Current Perspectives in Nuclear Safety R&D

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FOREWORD

About eight years ago the U.S. Nuclear Regulatory Commission inaugurated an informal annual conference called "The Water Reactor Safety Research Information Meeting (WRSR)." This conference, conducted in the fall of each year, proved an effective and timely channel of communication for advances in the field of nuclear safety research.

EPRI was invited to participate in the Eighth WRSR Information Meeting in October 1980 and to present highlights of its program and accomplishments in a dedicated session. The following papers were presented:

"Current Perspectives in Nuclear Safety R&D," by W. B. Loewenstein and A. G. Adamantiades

"Disturbance Analysis & Related Developments," by R. Kanazawa

"Status of EPRI Turbine Missile Research Program," by G. Sliter

"BWR Intergranular Stress Corrosion Cracking Research Program," by R. Jones

"Analysis of Small-Break Tests," by R. Duffey

"Validating Risks Analysis: Two Aspects," by G. Lellouche

The first paper of the session summarizes the broad spectrum of research conducted by EPRI and is published here as a special report. The other, more specialized papers are also available upon request.

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ABSTRACT

General guidelines and major current themes of nuclear safety research at EPRI are presented. Such themes include the importance of analyzing small-break and other lesser accidents; natural circulation as a cooling mode; adequate prediction of plant transient behavior; analysis of degraded core; realistic estimates of radioactive releases; analytic and experimental assessment of structural integrity; and risk assessment as a useful tool for reactor design and operation. Recent advances and current efforts are summarized in the following categories: collection of data and their analysis in operating nuclear plants; scaled thermal-hydraulic tests; large-scale demonstrations; realistic assessment of accident effects and consequences; and safety quantification and the assistance provided to the operator in the control room of nuclear plants.

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Current Perspectives in Nuclear Safety R&D

Introduction

In planning and executing a national, goal-oriented nuclear safety R&D program, a number of principles must be observed (Figure 1). First, proper balance must be maintained between short-term and long-term goals. Present and immediate needs must not detract from the pursuit of long-term objectives. At the same time, steadiness of purpose must be tempered with necessary degrees of flexibility and adaptability to the changing requirements and problems in nuclear power development. This necessary flexibility not only assures the capability of prompt responses to emerging needs, but also allows for the recognition of past errors and for the redirection and reorganization of efforts in an environment of limited resources. A third important guideline is to recognize the need to maintain coordination and perspective among the various parts of a multifaceted and highly interdisciplinary work to ensure maximum benefit from the overall effort and to concentrate on the points of highest importance.

In the light of events over the past two years, a significant reevaluation of emphasis and direction has taken place. Many ideas and concepts now attracting attention have been propounded for quite some time, but recent events have raised our awareness of their significance. These ideas (Figure 2) include, among others, the importance of analyzing small break accidents, the potential of natural circulation as an adequate cooling mode, the prediction of plant transient behavior under a wide range of parameters, the analysis of certain degraded core and plant scenaria, the realistic assessment of radioactivity releases and transport under accident conditions, the need for analytic and experimental assessment of structural integrity, and the conversion of probabilistic risk assessment into a useful tool of reactor design and operation.

Considerable efforts are also expended at EPRI in the areas of materials and systems behavior, as well as in the operational and engineering aspects of nuclear plants. Efforts directly bearing on safety include the integrity of the pressure boundary, human factors, radiation control, etc. A companion paper in this

- Long-term objectives must be pursued undetracted by immediate concerns
- Steadiness of purpose must be blended with a degree of flexibility
- Coordination and perspective among the various parts of the effort must be maintained
- Maximum effort must be directed toward items of maximum importance

Figure 1. Nuclear Safety R&D General Guidelines

- Importance of analyzing small-break and other lesser accidents
- Natural circulation as a cooling mode
- Adequate prediction of plant transient behavior under a wide range of parameters
- Analysis of degraded core conditions
- Realistic assessment of radioactive releases
- Analytical and experimental assessment of structural integrity
- Conversion of probabilistic risk assessment into a useful tool for reactor design and operation

Figure 2. Major Themes of Nuclear Safety R&D

session presents the status of the intergranular stress corrosion cracking research program for BWRs.

