EPRI Nuclear Safety and Analysis Research Program

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FOREWORD

A primary concern of the Electric Power Research Institute has been to effectively transfer the technology developed through its R&D efforts to the electric utility industry and to the technical and scientific community at large. In the safety and analysis area, close contacts of EPRI personnel with utility representatives, periodic meetings with advisory bodies, and dissemination of reports to interested practitioners within each technical area in the field have been the instruments for technology transfer to the utility industry.

A need exists, however, for communication with the broader scientific community, which has a strong interest in and which makes substantive contributions to the cause of nuclear power safety. Thus, a comprehensive article covering the activities of the Nuclear Safety and Analysis Department was published by one of us (WBL) in the November-December 1975 issue of *Nuclear Safety* magazine.

In order to update that article and to inform the nuclear safety technical community of developments, expansion, recent major accomplishments, and overall direction of the EPRI Safety and Analysis program, we were invited to submit a new, comprehensive, and detailed report. Our effort became the lead article in the July-August 1980 issue of *Nuclear Safety*. We extend our thanks to the editor of *Nuclear Safety* for permission to reprint that article here as an EPRI special report.

January 1981
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ABSTRACT

The motivations, objectives, structure, and current status of the Electric Power Research Institute (EPRI) Nuclear Safety and Analysis Research Program are presented. By using established experimental and analytical techniques and by developing new ones, the program aims to provide a better understanding of phenomena and behavior in nuclear power plants. This enhanced capability for understanding and prediction leads not only to a quantification of the margin of safety; but also to ways of improving the availability, productivity and, hence, the economics of nuclear plants. The activities of the program, which span many scientific disciplines, are integrated into program and subprogram areas: loss-of-coolant accidents and emergency core-cooling systems; light-water-reactor (LWR) system behavior; structural integrity; probabilistic analysis and application; reactor performances; steam generator technology; liquid-metal fast breeder reactor; and advanced systems. Major recent accomplishments and current emphases are presented.

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EPRI NUCLEAR SAFETY AND ANALYSIS RESEARCH PROGRAM

INTRODUCTION

Since its inception in 1973, the Electric Power Research Institute's (EPRI's) Nuclear Research Program has placed major emphasis on the assurance of safety and reliability of nuclear power plants. For these principal goals to be achieved, work has been performed across many fronts: improving the knowledge of basic phenomena and the understanding of system behavior under a range of normal and off-normal conditions; understanding and describing the characteristics and behavior of fuels and structural materials used in pressure vessels, piping, large nuclear plant components and structures; improving the performance and utilization of nuclear fuels; improving the engineering and operational characteristics of nuclear plants; contributing toward the development of alternative reactor systems and fuels; and providing technical assistance toward the resolution of the problems of the external fuel cycle.

The EPRI program activities are coordinated and sometimes jointly sponsored with relevant federal agencies to ensure minimum overlapping and maximum benefit. Contacts and cooperation exist with foreign counterparts of EPRI through a series of formal agreements. Segments of EPRI programs are implemented by reactor vendors, architect-engineers, national laboratories, universities, and consulting and service organizations. The setting of priorities and the process of selecting projects are strongly influenced by the sponsoring utility industry, in periodic meetings, through an advisory committee structure that parallels that of EPRI's organization structure.

Within the total Nuclear Research Program, the Safety and Analysis Program is specifically responsible for addressing more closely identified safety issues and concerns and for providing the analytical tools for an adequate understanding and prediction of the behavior of the system and its components.

Many other activities of significance or implication to safety are being undertaken within the Nuclear Research Program at EPRI. These broadly include basic materials behavior (e.g., stress-corrosion cracking, crack initiation, propagation and arrest, nondestructive testing, fuel behavior, external fuel cycle, etc.) and research in engineering and operational aspects of the plant (e.g., component performance, fire-retardant materials, reduction of occupational radiation doses, human factors, and plant systems diagnostics). A high degree of integration and coordination exists among the various activities, especially those--such as the power shape monitoring system--that are highly interdisciplinary. Following the accident at Three Mile Island, Unit 2 (TMI-2), the Nuclear Safety Analysis Center was formed at EPRI to assist in attaining a higher degree of safety at nuclear plants through a detailed analysis of the TMI-2 incident and other lesser but significant events at nuclear power plants, through the derivation of the necessary lessons; and, in general, through a varied program of operational safety studies oriented toward application and procedures rather than research and development.

The achievement of a real and perceived level of nuclear plant safety is a varied, complex, and synergistic task requiring continuous and painstaking attention to technical and management detail. Research and development is thus not the sole ingredient in achieving the objective. Often the results of EPRI R&D efforts do not directly alter the actual safety of the operating system. Instead, they permit a better understanding of the system, thus leading to greater assurance of safety. They can also provide the basis for hardware and operational modifications that enhance the design and operation of the integrated system.

The greater assurance of safety may have immediate impact as either facilitating regulatory compliance or increasing operational flexibility with potential for higher productivity. Although increased regulation does not necessarily lead to increased safety, the Safety and Analysis Program has sought to generate experimental information and to produce calculational methods to assist the electric power utilities in their formal safety appraisals. More important, the EPRI program has been aimed at real safety in a logically consistent context. This goal implies the ultimate establishment of quantitative safety criteria consistent with minimizing the risk to the public and reducing occupational radiation exposure, in proper balance with the cost of producing electric power, which is of considerable interest to the consumer. Results of safety research have, on several occasions, confirmed the safety margins in current designs vis-a-vis regulation and have identified overly conservative controls (as in the case of fission-product decay heat).

In investigating nuclear safety questions, considerable hypothesizing is inevitable because of the large number of potential accident sequences and the practical impossibility of exploring all of them experimentally. Given a finite amount of resources, safety may best be served when these resources are invested in research roughly proportional to the risk, which is nominally the product of the consequences times the probability of occurrence. Single-minded concentration on the high-consequence and highly improbable accidents may lead to overlooking the low-consequence and more probable ones. In this connection, the Three Mile Island experience is very instructive indeed. It indicates that the considerable accumulated operating experience, along with probabilistic evaluations of large nuclear plants, needs to be considered seriously in realistically assessing safety and in planning safety R&D. Thus, instead of pointing to large-break, loss-of-coolant accidents (LOCAs), operating experience highlights the importance of (1) slow transients and small-scale leaks; (2) assurance of adequate inventory and supply of coolant; (3) mass and energy balances, natural circulation, and heat sinks; and (4) balance-of-plant interactions, including the pressurized-water-reactor (PWR) secondary system.

The EPRI philosophy on safety has recognized that prevention and early detection of incipient accidents are particularly cost-effective and risk-reducing investments as opposed to a similar or larger investment in accident containment and mitigation.

R&D STRATEGY

To achieve its goals, the EPRI Safety and Analysis Program has proceeded with a spectrum of experimental and analytical efforts striving to (1) reach a fundamental but operational understanding of the basic physical phenomena that occur in nuclear power plants, and (2) produce reliable analytical tools with which to predict the behavior of a nuclear power plant system under a variety of normal and off-normal conditions.

As the work of the Safety and Analysis Program progresses toward its objectives, efforts are aimed at answering one or more of the following questions:

- What is the assurance that a minor or a major accident will not occur in a nuclear power plant?
- If a minor or major accident occurs in a nuclear plant, what are the realistically expected consequences to the public and to the plant?

- Can it be demonstrated that the probability of such occurrences has been reduced to the lowest point practicable with current technology?
- What kind of experimental facilities and analytical tools should be used to further quantify the safety margins of current and advanced reactor systems?
- What can be done in terms of R&D to highlight areas of operational safety concern, how can they be quantified, and what are the technical means necessary to improve upon them?
- How can the availability and plant factors of nuclear plants be improved?

