made in January 1985, and an interim version of MELCOR was installed at Brookhaven for quality assurance purposes in August 1985. The first released version of MELCOR (BWR only) is scheduled for December 31, 1985, and other releases (covering PWRs) are scheduled in 1986. The off-site consequences model MACCS was completed ahead of the main schedule and is being used as a stand-alone code in the current risk rebaselining program. MELCOR will readily accept new research findings in the areas now being pursued to resolve technical issues.

MELPROG (Developer: SNL, Sponsor: FSRB/DAE)

MELPROG (Ref. 4.18) was designed to analyze core melt progression leading to vessel meltthrough for severe accident sequences. It models vessel thermal hydraulics, liquefaction of core materials, relocation of core materials, stressing and ablation of structural features, and the release, transport, and deposition of fission products. The submodel for calculating release, transport, and deposition of fission products is called VICTORIA and is described elsewhere in this section. MELPROG has been coupled with the TRAC thermal-hydraulic code to provide analysis capability within the vessel up to the point of (and including) vessel failure. Although MELPROG provides accident analysis in the RCS from the time of accident initiation, it is not intended to model the heat-up phase in as detailed manner as SCDAP. Rather, the major emphasis in the overall MELPROG effort is on the treatment of core degradation and loss of geometry, debris formation, core melting, attack on supporting structures, slumping, melt-water interactions, and vessel failure.

MERGE (Developer: BCL, Sponsor: FSRB/DAE)

MERGE (Ref. 4.19) is a small utility code used only in the MARCH-based BMI-2104 suite of codes and the related Source Term Code Package (STCP). MERGE subdivides the control volumes in MARCH into as many as ten subvolumes in a manner suitable for TRAP-MELT aerosol transport analysis. MERGE has recently been subsumed into the TRAP-MELT code for use in STCP and MERGE will soon lose its separate identity.

NAUA (Developer: KfK, Sponsor: German government)

NAUA (Ref. 4.20) calculates the agglomeration and deposition of aerosol particles within the containment environment. NAUA does not treat the composition distribution of different size particles and it uses a simplified, but rapid, method of calculating particle agglomeration. NAUA was developed by the KfK laboratory in West Germany and was obtained by the NRC under an exchange agreement. NAUA was the best aerosol code available at the time of the BMI-2104 source term analyses and, with some modifications, remains in the related Source Term Code Package (STCP). NAUA is being replaced by the MAEROS code in NRC's newer analytical packages (CONTAIN and MELCOR).

QUICK (Developer: BCL, Sponsor: FSRB/DAE)

QUICK (Ref. 4.21) calculates the agglomeration and deposition of aerosol particles within the containment environment. QUICK keeps track of particle size distributions and particle composition distributions like MAEROS does, but QUICK uses a simplified method of treating particle agglomeration rather than the elegant mathematical procedure used by MAEROS. Since MAEROS has been selected for use in NRC's newer analytical procedures (CONTAIN and MELCOR), development of the QUICK code has been discontinued. However, QUICK does have a running speed advantage over the MAEROS code, and QUICK is maintained in operational form for occasional use.

RELAP (Developer: INEL, Sponsor: RSRB/DAE)

RELAP (Ref. 4.22) is a generic code that can calculate a wide variety of thermal-hydraulic transients in nuclear and non-nuclear systems involving steam, water, and noncondensible gas mixtures. The code features one-dimensional modeling with cross-flow correlations that can simulate three-dimensional behavior. RELAP was initially designed to analyze design-basis loss-of-coolant accidents and anticipated transients without scram. More recently, RELAP has been coupled with SCDAP to provide for the in-vessel portion of severe accident analyses with emphasis on arrested accident sequences. Since the TRAC code has been identified as the NRC thermal-hydraulic code for further development, the RELAP code effort has been limited to a maintenance level. There are many users of the RELAP code, and its uses include applications in the severe accident research program. Therefore, the RELAP maintenance program is being augmented by an effort to make the code more user friendly and portable.

SCDAP (Developer: INEL, Sponsor: FSRB/DAE)

SCDAP (Ref. 4.23) is a severe core damage analysis package that calculates the thermal, mechanical, and chemical behavior in a reactor vessel. It is a large, mechanistic code that includes detailed fuel rod behavior models and debris behavior models. Behavioral models cover oxidation, cladding deformation, fission gas release, fuel liquefaction, fuel material candling, and damage propagation. SCDAP has been coupled with the RELAP thermal-hydraulic code to provide analysis capability within the vessel up to the point of (but not including) vessel failure.

The development of SCDAP followed earlier fuel rod analysis development efforts at INEL related to loss-of-coolant-accident analysis, and SCDAP contains the most detailed and complete fuel bundle analysis capability available to the NRC. It has been used successfully in the planning and conducting of many tests in the PBF, LOFT, and NRU test reactors, and it is being used extensively in the analysis of the TMI-2 debris samples. While SCDAP overlaps MELPROG in some of its capabilities, SCDAP is more mature than MELPROG and was intended for analysis of arrested sequences that do not result in vessel breach. MELPROG, on the other hand, is intended to take advantage of the SCDAP technology and extend the analysis to describe vessel breach and establish input parameters for core-concrete interactions.

SPARC (Developer: PNL, Sponsor: CEB/DET)

In many accident sequences, aerosols and fission products must pass through water pools before entering the containment or auxiliary building atmosphere. Such pools will scrub a portion of the aerosols and fission products from the gases bubbled through the pool. SPARC (Ref. 4.24) describes the deposition of aerosol particles on bubble walls by accounting for a number of well known aerosol physics mechanisms.

SPARC is used the Source Term Code Package (STCP) and its predecessor, the BMI-2104 suite of codes. Models similar to those in SPARC have also been incorporated in MELCOR and CONTAIN.

STCP (Developer: BCL, Sponsor: FSRB/DAE)

The Source Term Code Package (Ref. 4.25) is an improved version of the BMI-2104 suite of codes that calculates the whole spectrum of phenomena leading to radioactive releases to the environment. STCP utilizes ten separate codes (ORIGEN, MARCH, CORSOR, MERGE, TRAP-MELT, CORCON, VANESA, NAUA, SPARC, and ICEDF), which have been integrated or coupled to the extent practical. These codes have been extensively reviewed as part of NRC's source term reassessment effort (NUREG-0956).

There are a number of limitations and uncertainties in STCP as described in NUREG-0956. Nevertheless, STCP is the best analytical tool available today for calculating source terms and plant risk. STCP is currently being used in NRC's six-plant risk rebaselining study (NUREG-1150) that will form the basis for NRR's implementation of the Severe Accident Policy Statement (NUREG-1070) and related regulatory changes. An active verification and validation program is being pursued for STCP, and essential code modifications will be made to maintain the code's efficacy. However, active code development to achieve major improvements is not being performed for STCP. Instead, STCP will eventually be replaced by MELCOR, which is now at an advanced stage of development.

TRAC (Developer: LANL, Sponsor: RSRB/DAE)

TRAC, a transient reactor analysis code, calculates the thermal-hydraulic behavior of light-water reactor coolant systems. There is a PWR version (Ref. 4.26) and a BWR version (Ref. 4.27) of TRAC. The code features either one-dimensional or three-dimensional modeling of the pressure vessel and its associated internals, and it incorporates two-fluid models with a noncondensible gas field (for water, steam, and noncondensibles such as air, nitrogen, and hydrogen). TRAC is designed to analyze design-basis loss-of-coolant accidents as well as anticipated transients both with and without scram. More recently, TRAC has been coupled with MELPROG to provide for the in-vessel portion of severe accident analyses including vessel breach. TRAC has been identified as the NRC thermal-hydraulic code for which further improvement will be pursued. Its three-dimensional vessel capability makes it the preferred code for many applications and therefore development of the TRAC series of codes will continue.

TRAP-MELT (Developer: BCL, Sponsor: FSRB/DAE)

TRAP-MELT (Ref. 4.28) calculates the transport and retention of fission products in the reactor coolant system. Vapor species like CsI and elemental iodine are allowed to condense on (and evaporate from) walls and aerosol particles as the temperature dictates. Reactive species like CsOH and Te are chemi-sorbed on surfaces at appropriate rates. And aerosol particles are calculated to agglomerate and be deposited according to a number of well known aerosol physics mechanisms. TRAP-MELT was used in the BMI-2104 source term analyses, and it is part of the Source Term Code Package currently being used in the NUREG-1150 risk rebaselining analyses.