The EPRI Nuclear Safety Research program, while striving for continuity and consistency, incorporates the foregoing concerns to provide an integrated picture of current perspectives in nuclear safety. Examples from the EPRI safety program are presented on the following pages to illustrate these basic themes and to outline advances made in the past year in several important areas.

Emphasis on Operational and Plant Data

The gradual accumulation of operational experience (1) has given new insights into the problems of nuclear plants and has influenced the impetus of research and development work. Consequently, the performance of tests and measurements at operating plants has been emphasized. These investigations provide: (1) an improved understanding of the behavior of the system as a whole made up of interacting parts, (2) a means of testing and qualifying the computer codes, and (3) a narrower range of the parameters that need analytical and experimental investigation.

Three examples will be mentioned here: First, the extensive gamma-scans and special TIP detector measurements, performed in both PWRs and BWRs for measuring power distributions, have provided two complete and accurate sets of benchmarks for qualifying core calculational methods (2). Figure 3 shows a cross section of the Hatch 1 core, the TIP locations, and the 106 bundles on which gamma-scans were performed, including a complete core octant and six additional four-bundle cells chosen to reveal any power asymmetries. The insert in the figure shows a comparison between measurement and calculation performed with the code SIMULATE, which is now ancillary to the Advanced Recycle Methodology Program (ARMP). The ARMP package, comprising a set of about 20 codes and continuously being expanded, has been extensively tested and qualified with the data from the gamma-scan measurements.

The SIMULATE code results are generally in good agreement with the gamma-scan measurements. In cases of deeply inserted control blades, however, the code tends to overpredict the power peak. Comparisons were also made between code predictions and axial power distributions derived from the process computer (either P-I or BUCLE) at selected intermediate points of the cycle. Since the process computer distributions have been shown to be quite accurate (3), these comparisons provide a good test of the predictive capabilities of the code SIMULATE. The