The EPRI Safety and Analysis Program is continuously sensitive to the evolution of both real and perceived safety concerns. The various steps used in R&D strategy for implementing the safety and analysis goals are described in the Appendix.

PROGRAM STRUCTURE

The Nuclear Power Program at EPRI is divided into four major areas, each of which is further divided into subprograms (see Figure 1). The safety and analysis activities are involved, to some extent, in all these areas, with much of the effort within the Water Reactor System Technology Program. The breakdown of this program into subprograms and their specific objectives are given, as an example, in Figure 2. The safety and analysis activities focus on the first four subprograms listed. The safety and analysis work in each of the other three programs is indicated below. Further breakdown of the subprograms into "approaches" and "technologies" is shown for the LWR System Behavior Subprogram in Figure 3.

PROGRAM HIGHLIGHTS

The following are highlights of the Safety and Analysis Program and some of the recent and significant accomplishments.

Loss-of-Coolant Accidents and Emergency Core-Cooling Systems (LOCA-ECCS)

Two-phase flow, heat transfer, hydraulic phenomena, and transient fuel behavior studies aim at improving understanding of separate effects and at providing guidance for follow-up investigations of increasingly complex combined phenomena and of system behavior under LOCA conditions. EPRI's approach is to critically appraise current models and correlations through a combined epxerimental and theoretical program and then to proceed with new and improved analyses. The

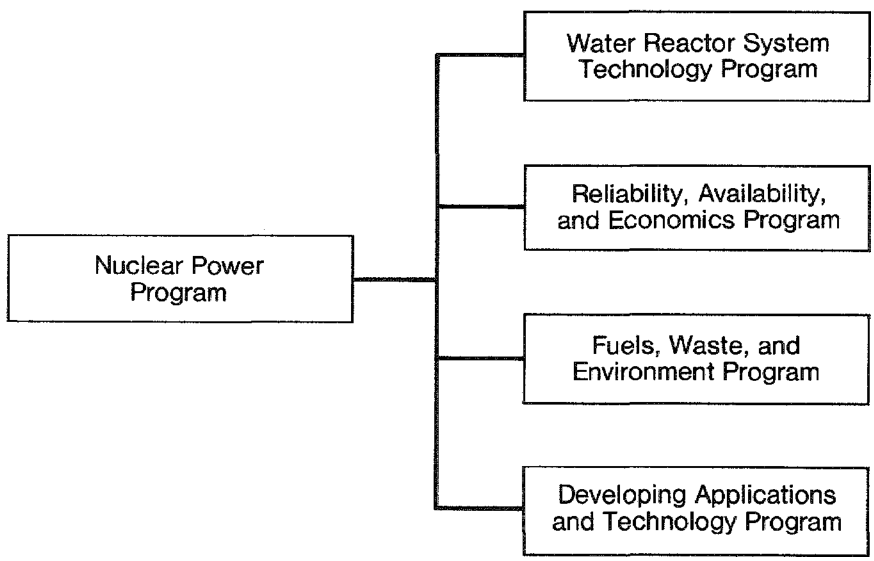


Figure 1. EPRI's Nuclear Power Program.

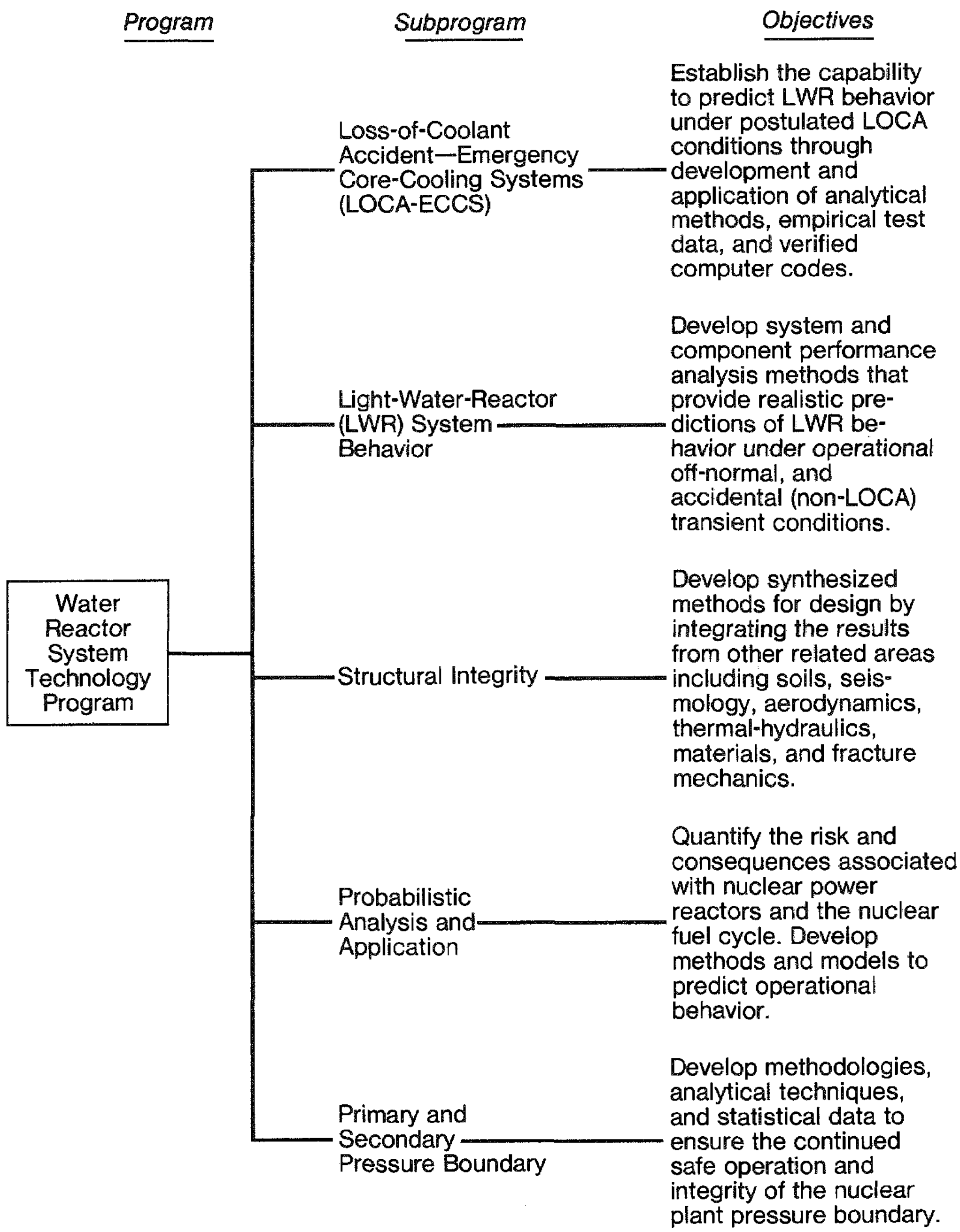


Figure 2. Water Reactor System Technology Program.