TRENDS (Developer: ORNL, Sponsor: CSRB/DAE)

TRENDS (Ref. 4.29) calculates fission product transport and retention in the reactor coolant system and containment for BWRs. It is used in connection with MARCON in the Severe Accident Sequence Analysis (SASA) program to calculate source terms. TRENDS incorporates many features that are similar to the Source Term Code Package; TRENDS incorporates a CORSOR-like in-vessel release model, it utilizes VANESA for ex-vessel releases, and TRENDS uses QUICK for aerosol behavior in containment. In addition, TRENDS treats a variety of chemical effects (particularly for volatile forms of iodine) in the RCS and containment. TRENDS uses a free-energy-minimization technique for species distribution; it calculates the deposition of vapor species on surfaces and aerosols; it calculates the dissolution of vapor species in water; and TRENDS accounts for radiolysis effects. These effects are being studied to determine their impact on source term analyses and to evaluate the need to include such models in the Source Term Code Package (STCP).

VANESA (Developer: SNL, Sponsor: CSRB/DAE)

VANESA (Ref. 4.30) calculates the rate of aerosol release from the ex-vessel core concrete interaction. The aerosols contain fission products, concrete decomposition products, and other core-debris materials. Using temperatures and gas evolution rates provided by the CORCON code, VANESA calculates simultaneously about 200 chemical reactions at the surface of bubbles in the melt; these chemical reactions lead to the vaporization of different chemical forms of approximately 25 elements (or groups of elements). As the bubbles burst through the melt surface, the entrained vapor species condense to form the aerosols.

VANESA is a very fast running code, and there are no other codes in NRC's severe accident program that perform this function. VANESA is used in the STCP and MELCOR integrated codes and in . VANESA with CORCON are used as a stand-alone detailed mechanistic pair of codes.

VICTORIA (Developer: SNL, Sponsor: FSRB/DAE)

VICTORIA (Ref. 4.31) is intended to be a submodel in MELPROG, but VICTORIA has been developed as a stand-alone code and is thus described here as a detailed mechanistic code. VICTORIA describes fission product transport out of fuel material, species chemistry, transport of gaseous species, aerosol evolution and transport, and species deposition on structures.

VICTORIA is not fully developed at this time. As a relatively new effort, VICTORIA benefits from the fission gas release experience of FASTGRASS and the chemistry treatment of VANESA, and VICTORIA has incorporated models directly from the MAEROS code for aerosol behavior. VICTORIA is mechanistic in contrast to the CORSOR empirical model; it treats all fission products, not just gases as treated by FASTGRAS; it handles chemistry effects that are not treated by any other in-vessel code; and it is fully integrated with aerosol behavior models.

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5 FUTURE RESEARCH

If the course of severe accident research follows that of most other applied research activities, a number of topics can be expected to require further activity beyond FY 87.

As the work performed to date is applied to specific plants and to the development of regulations, gaps in knowledge will be revealed in addition to those we already know of; these will need filling. Of course, in this plan we can only address resolution of the issues we know about.

In addition, new applications for nuclear power plant licenses may be forthcoming in the latter part of this decade, if projected needs for power continue along their current trends. Should these applications involve a design substantially different from current plants, then application of current data and methods may reveal the need for further work. In particular, if the proposals now being made abroad to use mixed-oxide fuel for very high burn-ups are introduced into the U.S., extensive additional work will probably be needed. Since aside from the ACRR (suitable for special, separate effects tests) no suitable test facilities will then exist in the U.S., some delay will be inevitable if tests similar to those performed over the last four years in PBF are needed. Either older facilities will be renovated or arrangements will be made to use foreign test facilities, if suitable. Such facilities, each with their own limitations, exist in Canada and France.

The accident management studies just starting in FY 86 should play a major role in the period beyond FY 87, since the analytic tools needed to perform such studies should be well-established by then, even though some codes may not be well tested. A reasonable expectation is that a significant focus will appear on arrested sequences: sequences with a large amount of severe fuel damage, but not gross core melting. If warranted by the need to remove uncertainty, tests simulating such sequences would require a major integral test facility, such as LOFT. (The LOFT FP-tests were examples of such simulations.)

The gaps predicted to exist stem from an examination of the state of the art. During FY 87, the initial critique of NUREG-1150 should have been completed and a detailed understanding established of the initial attempt to estimate the effects of the dominant uncertainties on risk. Since NUREG-1150 will embody the first attempt to make such an estimate it is unwise to predict complete acceptance of the estimates. First, the estimates will themselves be subjected to critical interpretation, since they will be based on the BMI-2104 suite of codes. That set of codes is known to be limited in a number of ways. The newer codes discussed in preceding chapters set out to remedy some of these limitations. Second, the choice of dominant uncertainties will probably be widely questioned, simply because the depiction of containment event trees is also a new item and those trees will be deeply involved with the determination of dominant uncertainties. The net result, however, is that the estimate of the effects of dominant uncertainties will eventually yield a priority scheme for further research. Some of the problems that stand a good chance of being among the dominant uncertainties are: retention vs. vaporization of fission products after vessel failure, e.g. in the steam separators of BWRs; iodine pathways in radioactive environments; the effects of in-vessel flow recirculation; trans-actinide releases, in- and ex-vessel, including plutonium, especially for high burn-up fuel; containment failure modes and response to loads; time and mode of pressure vessel failure; and, (possibly) the core-concrete interaction. With respect to the last item, although a significant amount of work will have been completed by the end of FY 87, the proposed introduction of a mixed atmosphere model can be expected to lead to a need for further testing, especially with the less common concretes. Most of the other topics are not being treated, or else, as in the recirculation and pressure vessel failure questions, extremely difficult to test.

If there are new types of plants submitted for licensing, such as advanced LWRs, then the focus of work may reasonably be expected to shift. The thrust of new design approaches is to establish a very high degree of reliability for shut down heat removal, so that attention may be directed to different types of accidents than are now usually considered.

5.1 Risk Evaluation and Sequence Analysis Research

There is extensive research planned to extend the severe accident analysis work, integrate it with continuing research on improved PRA methods development aimed at reducing or better understanding uncertainties in present PRA techniques, applying the results of the research to day-to-day regulatory applications, and assessing the existing capabilities of various plant designs to terminate or mitigate the consequences of severe accidents (accident management).

The improved PRA methods development activities include the areas of uncertainty analysis, dependent failure analysis, root cause, and cognitive error. The results of such research might need to be applied to the severe accident risk assessments described below.

5.1.1 Severe Accident Risk Assessment

As a continuation of the present six reference plant analyses, work will begin the latter part of FY 86 to assess the risk from a B&W plant, and this work will be extended to a CE plant in FY 87. Also, as research is completed in the phenomenological areas (e.g., experiments involving direct heating and core concrete interactions), the results will be assessed from a risk perspective to determine whether the risk insights drawn in NUREG-1150 need to be altered with regard to either the absolute risk of the plants or the reasonable uncertainty bounds associated with the estimated risk. As more becomes known about the methodology of estimating the risk from external events, a decision will be made whether to extend the plant-specific risk analysis in NUREG-1150 to include the risk from external events (seismic, fire, floods).

Beginning in FY 87 NRR will begin the process of extrapolating the insights gained in assessing the risk from the six reference plants to reactors without PRAs. This extrapolation process will utilize methodology developed by IDCOR, as modified and approved by NRR. It is anticipated that research will be called upon from time to time to assess some of the methodological or risk questions that arise from such analyses.

5.1.2 Regulatory Applications of Risk Information

There are a number of research programs aimed at the effective utilization of risk information in performing regulatory analysis and assisting regulatory decisions. These are summarized below.