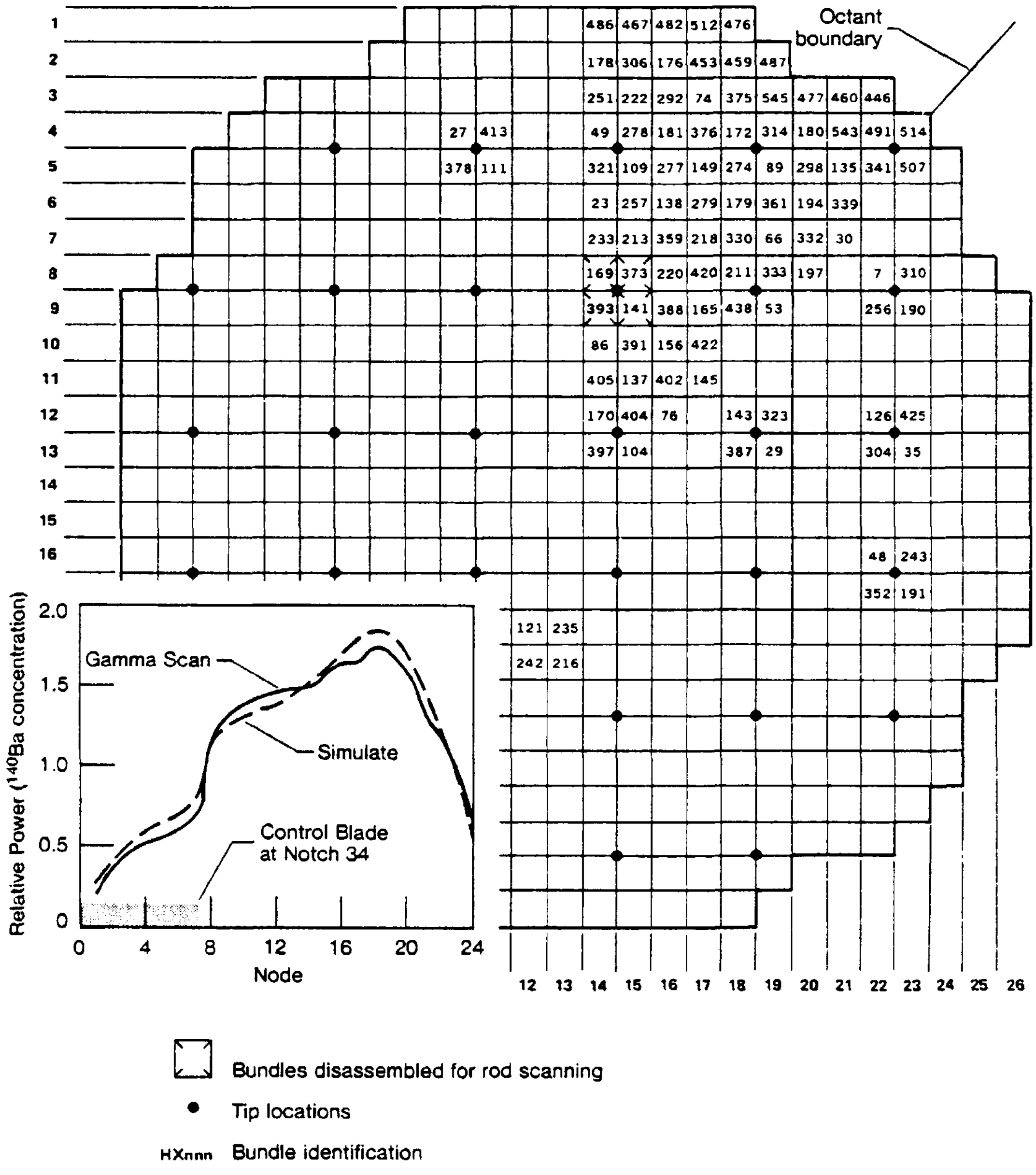


Figure 3. Octant Normalized Axial Power Distribution: Bundle 393 (Edwin I. Hatch Unit 1 BWR)

qualification process has helped define with confidence uncertainties in the calculations and has allowed possible reduction in the operating margins, thereby increasing the safety and the productivity of the plant.

A second example of plant test data is the transient and stability test performed at the Peach Bottom-2 (4). Three turbine trip tests, from varying power levels and coolant flows, were performed with extensive measurement of plant parameters taken with both regular and special instrumentation. Analysis of the plant transient response with the RETRAN code gave new and valuable insights into plant behavior and the sensitivity of important variables (such as vessel dome pressure) on a number of plant parameters and component behavior. The interaction between neutronic calculations and thermohydraulic analysis was highlighted.

These BWR turbine trips and their analysis provide a particularly stringent test of any system dynamic code because of strong coupling between pressure, core void, and reactivity; dominance of acoustic phenomena during the early part of the transient; and the significance of later transient stages when pressure begins to drop.

The Peach Bottom-2 plant investigations also included a series of stability tests performed by means of pressure perturbation at low-core flow rates (Figures 4 and 5). It was demonstrated that (a) this technique can provide data superior to those obtained with the rod oscillator technique, with minimal disturbance to plant operation; (b) there is considerable stability margin designed into a large BWR with decay ratios estimated between 0.12 and 0.35; (c) safe operation can be obtained at low-flow conditions above the rated power/flow rod line. This latter demonstration can yield increased operational flexibility and allow a reduction in the number of precondition ramps on the ascent to full power. It is also possible to realize considerable fuel cycle economic gain by allowing operation at full power but less than full flow, since this condition hardens the spectrum and causes higher plutonium production.

These stability tests, along with data from overseas reactors (Barsebäck 2, in Sweden and TVO-1, in Finland), will be used in the qualification of a BWR stability code currently being developed under EPRI sponsorship for utility use. This code, with several others, is intended for incorporation in the Reload Safety Analysis Methodology package being assembled by EPRI.