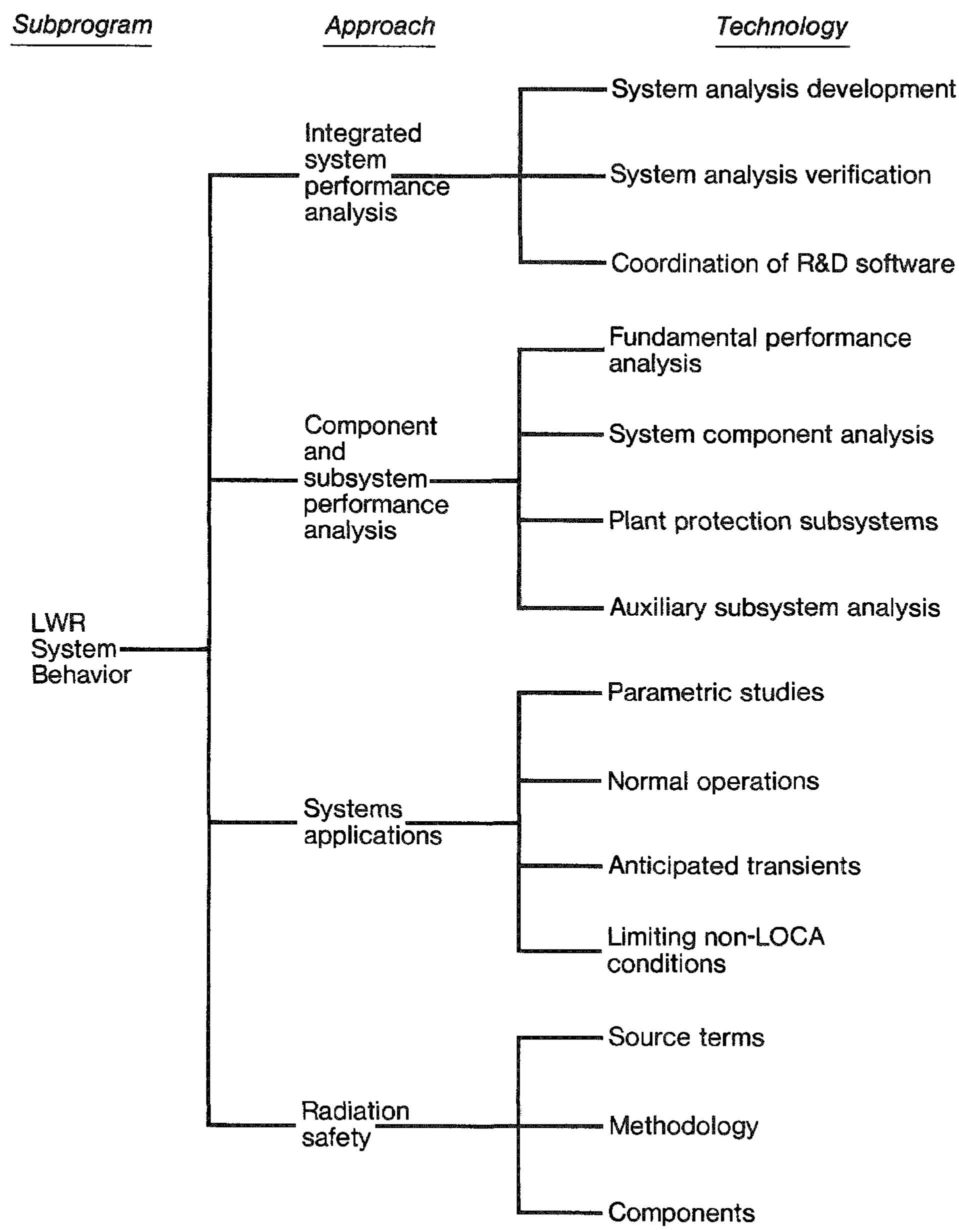


Figure 3. Breakdown of LWR System Behavior Subprogram.

development of basic physical models from single-rod and large-bundle data is stressed. These models are intended both for incorporation into and assessment of large system codes (such as RETRAN, TRAC, and RELAP). Through coordinating a wide range of contracts and in-house research, important advances have been achieved in the past two years at EPRI:

- Demonstration, in bundle geometry, of the more rapid quenching of Zircaloy cladding compared to the stainless steel used in experiments (1). Further analyses show how both the subcooled boiling heat transfer and the mass carry-over rate function are directly coupled to the quenching velocity (2,3).

- Recognition of the importance of the condensation and two-phase flow phenomena in pressurized-water-reactor (PWR) and boiling-water-reactor (BWR) upper plena. Several studies were completed on combined injection (4), entrainment, and two-phase turbulent condensation (5), showing how core behavior is coupled to the plenum phenomena. The large-scale Nuclear Regulatory Commission/Electric Power Research Institute/General Electric Company (NRC/EPRI/GE) refill-reflood experiments were initiated.

- The influence of flow blockage on heat transfer was studied experimentally, by four-rod experiments, and theoretically. Results show early quenching of deformed (ballooned) zones following reflood and the coupling of the deformation response to the heat transfer (6). As a result of these endeavors, changes were made to the original NRC/EPRI/Westinghouse FLECHT (full-length emergency-cooling heat transfer) effort to examine blockage in a 21-rod bundle (7-10). This work, particularly blockage studies in the Japanese Slab Core Test Facility, has been incorporated into the international 2D-3D program.

- The importance of natural circulation and reflux boiling for operational transients and small breaks was recognized following the TMI accident. New work, with a model of the TMI-2 core, showed that natural circulation was effective for a wide range of core resistances, primary water inventory, and secondary flow rate, and that reflux boiling was stable even with noncondensable gases present.

- An analysis of the core uncovery flow and phase separation was completed, and new experiments were conducted in tubes and rod bundles. The importance of modeling the two-phase level was demonstrated (11).

- Countercurrent flow was examined in detail (12-14). Not only were the effects of inlet geometry and liquid physical properties isolated and evaluated (15), but also a new, first-principles analysis was developed on the basis of wave stability (16). This theory did not invoke any empirical flooding correlation, as in earlier models.

- The modeling of two-phase flow has been the object of considerable effort (17). In trying to obtain a better critical-flow model, a comparison of the experimental, two-phase, critical-flow data is being made. This comparison will determine the possible need for further modeling work or for improving existing models. Another current EPRI project focuses on the dynamic prediction of transitions in the flow regime and associated changes in the exchange coefficients between the vapor and the liquid phase. The transition from the bubbly to the slug flow regime in two-phase flow in a pipe has been successfully predicted through this formulation.

- Chugging and condensation oscillations, which can occur in the BWR wet well, have been studied in simple geometric configuration. Extensive experimental data were gathered from pressure-suppression pools of different scale models. The analytical models, representing condensation at the bubble-liquid interface, downcomer internal chugging, pool dynamics, and dry-well gas dynamics were developed (18-21). The SAMPAC code was developed to describe local motion of the interface during bubble growth and collapse. Good agreement between calculated and measured values was obtained.

- The behavior of PWR primary pumps under postulated accident conditions has been investigated extensively both with experimental scale models and by analytical modeling (22-24). More recently, pump tests with a 1/5-scale model have shown that pump degradation is less severe than previously indicated by "semiscale" results (25).

- In the area of transient fuel response, heavy reliance has been placed on NRC-sponsored test programs as input to EPRI's analytical activities. These programs include:

  --A DATATRAN framework for including transient fuel behavior experiments within a transient fuel code qualification and assessment data base. Some initial tests (primarily from the Power Burst Facility [PBF] at the Idaho National Engineering Laboratory) have been incorporated.

  --Development of the transient fuel behavior code, FREY (fuel-rod evaluation system). This code is a combination of COYOTE, CREEP-PLAS, MATPRO, and the heat-transfer package from FRAP-T/RELAP/RETRAN. Completion of the first version was expected in the fall of 1980.

  --An exploratory study to define an approach to a constitutive model for Zircaloy evaluated the endochronic theory of plasticity as a means to provide an equation of state for Zircaloy under transient conditions (26).

  --An evaluation of the PBF test series, as planned in the NRC "experimental requirements" document, was performed (27) against the EPRI "design-basis fuel behavior analysis requirements" document (28).

LWR System Behavior

In this subprogram emphasis is placed on developing comprehensive analytical models and computational tools to describe system behavior. These tools provide a quantified understanding of light-water-reactor (LWR) system performance under normal conditions and during anticipated transients, as well as during unanticipated transients that do not lead to a loss-of-coolant accident. The activities in this subprogram include system modeling and computer code development.