The Systems Analysis and Risk Assessment System (SARA) is intended to assist the CRGR in tracking the progress that required/proposed plant modifications did/will make toward improved safety levels. In addition, the system is to be designed as a flexible tool to support different levels of users requiring risk and reliability information for decisionmaking and regulatory analysis. SARA will develop a capability for computation and analysis of information on NPP risk characteristics, using state-of-the-art, user-friendly and modularized computer software and existing NPP risk information developed under current programs. The SARA system is being designed to meet a priority need to improve the way that individual generic issues have heretofore been integrated by various offices in their Regulatory Impact Analyses. This need for an improved integration and cumulative accounting process for the generic issues is discussed in some detail in minutes of CRGR Meeting No. 60. The NRC needs to take into account the fact that the risk base has been changed through imposition of many generic safety issues with time and that the cumulative backfitting burden on licensees in quite large. Also, generic safety issues yet to reach the CRGR review stage will be operating from some reduced base of risk, which could result in net safety benefit which is somewhat smaller than currently envisioned.

The Procedures for Evaluating Technical Specifications (PETS) Program is intended to develop and demonstrate methodologies to utilize reliability and risk techniques in evaluating the scope, detailed requirements, and safety impact of plant technical specifications. The procedures developed are to provide a quantitative basis for making engineering judgments in revising the specifications and in responding to licensee submittals. This program responds to an EDO memorandum of November 14, 1983 wherein RES is to provide research and analytical support to NRR to assist them in carrying out the EDO Task Group recommendations in NUREG-1024 concerning the safety aspects of plant Technical Specifications. This goal is reaffirmed as Recommendation 4 in the final report (Sept. 30, 1985) of the NRR Technical Specifications Improvement Project (TSIP). Interim products are to assist NRR/DSI and NRR/DST in their response to Generic Issues B-56 and B-61 as requested in a June 27, 1984 user need memorandum RR-NRR-8405.

The Plant Risk Status Information Management (PRISIM) is designed to evaluate the use of risk assessment (PRA) results for NRC inspectors and to supply inspection personnel with PRA results in a format that will help them decide where to focus their efforts to reduce plant risk. This information will be displayed on a microcomputer (IBM PC) in a graphical manner. A study to evaluate the results for the Arkansas Nuclear Unit 1 plant is being completed in FY 1986. Current plans for FY 1986 and FY 1987 are to develop a comparable display for the six reference plants of the NUREG-1150 evaluation: Surry, Peach Bottom, Sequoyah, Grand Gulf, Zion, and LaSalle.

5.1.3 Accident Management

The risk attributed to accidents in nuclear power plants is limited by multiple barriers against the release of radiation and by multiple levels of safety. Multiple barriers consist of fuel rod cladding, primary coolant systems, and the reactor containment. Multiple levels of safety are prevention of the initiation of accidents, the mitigation of the progression and consequences of accidents, and the emergency response measures to reduce the risk of exposure to the public. Within this context, accident management is that set of actions taken by the plant operating crew to gain control of the outcome of an abnormal event at the earliest possible time and with the minimum adverse consequences.

Accident management's programmatic origin lies in the phenomenological studies sponsored by the Severe Accident Sequence Analysis program (see Chapter 3). IDCOR and the nuclear industry's research projects also provide a technical basis to develop accident management strategies and to provide evaluation of the effectiveness of the change in risk from application of preventive or mitigative steps.

In September 1984, an initial study was completed and two reports were issued:

Management of Severe Accidents: Perspectives on Managing Severe Accidents in Commercial Power Plants, NUREG/CR-4177, Volume 1, R. DiSalvo, et al., Battelle Columbus Laboratories, and Management of Severe Accidents: Extending Plant Operating Procedures into the Severe Accident Regime, NUREG/CR-4177, Volume 2, J. Wreathall, et al., Battelle Columbus Laboratories.

An extension of the Battelle approach plus additional analysis resulted in an Accident Management Program Plan which had the following objectives:

(a) apply the proposed methodology of accident management analysis of risk to two severe accident sequences of a BWR, Mark I, to identify risk reductions, if any, that result from modest modifications to plant operations or hardware, extension of EPGs into the management of severe accident sequences, and effects of operator training in mitigation strategies;

(b) assess the value/impact of these proposed changes to identify their usefulness, applicability, and estimated cost to the NRC, the licensee and the public; and

(c) provide recommendations for an enhanced NRC capability to respond to expected industry submittals in response to the Commission's Severe Accident Policy.

The Program Plan was documented as NUREG/CR-4148 in August 1985 and was submitted to other Offices for comment. A major consequence of the proposed plan was the decision to perform an accident management analysis of the effectiveness of containment venting for BWR, Mark I. The results of the assessment of the change in risk or radiological exposure that occurs from venting would provide a determination of the probability and consequence of such actions in response to containment over-pressure for two postulated accident sequences, TC and TB, at Peach Bottom Atomic Power Station. Further, the venting analysis' results will be used as part of NUREG-1150 in consequence modeling. Pending final results of the venting project in Spring 1986, verification of the accident management methodology would be performed at other BWR plants.

5.2 Source Term and Containment Loads Research

This topic is divided into three parts: In-vessel research; ex-vessel research; and, code validation and assessment.

5.2.1 In-vessel Research

The issues that persist in this area are: (1) The effects of in-vessel structures, especially with regard to their capability to retain fission products released from the fuel, and even more, with regard to their influence on the initial conditions for containment loading. The latter effects are estimated to have a major and direct influence on total risk; this may be a consequence of our ignorance of the details of the process, or it may reflect an all too real dependency; (2) The effects of water and vapor flows in the reactor coolant system, especially recirculation of steam and revaporization of fission products; these effects affect the timing of the release of radionuclides into the containment and may affect the nature of failure of the coolant system. One such effect would be the failure of steam generator tubes induced by high temperatures induced by recirculation of superheated steam via the steam generators. Such a failure would open a path to the environment that would bypass the containment.

5.2.1.1 Effects of In-vessel Structures

The steam separators in a BWR present a large surface for deposition and, possibly, adsorption of fission products released from the core. Since the steam separators commonly separate steam from water to one part in ten thousand, they should be a good filter for aerosols generated by the melting core. Tests of such filtering action were part of the original MARVIKEN V test schedule. If such tests are not performed there, a substitute will be needed. If the steam separators are as effective as hypothesized, they will constitute an important secondary source of retained radioactivity in BWR severe accidents.

The current efforts to verify the TRAP-MELT code involve small scale tests. Since the phenomena involve both bulk effects (aerosol agglomeration) and boundary layer effects (deposition and adsorption), scaling remains a major problem and some work in this area should continue beyond FY 87. Mediating against this is the possibility that fission product revaporization will render the outcome insensitive to the details of TRAP-MELT.

Below-core structure is expected to play a role in melt progression to vessel failure and in the strength of the in-vessel steam explosion. There are few data available to check estimates for this problem, and no plans are afoot to get any, except through scaled tests up to 100 kg in size. Reliance will be placed on largely untested codes, even though melt progression is the chief determinant of the core-concrete interaction. There are some data from fast reactor experiments and some from TMI-2 that indicate that the following events can be expected to take place:

1. Molten or liquified fuel (fuel dissolved in molten, unoxidized zircaloy) can be expected to collect on cool metal surfaces, freezing there in relatively strong shells capable of bridging gaps of at least a few inches span.

2. As fuel collects and the metal heats up, the structure collapses through loss of strength as temperature increases. The mix falls to the next surface that is cool.

3. For a relatively massive structure that can radiate to a pool of water the structural temperature can be kept below the point of onset of rapid creep

for periods ranging from 30 minutes to several hours, depending on the effective emissivity.

4. If the structure is warm enough, failure as a consequence of rapid creep is predicted. Should there be a massive amount of corium collected at this point, the mechanical forces may induce failure in some below-vessel structures.

5. Speculations have been made that in the case of vessel penetrations through the lower head, the molten core should fail these penetrations and drain through the openings. Calculations with the fast reactor code, PLUGM, which solves this problem, have not been made, but penetrations much longer than 30 cm are unlikely to be effective at draining the core, if fast reactor calculations can be used as a guide. Such failures may, however, furnish an escape path for water still in the lower plenum.