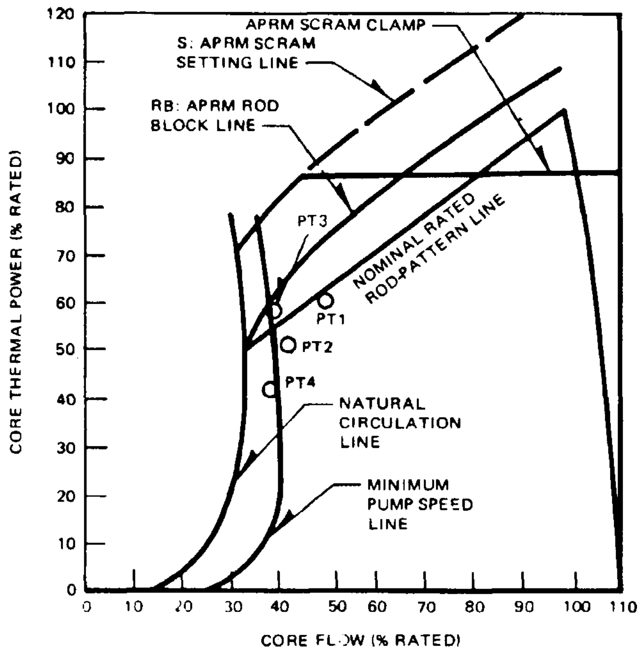


Figure 4. Peach Bottom-2 Low-Flow Stability Tests: Actual Test Conditions

Test Number	*Power (% rated)*	*Core Flow Rate (% rated)*	*Decay Ratio*
PT1	60.6	51.3	0.259
PT2	51.7	42.0	0.303
PT3	59.2	38.0	0.331
PT4	43.5	38.0	0.271

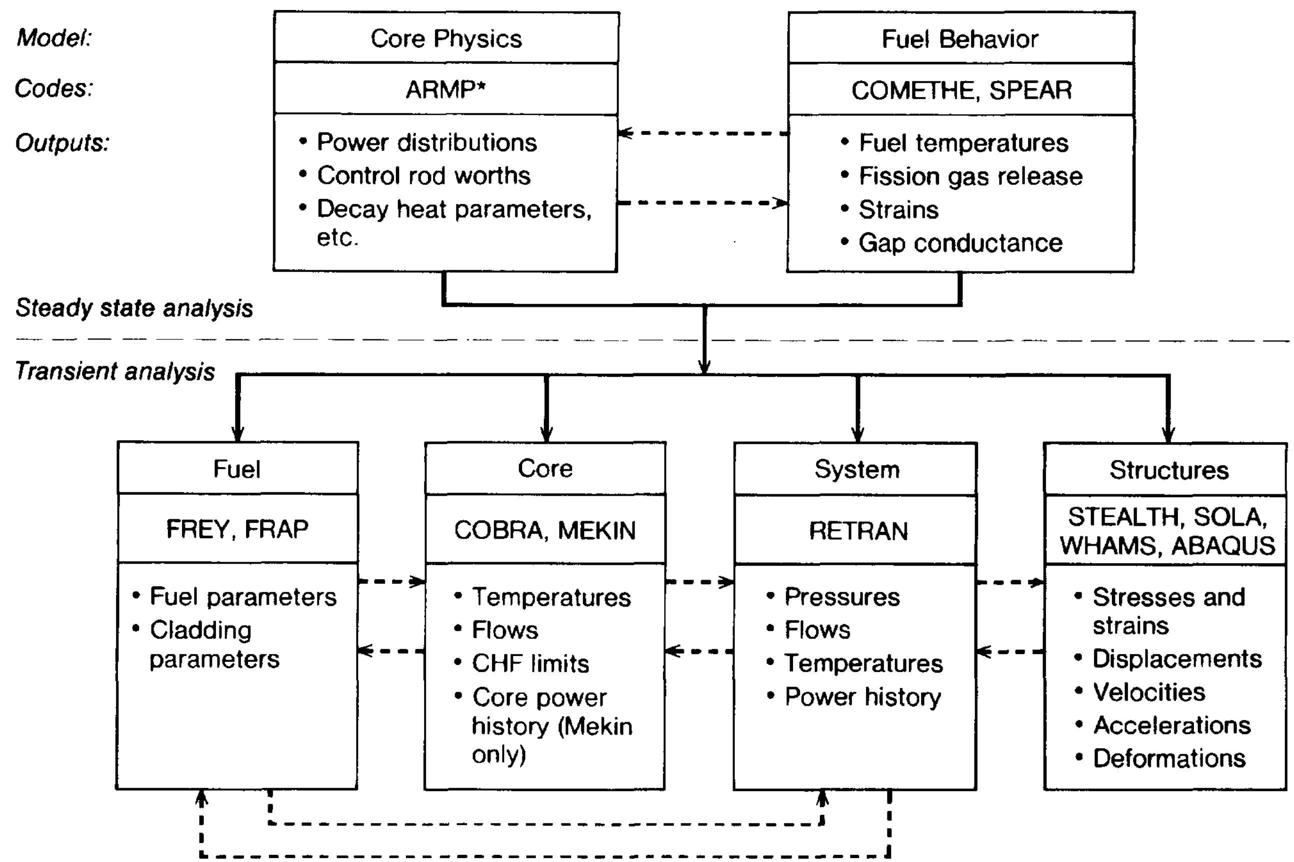
Figure 5. Low-Flow Stability Tests at Peach Bottom-2, EOC 2 Core Stability Margin Estimates