- A major contribution during the past few years has been the development and continued testing of the MEKIN code (29), which couples a three-dimensional reactor kinetics model with thermal-hydraulic phenomena and feedback. This elaborate and highly detailed code can serve as a benchmark for faster, more approximate methods for transient behavior.

- The RETRAN code (30,31), developed as an improved, reliable, systems-thermal-hydraulics tool for analyzing LWR system transients, was completed and released in December 1978. The code was primarily developed for use by utilities to (1) evaluate and improve design and operation, (2) evaluate safety considerations, and (3) support licensing submittals. It was also developed for use by EPRI and its contractors to (1) interpret safety- and operations-related experiments and analysis and (2) make generic evaluation of safety issues, proposed regulations, and new concepts.

Approximately 25 utilities are now using RETRAN (in both CDC and IBM versions), and plants manufactured by the four major vendors have been analyzed and documented. Note that the primary emphasis in RETRAN is on modeling important non-LOCA and operational transients. The main features of the current version of RETRAN are (1) one-dimensional, homogeneous, thermal-hydraulic models for the reactor cooling system, (2) a point neutron kinetics model for the reactor core, and (3) a simulation model of the control system. This level of sophistication permits analysis of most incidents specified in the Safety Analysis Report, Chapter 15 (except the reflood portion of a LOCA), for both PWRs and BWRs. Efforts in space-dependent kinetics, incorporation of dynamic and algebraic slip, and provision for core uncovery and reflood are under way.

A model was developed for RETRAN analysis of the three turbine triptests performed at the Peach Bottom Power Station, Unit 2, during April 1977 (32) The results of the analysis of all three tests compare favorably with the actual measurements. Measured and calculated steam dome pressure transients are shown in Figure 4.

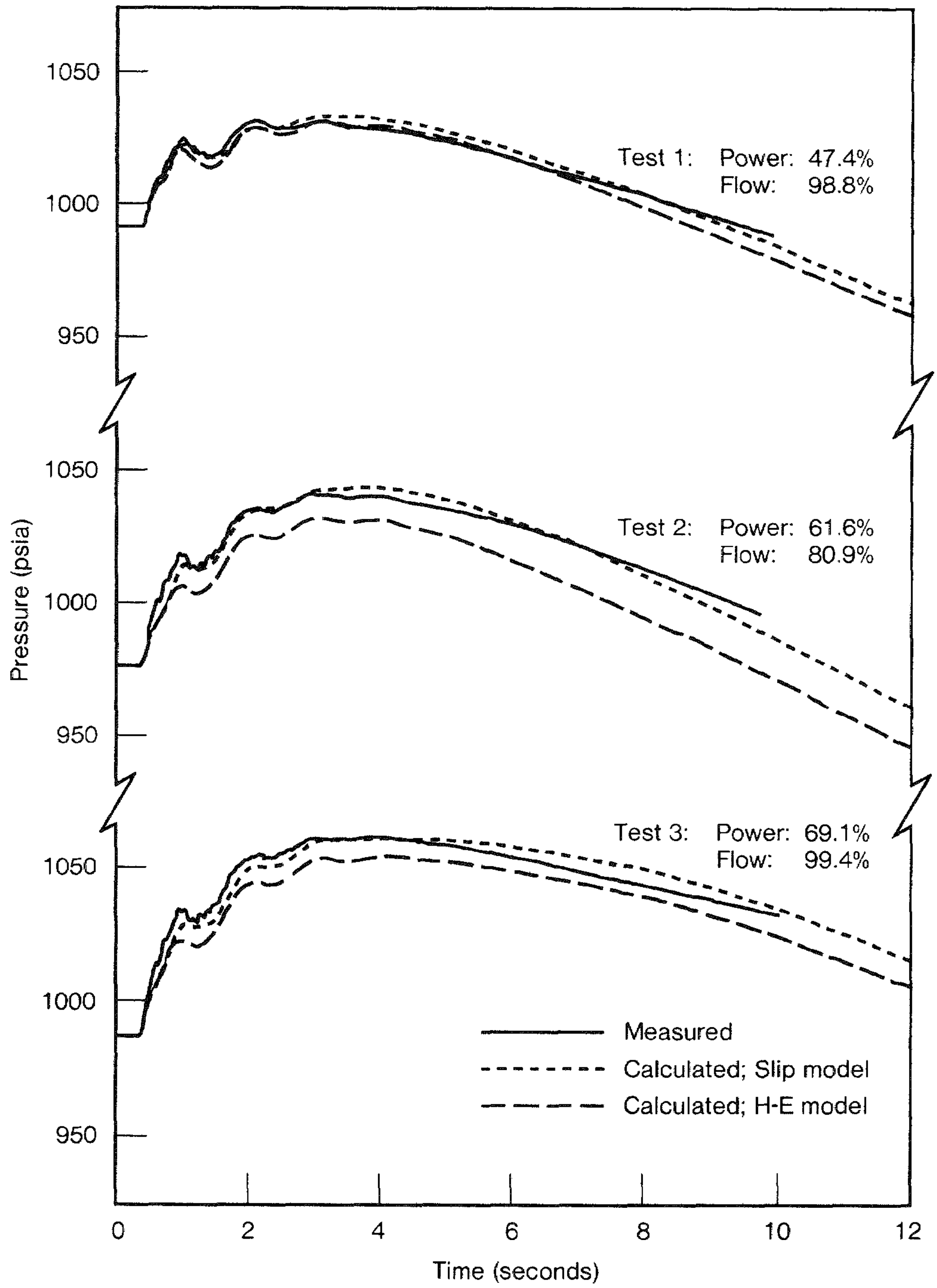


Figure 4. **Steam dome pressure transients during turbine trip tests at Peach Bottom 2.**

Extensive sensitivity studies were also performed to explore uncertainties both in input parameters and modeling options. This work has been a significant part of the RETRAN code qualification effort (33-35), has helped to identify areas of needed code improvement (36,37), and will be an important aid in the proper use of the code.

RETRAN has also proved extremely valuable to electric utilities since the recent accident at Three Mile Island. General Public Utilities, a member of the RETRAN Working Group, had previously modeled a TMI-1 test and a TMI-2 cooldown incident with good results as part of the RETRAN verification activity. Their models were modified to simulate the TMI accident and successfully predicted the early thermal-hydraulic response (38). The RETRAN calculations are compared with reactimeter measurements from the TMI-2 accident in Figure 5.

Further validation of RETRAN (as well as of MEKIN) is expected with EPRI-sponsored tests at the Arkansas Nuclear One, Unit 2 PWR. The tests include (1) four-pump coast-down at 80% power, (2) full-length and partial-length rod drops at 50% power, and (3) complete loss of load at 100% power. More transient tests with PWRs and BWRs are under study. RETRAN is also being validated through its use in the standard problem activities and the analysis of the results of the test series at the Loss-of-Fluid Test Facility (LOFT).

- In the light of recent experience, concern about the performance of overpressure protection systems for PWRs was heightened. EPRI has instituted a program to test relief and safety valves of different types, taking full advantage of domestic and foreign experience and of ongoing programs in this area. Test results are expected to contribute toward resolving the formal questions raised.

Structural Integrity

The Structural Integrity Subprogram integrates the results from aerodynamics, thermal hydraulics, fracture mechanics, soil mechanics, and structural mechanics to provide a realistic approach to defining both loads and structural response in a nuclear power plant. This subprogram has taken advantage of the technology transfer developed in defense programs and has modified it to meet current needs. A major continuing effort is the establishment of a relevant experimental data base both to verify design techniques and to develop simplified design methods. The principal activities and accomplishments are briefly described below.