6. If there is a massive collapse into a lower plenum still containing water, a steam explosion is predicted which will probably cause failure of the lower dome of the vessel, rapidly teeming the contents of the vessel into the cavity below the reactor. Otherwise, accumulation of core materials will continue until the lower dome fails as a result of necking down following rapid creep. Then the contents will teem into the cavity.

7. In the case of BWRs, which have a distributed core support furnished by the control rod guide tubes, a reasonable hypothesis is that a less coherent collapse will occur.

In view of the expected sensitivities of source term to containment processes, and their dependence on the initial conditions, it is reasonable to expect that these unsupported speculations will be subjected to tests in the period after FY 87.

5.2.1.2 Water and Vapor Flows in the Reactor Coolant System

Metalographic measurements at TMI-2 have indicated strong recirculation within the core. The codes that are usually used to study core thermohydraulics, TRAC and RELAP, were not designed to handle problems of steam recirculation; such a problem requires a 3-D capability (available in some versions of TRAC) and the proper constitutive equations. (The negative friction factor that appears in some TRAC equations is an artifice that is surely not proper in a steam recirculation problem.) Special purpose codes do exist that can be used, when the proper consititutive equations are used, to model the problem. A reasonable expectation is that such codes can be used to scope the problem and indicate how the TRAC code used in MELPROG can be tailored (through input choices) to attack the problem more generally, with an acceptable approximation. Two special purpose codes currently being used for this task are COMMIX and COBRA-NC.

The major effect of natural convection may be to transport energy from hot to relatively cool structure, causing desorption and transport of previously adsorbed materials. It has been hypothesized that energy transport might be sufficient to cause failure of the reactor coolant system before vessel failure. As mentioned above, this could either lead to a direct path to the environment via the steam generators, or it could depressurize the system into the containment at a slow rate, avoiding failure of a pressurized vessel.

5.2.1.3 Fuel Research

Large scale use of high burn-up fuels (ca. 6 a percent) is expected to start in FY 86 and be common by the end of FY 87. Although these fuels will be UO_2 based, they will, toward the end of life, not be much different in composition from the mixed oxide fuels being introduced in Europe.

Fuel tests are conducted out of pile via indirect heating, or in-pile with direct heating. No facilities are available in the U.S. for in-pile tests except the ACRR; reactor facilities do exist in Canada and France. Instruments to identify released products directly from their radioactive emissions include gamma spectrometers and mass spectrometers. The former can not detect alpha emitters such as plutonium or the chemical compound involved. The mass spectrometer can, over a defined range, identify the mass specific compounds. Their sensitivity and dynamic range are very good, and they are well suited for use with the fluid transport loops characteristic of integral system tests. Such tests will have to be carried out in foreign facilities if PBF and LOFT are decommissioned, as currently planned.

Fuel modelling is largely successful when it can be based on tests of well-characterized samples and accompanied by detailed PIE. Since significant changes in fuel morphology can be expected either from the use of high burn-up or mixed-oxide fuel, fuel modelling work should continue at a steady level past FY 87.

One challenge to successful fuel modeling of trans-uranic release in severe accidents is the combined effect of lattice dynamics and surface conditions on the chemical form and rate of release. Thus, Alexander (BCL) has observed the release of PuO from 39000 MWD/T fuel, and has assigned an activation energy of 146,000 cal/mol to this compound. Ordinarily, one would expect to find Pu as PuO_2. But it is known that nonuniformity dominates calcite-type lattices as they approach solidus. In particular, at temperatures near the solidus, UO_2 lattices appear as though they were mixtures of lattices of a wide range of oxides. Thus, the relative release rate of PuO at 2400 K is reported to be almost an order of magnitude greater than a common surrogate, LaO, and much larger than another, chemically more accurate, surrogate, CeO. At higher burn-ups, or in mixed oxide fuels, this effect should become stronger. Curium also shows an anomalously high release rate as CmO. See also the discussion in Section 5.2.2.2.

A topic of lesser priority, but still important, is more accurate data on hydrogen release rate during cladding oxidation. Integral tests to date have been used to tune the SCDAP code to the extent possible since the data have been difficult to interpret. Tests conducted in NRU are being used to check this feature of SCDAP, but the possible role of steam starvation from coolant channel blocking may be difficult to assess.

5.2.2 Ex-vessel Containment Loads Research

Once the pressure vessel fails, radionuclides and the core's thermal energy are deposited in the containment. Chemical and physical reactions can occur in the containment that either increase the loads (e.g., burn in the containment, increasing the pressure and temperature) or decrease the load, as happens when aerosols agglomerate and settle out. The latter process is inevitable if containment failure is delayed. Hence, there is a major interest in developing an understanding of those processes which can cause early containment failure.

The Commission has directed the staff to prepare a containment criterion. Regardless of the final form of such a criterion, the data and codes produced under the activities described here and in Chapter 3 will form the basis for regulation under that criterion.

5.2.2.1 Initial Events - Direct Heating, Steam and Hydrogen Spikes

When the contents of the pressure vessel are suddenly teemed into the cavity, small scale experiments indicate that the escaping gas will atomize large amounts of melt before the vessel is depressurized. (Unless, of course, the vessel was depressurized earlier.) The fine particles (20 to 300 microns) will either transfer their intrinsic heat to the containment atmosphere rapidly, or they will burn. In addition, if the accumulators now dump onto the molten material, large amounts of steam (the steam spike) are expected to be generated in a few minutes. The hypothesis has been that the steam will inert the atmosphere against hydrogen burning, but this is by no means certain. The main effect of the steam is as a coolant, and if dynamic effects can produce a flame in a local volume, or the atmosphere is hot enough to superheat volumes of steam, hydrogen could burn, adding to the pressure and temperature load. These load combinations can be sufficient to cause early containment failure in the smaller and/or weaker containments.

Although small scale experiments have been carried out, and models derived that are fairly transparent, the gains to be achieved from reducing the uncertainty in prediction of early containment failure are such as to indicate continued efforts in this area.

5.2.2.2 Core-Concrete Interactions

A very substantial portion of the work projected for this topic will be complete by the end of FY 87, and reflected in the CORCON and VANESA modules of the CONTAIN code. The picture that has emerged to date is that when corium and other molten materials are teemed onto concrete the resulting interaction generates large quantities of aerosol, combustible gases, and, as indicated by chemical kinetics calculations, release of fission products not volatilized earlier. The strength of the interaction is sensitive to the type of concrete used; some concretes, typically basaltic concretes, yield relatively little interaction as compared to mere common limestone concretes. Tests in Germany with molten steel, alumina, and CaO yield very little interaction with concretes typically used in reactors in Germany. This difference is variously ascribed to the nature of the concrete and the temperature of the reactants. It is known from earlier studies that the interaction of corium with alumina concretes is likely to be very low. There remain some major uncertainties that affect the source term markedly. The temperatures reached in the interaction are thought to be very important in determining the release rate of the transactinides. It is unknown how sensitive this rate is to the chemical form of the transactinide. In particular, if PuO is released rather than PuO_2 and if CmO or Cm_2O_3 are preferred forms, the rate may not be as sensitive to the peak temperature as earlier studies indicate. If cerium is a valid surrogate for plutonium, the chemical form in the melt makes little difference in VANESA as it is currently written. The radiological virulence of the transactinides, especially in the form of soluble or breathable particles is such that it is important to know the fate of these species to reasonable accuracy; how accurate, say a factor of two versus 25 percent, will be judged by follow on studies to NUREG-1150. Such studies will be carried out after FY 87, if they are carried out at all.

Studies based on the use of lower burn up fuels and the projected release of 1% of the available plutonium as PuO_2 are likely to be deceiving because of the lower Pu inventory and the different chemical form. (PuO_2 is insoluble and is assumed to agglomerate into non-inhalable particles.)

The need to know the transactinide release is indicated by comparing the total potential dose by inhalation of I-133 and Pu-241. The total potential dose is 34000 mega rem for the former; 5,250,000 for the latter. Thus, in order to neglect Pu-241 against I-133 in source term calculations, it is necessary to know that the release of Pu is much less than 1 percent of the release of the I-133. The effect will be more striking for high burn up and mixed oxide fuels.