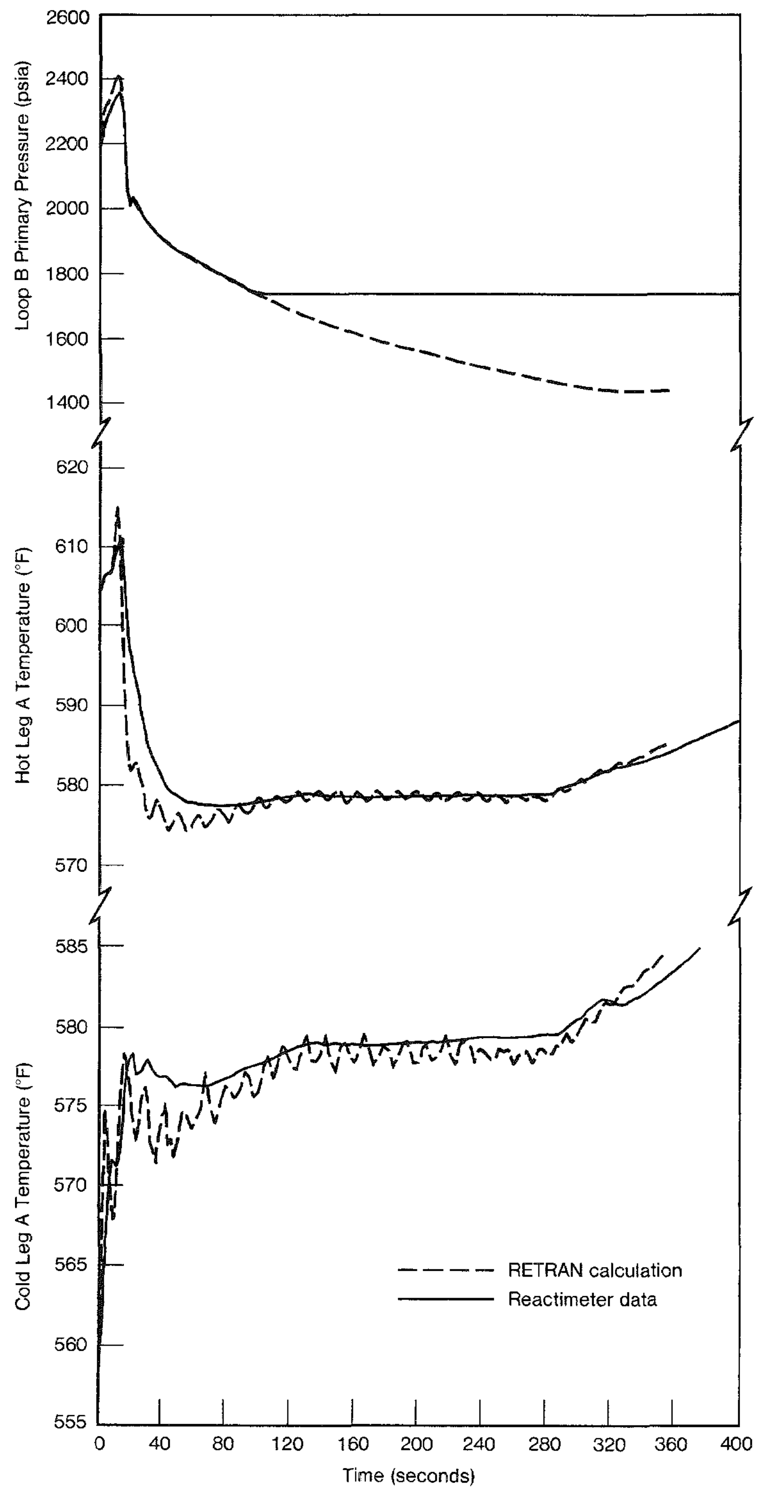


Figure 5. Loss of feedwater without auxiliary feedwater and with a stuck-open pilot-operated relief valve at TMI-2.

- Using an interdisciplinary approach, missile impact research provides data and analyses to quantify conservatisms in the current design of nuclear plants for missile impact effects. Full-scale tests in which wooden poles and steel pipes attached to a rocket sled were propelled against reinforced concrete walls were completed in 1977. These tests demonstrated the adequacy of balance-of-plant concrete walls to resist design-basis, tornado-missile impacts (39,40). Separate trajectory studies provide a more realistic basis for missile injection and impact conditions. The turbine-missile program is an outgrowth of this project. Two casing exit tests in 1978 provided full-scale benchmark data on the energy absorbed by the turbine casing in slowing down or stopping fragments from a failed turbine (41), as shown in Figure 6. Analytical and experimental efforts in this area are continuing. Full-scale tests of turbine missiles impacting reinforced concrete structures representing containment building walls will be completed in 1981. Scale model tests and scaling analyses will be conducted to ascertain how well extrapolation over a broad range of design parameters can be confidently accomplished.

- A major effort in the development of advanced numerical methods culminated in 1978 with the completion of the STEALTH 1D and 2D computer codes (42). The code provides a user-oriented, non-linear, transient stress analysis capability for the solution of safety-related problems in reactor engineering. The architecture of STEALTH was designed for easy adaptation to special applications, as indicated below.

- In the area of nonlinear soil-structure interaction, the objectives include controlled experiments for a three-dimensional data base and the provision of optional tools for the analyst (STEALTH-SEISMIC) (43-47).

- For piping system analysis, the ABAQUS-PIPEWHIP code was developed to provide a state-of-the-art, user-oriented, efficient, finite-element analysis code for pipe whip problems (48). The development of the ABAQUS-SEISMIC piping code is also in progress to provide a tool to study large displacement, high damping, and the effects of system restraints.

  The Indian Point 1 piping tests are designed to provide a data base for evaluating the piping response with respect to damping, insulation, and modern seismic support design. These data are expected to provide a basis for more realistic damping values and improved design guidelines.

- In the fluid-structure interaction area, the SOLA series codes are being coupled to the WHAMS code for application to the analysis of BWR containment transients. Also, the STEALTH-WHAMS coupling is being completed, and work is proceeding on application to PWR asymmetric hydraulic loads. Current work is on two-phase, three-dimensional modeling. Single-phase, three-dimensional fluid calculations are in progress, and initial couplings of 3D WHAMS to 3D STEALTH is completed and is being debugged. A near-term goal is to attempt three-dimensional calculations of the German full-scale blowdown tests in the HDR (Heiss-Dampf Reaktor).