Aside from the difficulty of getting good basic data on the chemical kinetics of the solutions in the core-concrete interaction, there is the basic problem of determining the mixture in which the radionuclide is found. In the current model of the VANESA code the assumption is made that the melt consists of two condensed phases, and oxidic phase and a metallic phase. The phases, for the most part, are assumed to be located in a vertical stack.

It is known that such a description is adequate for many tests and completely inadequate for others, depending on temperature and composition. In any event, the uranium, and presumably the plutonium will emerge from the vessel in hypo-stochiometric forms, probably grossly so for the transactinides.

In the model based on two condensed phases, it follows that metallic uranium and transactinides will appear in the metallic phase, probably in significant amounts. The VANESA model assumes that the oxygen potential in the two phases are equal.

The speciation of the elements as they are transformed from liquid to vapor is largely based in VANESA on the known vapor species in the M-O-H system, where M is the element in question. It is felt that this model is probably weakest in the area of vapor phase hydrides. Moreover, the VANESA code does not currently model transactinides explicitly. Lanthanum is sometimes used as a surrogate, but this is certainly incorrect, and the use of Cerium as a surrogate is probably somewhat better.

Two efforts are needed to remedy these problems. One is to replace to separate condensed phases model with a model involving a law of atmospheres which proportions components properly. Such an effort is currently in the conceptual stage. The other is to model the transactinides explicitly; that effort is not yet planned.

Clearly, the problem of transactinide release will not be resolved until after FY 87. Tests with fission product mockups will probably be needed; some may be done before the end of FY 87.

5.2.2.3 Gas and Aerosol Transport

The CONTAIN code, and the hydrogen burning sub-code, HECTR (also usable by itself) are partially verified now, and current test programs should verify these codes at a scale intermediate to past tests and full scale plants. There are no major problems known at this time except for the problem of flame acceleration and the longer term gas stratification. With respect to the latter, the assumption is made that an initially well-mixed atmosphere in a typical containment will stratify very slowly, if at all. Although plausible, this assumption should be checked against industrial experience.

No substantial program beyond FY 87 is foreseen in this area except for special efforts related to applications to equip· nt qualification.

The part of the risk assessment code, MELCOR, that addresses containment loads is an adaptation of CONTAIN.

5.2.3 Code Validation and Assessment

The processes of code validation and assessment are to assure that the codes solve the problems they address in a mathematically correct way, and that the predictions are a reasonably accurate reflection of what is obtained in nature. The past four years have seen the development of a number of parts of two sets of major codes that will be the mainstay of future work. These parts, or sub-codes, are described in Chapter 4.

5.2.3.1 PRA Codes

One of the two sets of codes is the PRA code MELCOR, designed to replace the MARCH, CORRAL, and CRAC codes used in the WASH-1400 study. A practical guideline established for MELCOR is that a representative problem be executed with a CPU time of no more than an hour or two on a top of the line computer (such as the Cray XMP). This guideline restricts the complexity of computational models incorporated into MELCOR. When the mechanistic sub-codes described in Chapter 4 conflict with the running time guideline, a parametric model is substituted. Otherwise, the mechanistic code is adapted. At present, it has been possible to adapt the CONTAIN code (with such sub-codes as CORCON, VANESA, MAEROS, HECTR, etc.) into MELCOR.

The reason for the guideline on execution time is that to meet the needs of users, who typically need to complete a study in 4 to 6 months, the total computation time should average about 10 percent or less of the staff analyst's time. As experience is gained with the mechanistic codes, and they are fully assessed, they can be adapted into MELCOR in either of two ways:

1. By generation of tables of output data indexed by input variations, since table look-up can be made a very efficient computation.

2. By straight-forward adaptation to take advantage of major advances in computing hardware capability.

Thus, assessment of the physical adequacy of MELCOR rests on the assessment of the mechanistic codes.

5.2.3.2 Mechanistic Codes

The mechanistic sub-codes described in Chapter 4 are assigned to either of two major codes: MELPROG (in-vessel) or CONTAIN (ex-vessel). This topographically impeccable ordering also accentuates a great gap: Containment Response. At the present time there is no code useful over the entire range of problems expected to arise in estimating containment behavior. This is discussed in the following paragraphs.

The practice in validating the MELPROG and CONTAIN codes has been to validate individual subcodes, hoping to validate and assess the entire code via integral system tests. This is the normal process of code development, earlier exemplified by the construction of the TRAC and RELAP codes. For the latter codes the LOFT facility served as an integral system test; it is not clear what facility, if any, could serve such a purpose for MELPROG.

The CONTAIN code is being assessed against some integral tests, none of which incorporate all the major effects, especially the core/concrete interaction.

Code assessment work can be expected to continue for some time, especially in response to needs highlighted by applications to plants and regulatory issues.

5.3 Containment Behavior Research

The complement to an estimate of the containment load is the response to that load. If a sensible criterion is to be implemented, the response needs to be known as well as the load.

This element of work attempts to identify when and how the containment may be expected to fail under the estimated loads, and how equipment associated with the containment, such as fans and filters, will work.

Filters of the sort usually found in reactor containments will almost certainly clog early under the aerosol load; no substantial work is projected. If a filtered vent scheme is proposed for the weaker containments, as in Sweden and France, this problem will have to be revisited in the time period beyond FY 87.

Fan loads are affected by pressures developed by accelerated hydrogen flames induced in the fan's wake. For Sizewell-B, the CEGB have called for strengthened blades to withstand the projected loads. No research beyond FY 87 is indicated, except as may arise in the context of accelerated flame fronts.

5.3.1 Containment Structural Response in PWRs

The current program that addresses the response of containment structures to pressure is described in Chapter 3 of this report. That program will probably continue well beyond FY 87. Separate effect tests can scope the extent to which radioactivity, humidity, and elevated temperature play a role, especially with respect to integrity of seals around major penetrations. If it should appear that there are competitive effects, or interactions, some program expansion may be needed. This is especially so since the major program is to test the ability of current codes to predict the failure of readily modeled structures. The inclusion of complex penetrations on a similar basis is a major undertaking if environmental conditions are to be included. If the separate effects tests on these problems are to be credibly incorporated, special demands on code capability may arise that will have to await the development of advanced computers.

5.3.2 Containment Structural Response in BWRs

No program is in place that directly addresses the problems faced in estimating the response of BWRs. The fundamental idea of BWR containments is to vent steam to a suppression pool instead of a turbine. They are not well adapted to coping with large amounts of molten fuel, or with suppression pools at saturation temperature. In many older model containments, pathways for fission product transport that bypass the pool have been identified. The newer models of containment bear some resemblance to PWR containments. The tests of steel containment vessels described in Chapter 3 serve as a benchmark for modeling remaining the reponse of the steel shells found in BWR designs, but that is not the major problem, since current estimates point to early failure in many cases for older (Mark I) containments.

Setting priorities and developing a coherent program for these reactors will depend on a study such as NUREG-1150, plus some work under Severe Accident Sequence Analysis that is currently in progress.

Work can be expected beyond FY 87, but program details have yet to be established.

5.3.3 Analysis of Containment Peformance

Even if the currently planned work addressing containment performance establishes the credibility of current methods (chiefly finite element models of the stress-strain distribution in a structure), the need will still exist to establish the credibility of the representation of complex features, such as penetrations. But it is not reasonable to expect such work to establish credibility in the same sense as other mechanistic codes.

The reason for this is that failure of even a simple, homogeneous material, is not readily defined. The strain observed in response to applied stress is frequently a function of the rate, the level, and even the history of the stress application. The strain at failure in some test vessels under very rapid loads was reported in the journal "Nuclear Safety" in 1966 to range from 18 percent to 21 percent; yet under steady, slowly applied loads, strains at failure may be of the order of only a few percent. The current process for predicting failure is:

1. Run the problem for a series of increasing loads until the code fails to converge.

2. Plot several of the points preceding these loads and from extrapolation of that plot, estimate the failure load.

This process is not suitable for automated representation. That doesn't mean the process is unacceptable; but it does present a problem in assessment, since the estimator is now part of the assessment.

Thus, the problem is not purely one of securing advances in computational methods and hardware. Significant engineering physics advances in the description of materials behavior in the extreme ranges of plastic deformation are needed.

It is known from current experience that complete representation of a containment system in a numerical computation requires more hardware capacity than is available right now.

5.4 Future Program Costs

The chief uncertainty in projecting future program costs is the possible need for further in-pile tests. Past U.S. experience has been that such tests are very expensive, with about one half the cost assigned to base costs for plant operation, and one half to test trains and instruments. Experience in France is that base costs are much lower, but these are with plants that do not operate continuously. Part of the U.S. cost experience involves a learning experience, which, if records are well kept and personnel maintained, need not be repeated. In the similar Franco/German experience, about one third of the total project cost can be ascribed to the learning phase.

On the basis of these considerations, any future in-pile test program should be expected to run about $15 to $25 million dollars per year, depending on the test reactor site. This is for a schedule of four tests per year.

The costs of the other efforts are fairly evenly divided among the major components of the problem and are estimated on a manpower basis which assumes current experience is valid. The total estimate is $15 to $20 million dollars per year for about three years. Any significant test facility costs would raise this figure. On the other hand, as some of the shorter term work is finished, manpower costs should drop.

APPENDIX A

SUMMARY OF MAJOR TECHNICAL ISSUES

In the following paragraphs a number of specific technical issues are identified that are either major contributors to the uncertainty in plant risk or represent significant differences in modeling assumptions between the NRC and IDCOR. It is important to identify another difference between NRC and IDCOR methodologies that is more basic. In the Severe Accident Risk Rebaselining and Risk Reduction Program (SARRP) uncertainties in phenomenological behavior are treated explicitly. Not only is a best estimate analysis performed, but pessimistic and optimistic analyses are also performed to provide an understanding of the range of possible outcomes. Although some sensitivity studies have been performed in the IDCOR program, their purpose appears to be to support the selection of assumptions leading to point estimate results. It is the opinion of the NRC and its supporting contractors that the uncertainties in severe accident processes are quite large and that a meaningful evaluation of plant safety and the possible need for plant modifications must include the explicit consideration of uncertainties.

One significant source of uncertainty is the definition of the accident sequences. In several cases where IDCOR and NRC have analyzed the same or similar sequences, large differences in the calculated consequences have resulted because of different assumptions about the sequence of events in the accident. These differences often reflect a real uncertainty in the accident sequence definition. The NRC believes that it is particularly important to reflect these uncertainties in severe accident analyses.

I. CORE HEATUP STAGE

ISSUE #1 - Fission Product Release Prior to Vessel Failure

Uncertainties in fission product release prior to vessel failure can affect both the timing of release and the total quantity of release from fuel. Whether the fission products are released in-vessel or ex-vessel can be particularly important because of the potential for retention of fission products on reactor coolant system structures for the in-vessel release component. Several sub-issues contribute to this issue; namely

(a) Assumed temperature for beginning of fuel relocation

IDCOR models assume and employ a relocation temperature of 2800C (3100K); i.e., the melting point of uranium dioxide. This high assumed temperature in conjunction with their fission product release model (see below) allows all volatile fission products such as I, Cs, and Te to be released prior to core slumping and quenching in the lower plenum. As a result 100 percent of these F.P.'s are available to be deposited in the upper plenum and other structures of the primary system, thereby resulting in possible non-conservatism in the source term to the containment for certain sequences.

The BMI-2104 models assume that melting occurs at an average eutectic temperature of 2550K. In conjunction with the empirical CORSOR release coefficients, this temperature is high enough to release essentially all of the volatiles (except tellurium) in-vessel, but may underpredict the release of nonvolatile materials. A more mechanistic fuel melt progression model is being developed for the MELPROG coue based on experimental evidence from KfK, PBF, and ACRR.

(b) Modeling of in-vessel release of fission products

The IDCOR fission product release from fuel model is based upon the oxidation of the fuel by steam. The model assumes that sufficient steam and contact area (with the fuel) is present at all times during the heatup to oxidize the fuel to a higher state, thereby significantly enhancing the release of fission products. Each of the species considered is released from the fuel at the same rate. Partial pressures of the vapor species are used to determine the amount of released material that can be transported as a vapor. The condensed component is apportioned between aerosol and fuel surfaces according to an input factor.

The BMI-2104 model, CORSOR, is entirely empirical based on a variety of in-pile and simulant experiments. The BMI-2104 model does not attempt to account for differences in mass transfer limiting processes that may exist between the experiments and the real system.

The PBF experiments indicate that the CORSOR and IDCOR models both probably overestimate in-vessel release of fission products. The VICTORIA code which is being developed as an element of the MELPROG package will provide a more mechanistic model for fission product release which can be compared with integral experiments in PBF and ACRR.

A more recent finding indicates some questions as to the chemical form of the released iodine; i.e., as C_sI or some other more volatile form. Experiments are now underway at SNL to resolve this aspect.

(c) Tellurium retention in-vessel

A major difference exists in the treatment of tellurium (Te) release between the IDCOR model and the BMI-2104 model. Experiments at ORNL and at the PBF strongly indicate that the tellurium release is reduced during this stage of the accident if approximately 25 percent or more of the zircaloy remains unoxidized. Because of the rapid heat up of a core during boiloff, most sequences result in unoxidized zircaloy contents greater than 25 percent. Therefore, in most sequences, the CORSOR release model allows most of the Te to be retained in the core while it is in the reactor vessel. The significance of this effect is that if the Te is released early - as IDCOR advocates - it will be deposited in the upper plenum and not be available in the source term estimate; however, if it is retained in the molten fuel at this stage - as experimental evidence indicates - it will be released when the core exits the R.V. and interacts with the concrete. Thus, the Te would be released to the containment volume without the potential for being deposited in the reactor coolant system and, perhaps, at a time closer to the time of containment failure. Although the CORSOR model accounts for the interaction of tellurium with the zircaloy cladding, the model is crude and the supporting data are sparse.

The overall significance of this issue is that, for certain sequences in both PWR's and BWR's, significantly higher amounts of volatile fission products (I, Cs, and Te) may be available for release to the containment during the core/concrete interaction stage than computed in the IDCOR models. Depending on the containment failure time, this effect can and does lead to much larger source terms than computed by IDCOR. The PIE of the PBF tests and results from small separate effects tests at BCL should help clarify this issue in FY 1986.

ISSUE #2 - Recirculation of Coolant in the Reactor Vessel

Neither IDCOR nor current NRC (BMI-2104) models are able to calculate recirculation patterns of steam in the R.V. after core uncovery. Based on simplified analyses at SAI (EPRI) and Purdue, it appears that recirculating flow could have a significant effect on the core heatup behavior of PWR's at least for high pressure sequences. It is speculated that recirculating flow can affect the core heatup rate, in-vessel release of fission products, quantity of hydrogen produced in-vessel, structure temperatures, and deposition of fission products in the reactor coolant system. Analytical studies have been initiated with the COBRA, COMMIX, and TRAC codes to resolve this issue. Preliminary studies will be completed in FY 1986. Experimental results are being obtained by Westinghouse which should offer an opportunity for model validation.

ISSUE #3 - Release Model for Control Materials

A major discrepancy between the IDCOR and the original BMI-2104 models is the mode of behavior of silver and cadmium during core meltdown. The IDCOR model allows the control rod material to rapidly melt and runoff to cooler regions of the core where it freezes and takes no part as a potential source of inert aerosol material. In contrast, the BMI-2104 models ignore the potential for runoff of liquified control rod material prior to regional slumping and, as a result, may tend to overpredict the release of control rod silver and cadmium. Later versions of the BMI-2104 models as used in STCP attenuate the amount of silver alloy that is released, but the release is not reduced to zero.

Neither the IDCOR or BMI-2104 analyses consider the possibility of chemical reactions with boron carbide control material that could result in increased hydrogen production or changes in the chemical form of fission product species.

The SFD-4 experiment in PBF is designed to study questions related to the behavior of silver-indium-cadmium control rods. Investigations at ORNL are addressing boron carbide control material behavior.