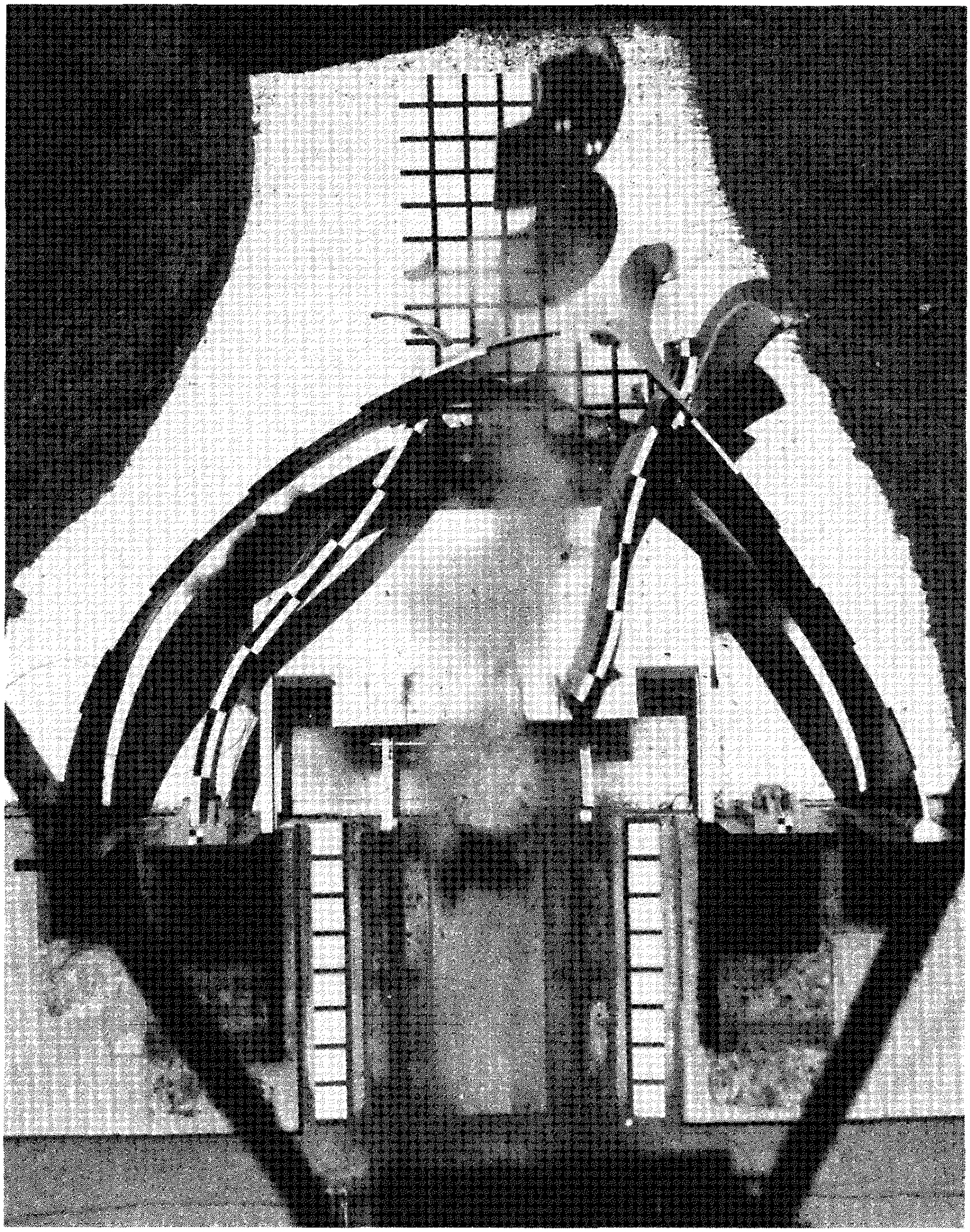


Figure 6. Turbine missile test. Grated background helps determine missile velocity after it breaks through cylindrical steel casing, permitting an estimation of the energy absorbed.

Probabilistic Analysis and Application

The tools of probabilistic methodology are undergoing continuing development and application to permit statistical interpretation of reactor safety assurance. By definition, probabilistic analyses do not have the potential for mechanistic conclusions. Variations in using and processing existing data and incorporating them in an analytical framework generate much discourse and, frequently, mildly conflicting conclusions. The latter are further diffused because the standards for modeling of phenomena and systems do not always conform to "common knowledge." Nevertheless, much progress has been made in recent years, and various predictive capabilities show a trend for convergence. Some recent highlights include:

- A Monte Carlo methodology was used to develop the TORMIS computer code to simulate risk from tornado-missile impacts on power plant structures (49,50). Missile generation, transport, and impact have been simulated, with uncertainties treated in a probabilistic framework. Sequential event formalism permits the treatment of both single- and multiple-missile generation. Two cases studies--a single unit plant using a set of missiles as specified by the NRC and a two-unit plant with an expanded missile set to test sensitivity--were treated. Preliminary results indicate that the risk of tornado-missile impact is very low, and the probability of damage to walls 30.5 cm thick is, at worst, about 10^{-7} per reactor-year. Although the validity of such low numbers needs further verification, they suggest that the usual, formal worst-case analysis may not have a sound technical risk basis.

- Efforts have been initiated to establish a technical basis for the use of value-impact (or cost-benefit) analysis by the nuclear power industry. A first attempt at assessing the state of the art (51) suggests numerous disagreements about the conclusions of the analysis that stem from varying assumptions over the correct method of calculating costs and establishing benefits. The establishment of a widely accepted method for applying value-impact analysis in the nuclear industry may require collegial action in order to consolidate the interests of diverse groups.

- This method was demonstrated by applying it to the issue of anticipated transients without scram (ATWS) (52,53). Using the principle that the optimum societal choice lies at the point where marginal cost equals marginal benefit, and assuming that all costs and all benefits have been properly accounted for, the analysis suggests that Alternative 3, as proposed by the NRC staff for ATWS resolution, should be more attractive to society and industry than Alternative 4. The four alternatives proposed by the NRC staff are described in Anticipated Transients Without Scram for Light Water Reactors, Report NUREG-0460, Vol. 3, December 1978. Briefly, they are: (1) no plant modifications, (2) modifications to reduce susceptibility to common mode failure, (3) modifications to reduce susceptibility to common mode failures and to mitigate most ATWS events, and (4) modifications to mitigate ATWS events. Further research in methodology development is needed to provide a sounder basis for routine value-impact application to practical problems in the nuclear industry and its affiliated community (e.g., global problems, retrofit or add-on decisions, regulatory guidelines, and priorities for safety R&D projects).

- Probabilistic methods have been applied to the risks posed by the external nuclear fuel cycle. The Status Report on the EPRI Fuel Cycle Accident Risk Assessment (54) indicates that the total front- and back-end risk of the nuclear fuel cycle is approximately 1% of the risk estimated for the power plant operation itself. Waste disposal proves the least hazardous operation by many factors of 10. This study is continuing with details of various phases of the cycle.

- An EPRI Review of the Lewis Report (55) on the Reactor Safety Study (56) (WASH-1400) shows that the Report reached many of the same conclusions as those in the original EPRI critiques of WASH-1400 carried out in 1975 (57,58) However, further studies show that Lewis Report conclusions concerning "great underestimation" of the uncertainties associated with overall risk may have been overstated. Indeed, a quantification of risk mean values and the associated uncertainties suggests that the median risk in WASH-1400 may be somewhat low, whereas the stated upper bound is still conservative (59) for the specific system(s) and situations analyzed.

- In the area of earthquake frequency and magnitude, the validity of the data-screening technique has been demonstrated by correlation with the Geodesic Survey data tapes. The existence of an empirical, universal frequency-magnitude shape function has been demonstrated from correlation with the observed data. An analysis to determine the applicability of the method to small regions is under way and suggests that the method may be used to provide some insight on local data (within the limits of available historical data and their accuracy) for a region $\sim 2 \times 10^4$ km^2 that is well characterized. This implies that, if the usually low-level and intermediate-level data (within a region of any given size) are sufficient to provide a statistically valid normalization, then the universal shape may be used to predict the frequencies of large-magnitude rare events (60-63).

- An analysis of the SL-1 accident (64) was undertaken using the in-plant and ex-plant consequence codes developed for the Reactor Safety Study (56) (WASH-1400). It is the first in a series of studies to benchmark the codes against actual data. Analysis results show that the dose prediction was within a factor of 2 to 4 of the measured values and on the conservative side--a satisfactory agreement, given the limitations of the data and the analytical techniques. It is also worth noting that even without an engineered, airtight containment, the amount of radioactive material released to the environs was a small fraction of total inventory (e.g., 0.15% of the ^{131}I inventory).

Plant Monitoring and Control and Test Instrumentation

Accurate, reliable, and useful information must be available to a nuclear plant operator for the efficient and timely monitoring of plant status during operation. The awareness and urgency for such needs have been heightened by recent events.

- The disturbance analysis system (DAS) study, initiated last year and now reaching conclusion, has shown the potential for increasing nuclear power plant availability by assisting the operator with timely recognition and correction of a variety of disturbances (65,66). The analysis methodology uses cause-consequence trees to embody the plant-specific design parameters required to recognize the cause and possible consequences of a disturbance. The system has been implemented and demonstrated by a nuclear reactor vendor in conjunction with a training simulator.