ISSUE #4 - Fission Product & Aerosol Deposition in the Primary System

The uncertainties in the existing predictive capability (TRAP-MELT 2 is the most detailed of the methods available) for deposition in the reactor coolant system are very large. The vapor and aerosol transport processes are complex and interacting. The upper plenum geometries and flow regimes are treated in a crude approximation of the prototype. In initial discussions with IDCOR, the focus of NRC concern was the use of a log normal aerosol size distribution in the RETAIN code. This formulation has now been replaced with an empirical aerosol model, the validity of which is of equal concern in the RCS geometry.

Results of verification experiments at ORNL and the integral Marviken tests will provide a means to evaluate the existing models.

II. MELT PROGRESSION AND FUEL RELOCATION STAGE

ISSUE #5 - Modeling of In-Vessel Hydrogen Generation

Substantial agreement exists in the modeling of the steam/zircaloy reaction in the reactor core as long as the original geometry is maintained. However, when cladding melts and slumps and flow channels begin to block, issues arise as to when blockage will occur, how effective it will be, whether cladding material will run out of the hot zone, how much oxidation will occur as it moves, and whether the relocated cladding will subsequently be reheated and exposed to steam. The IDCOR models employ two parameters: a metal-water reaction cut-off temperature and a flow blockage parameter that effectively limit the extent of hydrogen production that occurs in-vessel. Similarly the BMI-2104 code, March 2, has externally controlled parameters that can limit the extent of metal/water reaction. In practice, however, the "best-estimate" assumptions made by the IDCOR and BMI-2104 analysts have lead to substantially different results.

For a PWR SBLOCA sequence without ECCS, IDCOR models calculate 200 kg of hydrogen produced, whereas, the the BMI-2104 calculation yields 450 kg of hydrogen; a significant difference of a factor of 2.25. Experimental data of hydrogen production after fuel relocation is difficult to obtain, but careful analysis of the PBF Phase I program and planned experiments in the NRU facility wherein full-length elements will be tested to slumping and held for long times will help to resolve this issue. The MELPROG code will employ more mechanistic fuel slumping models and a two-dimensional fluid flow model which will also provide a better understanding of the issue.

ISSUE #6 - Core Slump, Collapse, & Vessel Failure Models

The IDCOR and BMI-2104 models of core slumping are greatly simplified representations of very complex processes. Some key features of the models are:

(a) IDCOR assumes that molten core material becomes isolated from the rest of the system until a slumping criterion is satisfied and then it instantaneously slumps to the lower core support plate (LCSP). The NRC model treats approximately the effect of in-core fuel relocation and growth of the molten region.

(b) IDCOR models assume failure of the LCSP after a user-input fraction of the core is molten. NRC models calculate failure of lower core support structures due to heating by slumped core material.

(c) IDCOR assumes immediate vessel failure after failure of LCSP. The NRC model permits the user to select early local head failure or later gross head overheating. The condition of the core material (mass, composition, and temperature) at the time of vessel failure can have a major influence on the subsequent loads on containment (steam spike, direct heating) and the extent of dispersal of the core debris.

The best hope for resolving this issue is to perform detailed best-estimate calculations of melt-progression using the MELPROG code. Since integral experiments in this area are clearly impractical, the code models will have to be validated with data from the ACRR debris formation experiments scheduled to be completed in FY 1986.

ISSUE #7 - Alpha Mode Containment Failure (Steam Explosions)

The major significance of this issue is whether sufficiently energetic molten fuel/coolant interactions can occur in the lower plenum to produce a missile of sufficient energy to breach the containment (commonly referred to as the alpha-mode failure). The general consensus on this issue is that alpha mode containment failure has a low probability. Recent work by T. Theofanous, UCSB and by C. Bell and W. Bohl, LANL has indicated that in the event of an energetic in-vessel steam explosion the more likely mode of vessel failure would be bottom rupture rather than an upper head failure which would generate a missile with sufficient energy to fail containment.

Work is continuing to resolve this issue with experiments at SNL, at ANL, and in the UK, and by better, more-detailed calculations of the masses involved. A report on this subject was issued in March, 1985 by an NRC-sponsored expert review group (NUREG-1116).

III. EX-VESSEL STAGE

Issue #8 - Direct Heating of Containment

This issue is critical to containment failure timing for high-pressure sequences, such as TMLB', in large-dry PWR containments. In the IDCOR calculation of the TMLB' sequence for the Zion Plant, it is assumed that half of the molten material ejected from the vessel under pressure is swept out of the reactor cavity onto the containment floor. The analysis does not account for potential rapid heating of the containment atmosphere by the core debris or the oxidation of the core debris during transit and further heating of the atmosphere. The potential for "direct heating" of the containment atmosphere is being investigated by the NRC in an experimental program which is underway at SNL. Results will be available in FY 86.

The issue of direct heating is related to Issue #2. If recirculating flow patterns within the reactor coolant system result in alternative failure locations such as the hot leg, then the RCS will depressurize prior to meltthrough of the lower head and the core debris will not be dispersed from the reactor cavity. Investigation into phenomena which would lead to primary boundary failure and system depressurization prior to core melt are not expected to eliminate the residual probability of high pressure melt ejection. Research to quantify this threat is expected to extend into FY 1988.

Issue #9 - Ex-Vessel Fission Product Release

A potentially significant modeling difference between the IDCOR model and the NRC models (CORCON and VANESA) is related to the release of refractory fission products during the core/concrete interaction process. The NRC models allow for the production of volatile oxides and hydroxides of these fission products by reaction with steam and carbon dioxides sparging through the melt. Uncertainties in the prediction of the ex-vessel release of fission products involve the composition (masses of materials and oxidation state) at the time of vessel failure, initial temperature of the melt, extent of core dispersal, modeling of core-concrete attack as well as the complex chemical behavior within the melt. Experiments are planned at SNL in which simulant fission products will be included in core-concrete attack tests. These data should provide a basis for testing CORCON and VANESA modeling assumptions by late 1986.

Issue #10 - Ex-vessel Heat Transfer Models

This issue is associated with the magnitudes and mechanisms of energy transfer from the molten core debris to the concrete, to the containment atmosphere, and to overlying water pools (if present). The issue can potentially impact the mode and timing of containment failure and the chemical forms of fission products. This is an area in which major differences exist in the modeling assumptions used in the IDCOR and BMI-2104 analyses. In general, the BMI-2104 analyses involve more heat going into concrete attack, more rapid production of noncondensible gases, more rapid pressurization of the containment, but lower atmosphere temperatures, particularly in the BWR analyses. These differences are partly the result of assumptions made regarding debris dispersal or spreading and, partly, differences in core-concrete attack models. When water is present in the cavity, it is assumed in the IDCOR analyses that a coolable debris bed will form. In the BMI-2104 analyses this possibility was treated parametrically. Core-Concrete tests in the BETA facility and at Sandia (involving sustained urania/concrete heating tests) in FY 85 and FY 86 will provide an expanded data base for improving and validating models.

Issue #11 - Revaporization of Fission Products

Both the IDCOR and NRC models for in-vessel transport and deposition of fission products predict that a large fraction of fission products released from the core can be deposited on structures in the reactor coolant system. Associated with these fission products is a significant portion of the decay heat with the potential to result in direct heating of structures and the revaporization of deposited fission products. In the most recent IDCOR analyses revaporization is taken into account resulting in some enhancement of the environmental source term for some reactor types and accident sequences. A number of major uncertainties remain, however, including: a very sparse data base related to the mechanisms of reaction between fission product species and surfaces (and the potential for revaporization), the modeling of natural convection driven flow patterns in the reactor coolant system prior to, and subsequent to, lower head failure and the behavior of RCS insulation material in the accident environment. Coupling of the TRAP-MELT and MERGE codes has been achieved to permit analysis of the effects of revaporization during the period prior to vessel meltthrough and in a parametric manner after meltthrough. A more rigorous treatment will be provided by the coupled RELAP5/SCDAP/TRAP-MELT code. Results of analysis should be available in FY 86.

Issue #12 - Containment Deposition Model for Fission Products

Because of the limitations in the applicability of the log-normal size distribution assumption for aerosols, which was incorporated in the RETAIN code, IDCOR has developed an empirical correlation for aerosol settling which has been included in the MAAP code. This new model has also been criticized for not including simultaneous removal processes, and "burying" particle size distributions within the correlation. This issue is related to Issue #4. Additional direct comparison calculations should be made for specific plant/sequence events using both the NRC and IDCOR models.

Issue #13 - Suppression Pool Bypass

Analyses of suppression pool scrubbing in the BMI-2104 study indicate that the effectiveness of suppression pools is quite good even when the temperature of the pool is at saturation. In general, the extent of pool bypass was found to be a more important source of uncertainty than the details of suppression pool modeling. In the IDCOR analyses suppression pool decontamination has been treated simplistically using constant decontamination factors and ignoring bypass under the assumption that bypass leakage would become plugged by aerosols. The potential for bypass is quite plant-design dependent. However, since pool bypass will govern the magnitude of environmental releases in Mark III BWR designs, it should be explicitly considered in the analyses for this type of containment.

Issue #14 - Modeling of Emergency Response

The predicted consequences of severe accidents can be very sensitive to the modeling of emergency response. This is particularly true for early fatalities because of their threshold nature and the high dependence of dose to proximity to the release point. In both the NRC and the IDCOR analyses, the population in the evacuation zone is removed at a given rate and assume a 5 percent straggler population that is slow to evacuate or refuses to evacuate.

This issue is now considered to be resolved.

Issue #15 - Containment Performance

The potential magnitude of the source term to the environment is largely controlled by the mode and timing of containment failure. If the containment remains intact for a number of hours following melting of the core and the release of fission products, the potential consequences will be substantially reduced either through natural deposition processes or by the action of containment safety features such as sprays, coolers, suppression pools, and ice condensers. Over the past few years, the analyses of containment performance and model experiments at Sandia National Laboratories have verified the expectation that substantial margins exist between the design conditions and conditions at which major leakage of containment structures can be expected to occur. Considerable uncertainty still exists, however, as to how rapidly leakage will grow as a function of pressure and temperature for different containment designs, the mode by which a containment will fail, and the location at which failure will be initiated. In the IDCOR analyses, containment failure is typically characterized by a leak rate sufficient to prevent further pressurization. This assumption has the effect of increasing the residence time of fission products in the containment and enhancing the effectiveness of evacuation. In the BMI-2104 analyses, failure is typically characterized as a large hole that leads to depressurization of the containment, but does not involve total destruction of the building. Sensitivity studies have also been performed for leak rates that vary as a function of internal pressure.

The importance of better determining the conditions leading to containment failure is plant and sequence dependent. Predicted consequences for a large, dry containment (such as Zion) can be insensitive to the pressure at which failure occurs, if the potential for early failure can be precluded. In contrast, the Mark I source term results are found to be very sensitive to the location of failure and the conditions leading to failure.

Experimental programs at Sandia will provide important data on the behavior of gasketed penetrations, electrical penetration assemblies, and the structural response of concrete containments. These, combined with further plant-specific analyses, should serve to resolve the outstanding issues.

Issue #16 - Secondary Containment Performance

The IDCOR analyses tend to give greater credit for secondary containment performance than the BMI-2104 analyses. The differences are particularly evident in the IDCOR V sequences in which considerable deposition is predicted to occur in the auxiliary building even in the absence of water pool scrubbing. It is also evident from ORNL SASA analyses that retention of fission products by the reactor building surrounding the Mark I primary containment could mitigate the consequences of severe accidents under a given set of conditions or assumptions. The principal uncertainties regarding the effectiveness of secondary containment buildings relate to: the mode of primary containment failure and its impact on the survival of the secondary building; potential for hydrogen deflagrations to occur in the secondary building; and the modes of leakage and failure of the secondary building. Secondary building performance is of particular interest in accident sequences involving early failure or bypass of the primary containment envelope and, thus, has the potential to mitigate consequences. The uncertainties in secondary containment performance could be reduced by additional analyses since the processes of interest are essentially the same as those in the primary containment for which methods have been developed. To the extent that the mode of failure of the primary containment influences the performance of the secondary buildings, the uncertainty will always remain large.

Issue #17 - Hydrogen Ignition and Burning

There are substantial differences between the IDCOR and NRC treatment of ignition, burning, and flame propagation in air/hydrogen/steam mixtures. IDCOR analyses base ignition on a calculated flame temperature criterion which is a function of the composition of the atmosphere within the compartment. NRC analyses, in addition to considering hydrogen and oxygen concentrations within a compartment, give explicit consideration to steam inerting and the availability of ignition sources. The IDCOR models appear to predict continuous burning in essentially all cases, whereas the NRC treatment tends to predict a number of discrete burns. The NRC's approach tends to allow the buildup of higher hydrogen concentrations and hence can lead to the prediction of higher containment pressures. The differences between the IDCOR and NRC treatment of hydrogen ignition and burning are particularly pronounced in multi-compartment systems, such as the ice condenser containment, and in the absence of deliberate ignition. A number of related sub-issues are noted below.

Effect of Natural Convection

IDCOR models include consideration of natural convection driven flow between compartments which appears to lead to enhanced hydrogen burning at low concentrations. NRC models do not include consideration of natural convection flows.

Flame Propagation

Due to a combination of differences noted elsewhere, NRC treatment of hydrogen combustion indicates greater likelihood of flame propagation into the upper compartment of the ice condenser containment where impact on containment pressurization is the greatest.

Potentially Detonable Concentrations

NRC's treatment of hydrogen combustion indicates the possibility of developing potentially detonable compositions in local areas, e.g., the upper plenum of the ice condenser. IDCOR's treatment appears to preclude such localized hydrogen buildups.

Effect on Chemical Form of Fission Products

There is experimental evidence that hydrogen combustion can alter the chemical form of airborne fission products, e.g., release of molecular iodine from cesium iodide aerosols due to hydrogen flames. It is not clear whether such changes in chemical form increase or decrease the consequences of severe accidents. Neither IDCOR nor the NRC analyses at present consider such changes in fission product chemistry.

Resolution of the outstanding issues will come, in part, from continued comparisons between experiments and analyses. It must be recognized, however, that experimental data may not be available to address issues related to burning in complex geometries.

Issue #18 - Essential Equipment Performance

Neither the BMI-2104 nor IDCOR analyses have provided a detailed assessment of the ability of essential equipment to survive and operate (if power is available) under the conditions associated with a severe accident environment (high temperature, high humidity, pressure differentials, flames, high radiation, and high aerosol loadings). This equipment has been qualified to survive the LOCA environment which is in many ways similar. However, experiments at Sandia have indicated that the more severe environment of core meltdown accidents could lead to the degradation of equipment, particularly electrical cables. Once substantial core degradation has occurred, protection of containment integrity becomes the key safety function.

Whether or not containment safety equipment is vulnerable to the severe accident environment depends on the design of the plant. Not only does the issue involve questions of the performance of the equipment, but also of the ability of the operator to monitor conditions in the containment and to control essential equipment. Thus, the degradation of monitoring and control systems could also potentially degrade the effectiveness of the containment or accident management strategies to protect the public.

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16. ABSTRACT (200 words or less)

Over the past six years a major research effort has been underway by the NRC to develop an improved understanding of severe accidents and to provide a technical basis to support regulatory decisions. A Severe Accident Research Plan (SARP) was issued in January 1983 as NUREG-0900. The purpose of this revision of the SARP is to describe current plans for the completion and extension of this research. The plan is focused on technical issues and does not have as a purpose the justification of the need for the research programs. This justification is provided in the Long Range Research Plan (NUREG-1080). The discussion of the research to be accomplished during 1986 and 1987 is presented in this report in three general areas, each of which is divided into two subareas. The subareas of research are (1) risk reduction and evaluation, (2) severe accident sequence analysis, (3) in-vessel melt progression and fission product behavior, (4) containment loads and ex-vessel fission product behavior, (5) containment performance, and (6) equipment survivability. Chapter 3 of this report discusses for each subarea the scope of research, the research accomplishments to date, the outstanding issues, and the planned activities. The plan reviews the 18 major NRC/IDCOR technical issues. The plan also covers the eight major areas of uncertainty in source term analysis identified in NUREG-0956.

17. KEY WORDS AND DOCUMENT ANALYSIS

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Severe Reactor Accidents	Sequence Analysis
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Outstanding Issues	Melt Progression
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