  Because of the success of this effort and in light of the TMI-2 accident, EPRI has initiated a study to determine if a similar type of disturbance analysis and surveillance system (DASS) with a significantly enlarged set of functional goals might enhance not only the availability, but also the safety of the plant.

  Two efforts are being mounted, one by EPRI and one by the Department of Energy (DOE). Proceeding along parallel but somewhat different approaches, both efforts involve teams composed of participants from utilities, architect-engineering firms, nuclear steam-supply system vendors, and consulting firms. The first step is a short-term, detailed technical scoping and feasibility study to determine the types of engineering analyses, subsystem configurations, and possible pathways for implementation that will be needed to accomplish the goals of a comprehensive DASS. This effort will establish guidelines for plant-wide monitoring procedures aimed at equipment and process variables as well as plant functions, such as reactivity control, primary system inventory, core heat removal, availability and capacity of heat sinks, and others (67).

  If this first step is successful, a full-scale demonstration project is anticipated to establish a plant-wide DASS based potentially on several hundred to several thousand analyzed event sequences. The disturbances, chosen by event analysis, will be modeled by cause-consequence trees; this analysis will offer guidance to the operator in the suppression of disturbances.

- The program for the development of state-of-the-art two-phase flow instrumentation is continuing. Several approaches have been tested for future promise. The most successful approach at this time appears to focus on the continued exploitation of the "Auburn meter." This instrument, based on conductance and capacitance, appears to be successful over the full range of void fractions. Further developments leading to film-thickness meters, assembly void fraction meters, and rough cross-section void distributions are under way.

Evaluation and testing of a recirculation flow meter for use in steam generators based on ultrasonics have been initiated.

The application of axial tomography was motivated by the need to obtain detailed spatial void distributions. The "moving patient" syndrome makes it unlikely that "clean" distributions can be achieved. The void distribution is in a dynamic equilibrium, resulting in a greater smearing of data than is desirable.

The Coriolis force instrumentation was successful in measuring void fractions below 20% and above 80%, but not in the intermediate range. These end regions are generally difficult for other instruments, and routine usage would require two instruments to secure information on the complete range of interest (0% to 100%).

Laser interferometry using fiber optics is being developed to measure velocities and sizes of droplets and bubbles. The equipment is expected to be operational in 1980 and appears capable of ultimately measuring stable fringe patterns. The possibility of a traversing microprobe capable of producing local void fraction and phase velocities appears real.

Steam Generator Technology

This subprogram includes a variety of experimental and analytical efforts to address some generic technical issues (metallurgical, chemical, and thermal hydraulic) that have surfaced in steam generators in recent years. The goal of the subprogram is, on the one hand, to evaluate the effect of degraded steam generator performance on PWR performance and, on the other, to improve plant availability by preventing steam generator-caused plant outages.*

- The objective is to provide the analytical methodology backed with experimental data for analyzing and predicting the behavior of the steam generator, secondary-side fluid in steady-state and transient situations. The first version of the three-dimensional code, URSULA 2, which applies to a U-tube steam generator, has been completed and debugged. Initial comparisons with experimental data from a scaled Freon vapor generator appear satisfactory, but further validation is required. Modeling experiments are under way to benchmark the analysis (68,69).

  Nonequilibrium treatment has been added in 1980, and the code will be generalized to include a once-through steam generator

*This work is coordinated with that of the independently sponsored Steam Generator Project Office at EPRI.

treatment capability. A more complete set of convergence criteria will be implemented, and input-output modules will be upgraded for general release.

- The COBRA code has been modified for application to steam generator geometry to predict both the global and the local flow fields.

Reactor Performance

The objective of this subprogram is to develop the technical base and computational capabilities required to obtain nuclear reactor characterization and performance prediction and to assess emitted radiation.

- The qualification of nuclear reactor simulation codes requires gathering experimental data against which analytical methods can be benchmarked (70-79). The extensive work performed to collect data from operating reactors has focused on the verification of power distributions. This verification is accomplished by comparing gamma scan measurements with analytical results and with in-core instrumentation data. Following earlier first-cycle or second-cycle gamma scan measurements, data on Hatch, cycle 3 and Quad Cities, cycle 4 were obtained in 1979.

- EPRI's work in the nuclear data base area (80-86) focuses on cross-section testing and evaluation for thermal reactor applications. Version 5 of ENDF/B, the first version of the file suitable for applications in thermal reactor analysis, was released, and intial testing was performed during 1979. Following further testing, the need for a complete evaluation of the thermal file will be determined, and additional work will be planned accordingly.

- The development of improved models (87-90) is continuing, with work on the application of a nodal method for reflector treatment and a void formation model for BWRs.

- The Advanced Recycle Methodology Program (91-93) has focused in recent years on BWR application in which difficulties appear because of a high degree of inhomogeneity and a strong dependence on thermal hydraulics. Benchmarking activities are proceeding to ensure determination at an appropriate confidence level. In 1979 work was performed with Hatch, cycle 1 and Quad Cities, cycles 1 and 2, using the data indicated in the first bulleted item of this section.

- An integrated reload safety analysis capability is being evaluated for development. Such activities may command considerable attention in the future, with emphasis placed on analytical procedures and code verification. A report on PWR physics procedures is scheduled for issuance during 1980, and a BWR procedures package is being evaluated. The interlocking and interrelationships between various codes are shown in Figure 7. The relationship of probabilistic safety evaluation codes to mechanistic codes is under consideration.

Developing Applications and Technology

The primary purpose of this endeavor is to identify and develop new and significantly better concepts in nuclear power generation, either as new components or as new systems. Major emphasis is placed on the liquid-metal fast breeder reactor (LMFBR). The motivation behind this emphasis is the perceived need for breeder reactor plants to be developed in time to avoid prohibitive fuel costs potentially arising from uranium shortages, restrictions on coal use, or inadequate contributions from alternative energy sources. Within this broad scope, the activities of the Safety and Analysis Research Program have played an important role.

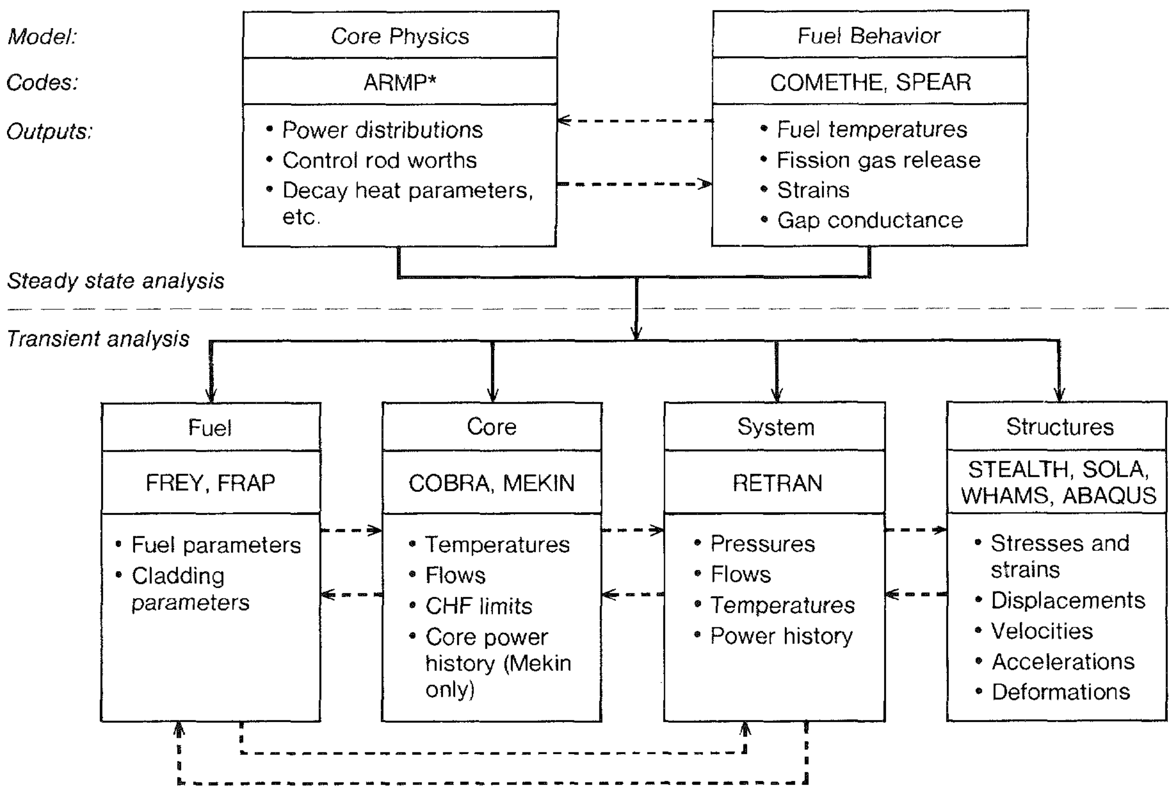
- Safety work on LMFBR designs has emphasized preventive measures. Work on a heterogeneous core design with a substantially reduced sodium void coefficient was initiated in 1975 within the framework of the Prototype Large Breeder Reactor design. More recently, the core design has been optimized (94). Work has been completed on the conceptual design of a self-actuated shutdown system that causes control-rod insertion with flow coast-down and primary pressure drop (95).

- A fast computer code, EPRI-CURL (96), for LMFBR transient analysis has been completed for a loop design and has been modified for application to pool-type LMFBR designs. The purpose of this development was to provide an efficient, fast-running analytical tool for the treatment of operational transients that do not include sodium boiling.

- A Fast Breeder Blanket Facility was constructed at Purdue University (97) to measure nuclear reaction rates through various simulated blanket compositions and arrangements. DOE participates in this project by sponsoring the operational costs of the facility.

- Evaluation work on alternative fuel cycles and designs has been pursued to investigate their feasibility and merits in extending existing uranium resources. The use of thorium and the merits of high conversion have been examined in such concepts as the high-temperature gas reactor, and LWBR-type core with heavy water as coolant and moderator (98,99). These studies suggest that there are no major new safety issues and that present and foreseeable uranium prices do not make the pursuit of alternative reactor designs or fuel cycles attractive. Serious disincentives include a multitude of institutional and market barriers and uncertainties in the nuclear industry, notably the currently unsettled question of reprocessing and recycling (100).

- The performance of combined top and bottom injection of emergency-core coolant (ECC) has been investigated (101,102), and some limited quantitative insight has been gained on this issue.



*Includes EPRI-CELL, CPM, PDQ-7, SIMULATE, NODE-P, -B, NORGE, etc.

Figure 7. Code interfaces.

RELATION TO OTHER PROGRAMS

Given the magnitude and importance of the safety task and the heavy involvement of material and human resources of many organizations in the United States and abroad, EPRI considers coordination among them a sine qua non. Based on the experience of the past several years, channels of communication and methods of cooperation must be continuously maintained and improved.

Cooperative programs on various levels and modalities have been undertaken with federal agencies, especially with NRC and DOE. In certain cases, the participation of industrial concerns forms tripartite arrangements (e.g., NRC/EPRI/Westinghouse and NRC/EPRI/GE) to conduct specific research projects. In other areas, work is coordinated to minimize duplication.

Formal coordination also exists with foreign counterparts that are involved in research and development efforts of interest to EPRI. The Central Electricity Generating Board of the United Kingdom, Electricite de France, the Central Research Institute of the Electric Power Industry of Japan, and the Bundesministerium für Forschung und Technologie of the Federal Republic of Germany are major participants in such formal interactions. Since some overseas efforts were initiated before similar U.S. programs, and since large facilities abroad might be used at incremental cost, cooperation and exchange with foreign entities have substantial potential for benefit. Some specific recent cooperative R&D ventures have included program activities in Sweden, Norway, the Federal Republic of Germany, France, and the United Kingdom.

UTILITY INPUT AND OVERVIEW

EPRI was established as the R&D arm of the utility industry in the United States. Thus, public power, rural electric cooperatives, and investor-owned electric utility companies provide continuing insight into near-term and long-range technical issues. Communication is effected through an advisory committee structure that parallels the EPRI management structure. The Advisory Committee is made up of representatives from the utility industry. The advice and overview of the Nuclear Safety and Analysis Task Force relates to most of the preceding subjects discussed. Periodic meetings with these groups serve a dual purpose: (1) They give utility members an opportunity to review, critique, and endorse the various EPRI-proposed projects, as well as to express their ideas for the research and development needed to solve pressing problems. This utility input is of paramount importance to EPRI staff in assigning priorities to the candidate programs and in allocating resources. (2) They serve as a convenient channel (in addition to the dissemination of formal reports) for the transfer of technology from the laboratory to the "real world," where the benefit from R&D can be fully realized. This latter purpose, one on which EPRI places prime emphasis, is also promoted by a series of technical workshops and seminars including, when possible, representation from the entire utility industry.

CLOSING REMARKS

Research in nuclear reactor safety is, for the most part, a long-term, expensive, and laborious undertaking because of the size and variety of test installations needed to simulate real conditions. The complexity of the real system, combined with the limitations in the test data base, places a premium on reliable analysis. The efforts of recent years have produced significant and reassuring results, in terms of experimental data and analysis, and have provided solid evidence of the conservative margins incorporated in nuclear plant design. Nevertheless, safety research programs must maintain a high degree of flexibility and a readiness to address changing needs and to provide swift technical responses to emerging safety questions. EPRI projects have had considerable success in this quick-response mode of operation.

The accident at TMI-2, even as it demonstrated the capacity of the reactor system to sustain substantial damage without undue risk to the public, has caused an appropriate reevaluation of approaches to safety R&D. The preoccupation with large LOCAs in the past is being replaced by an increased attention to the system analysis of small breaks, slow transients, decay heat removal, component reliability, and secondary side behavior. The probabilistic implications of such events are being scrutinized in some detail. Advances in many aspects of ergonomics research are being actively pursued. Even though these concerns and perceptions are not new to EPRI, and even though no dramatic shifts in program direction are envisaged, EPRI will strive, now as before, to respond to reality and to focus sharply on the goal of improved public safety and health.

APPENDIX: STEPS IN AN R&D STRATEGY

A given integrated set of investigations, aiming at a final product, proceeds by some or all the following steps:

1. Experimental and/or analytical work starting with first principles and leading to a fundamental understanding of physical phenomena and to the establishment of constants and empirical correlations.

2. The develoment of analytical and computational methodology to describe and predict aggregates or sequences of phenomena of increasing complexity. Code development applied to combined effects and systems behavior falls in this category.

3. Experiments performed with systems of stepped and increasing complexity, and under appropriate conditions to supply adequate data against which to try the analytical models.

4. Testing of the analytical models developed in step 2 against the experimental data obtained in step 3 leads to a validated, reliable analytical tool with technically defined uncertainties.

5. Maintenance and distribution of the validated analytical tools. This step often requires a special and sustained effort to effect efficient technology transfer from EPRI to the user organization (primarily the utility industry).

6. Finally, utilization of the above-mentioned tools in industrial applications and feasibility studies. This step frequently requires the active participation of EPRI.

These six items can be visualized as a progression of time steps, although there may be considerable overlap since each step can be and frequently is initiated before the previous step has been completed.

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