The extensive code verification and validation work derived from plant data will be assembled into an integrated and qualified package for use by the electric utility industry. A general view of the various codes or code groups and their interrelationships is shown in Figure 6. The activity aims at a calculational tool applicable to both conservative licensing procedures and best-estimate predictions, with minimum variation in computer code models. As shown in Figure 7, the purpose of this package, scheduled for completion in mid-1983, is to provide the utility industry with an independent, verified, and qualified calculational capability for both licensing and operational flexibility and optimization.

In the structural area, significant experience will be gained through the Dynamic Testing and Analysis Program on a piping system of the Indian Point-1 plant (5). The first phase of this program, already completed, consisted of dynamic testing of an as-built, lightly supported 8-in. diameter pipeline. Both snapback and forced harmonic vibration were used to excite the piping system. Subsequent phases will test and analyze the same pipeline without insulation, with modern seismic constraints (mechanical and hydraulic snubbers), and under high-level excitation (with strong nonlinear response). The test results will have two important functions: (a) to assess the accuracy of alternative analytic methods that might lead to improved, more realistic models of piping systems subjected to seismic, hydraulic, operating, and other dynamic loads; (b) to demonstrate that damping values permitted by current regulations entail significant conservatisms. It is also possible that new guidelines could be established for the development and use of simplified piping designs.

The Importance of Scaled Tests

Data from operating plants go hand-in-hand with results from properly scaled tests. These latter tests provide useful data where gaps exist in information from operating plants due to inadequate data collection systems and other factors. Also, they can extend the range of parameters to accident limits that operating plants are not allowed to approach. The following examples can be cited (Figure 8): A number of natural circulation tests have been and will be performed at operating LWR plants. Since a number of limitations in the performance of these tests exist, scale-model experiments are essential to supplement the operational data. A 4-loop natural circulation model has been constructed, under EPRI sponsorship at the Stanford Research Institute, to simulate the Trojan PWR 4-loop plant. The 2-loop reflux boiling facility simulates an actual 2-loop Combustion Engineering System 80 plant. The experimental results have been compared with



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

Figure 6. Code Interfaces

Applications

- Licensing activities
- Best estimate predictions of plant behavior in various steady state and transient situations

Purpose

- Increase utility independence from fuel suppliers
- Offer more flexibility in tailoring plant performance to utility system operating needs
- Improve utility understanding of plant behavior and performance

Figure 7. Development of Reload Safety Analysis Methodology and Code Package (RP1761)

- 4-loop natural circulation model; simulates the Trojan PWR plant
- 2-loop reflux boiling facility; simulates an actual 2-loop PWR plant
- Model of TMI-2 primary system to investigate effectiveness of natural circulation
- Comparison of core uncovery and phase separation model with single tube, rod bundle and TMI-2 data
- Level indicator experiments
- U-tube steam generator scale model with freon
- Coolability of a bed of debris, simulating a severely degraded core

Figure 8. Scaled Tests

analytic models. A model of the TMI-2 primary system has been used to show that natural circulation was effective for a range of core resistances, primary water inventories, and secondary flow rates (6). Experiments with noncondensible gases (helium and nitrogen) injected in the primary system demonstrated the stability of reflux boiling even with noncondensible gases present.

Thermal-hydraulic models in conjunction with scaled and full-size test data can supplement the existing guidelines and provide an improved basis for operational decisions. Such an analysis of core uncovering and phase separation was performed and compared with single tube, rod bundle, and TMI-2 accident available data (7).

The agreement between theory and experiments is encouraging. Experiments on level indicators were carried out (jointly with NRC/GE) with TLTA for small-break transients. New work on level indicators for PWRs is underway with utility support and involvement.

Transients on the secondary side, including loss-of-feedwater and steamline break, have been studied with a full-scale model of a U-tube steam generator. Finally, the coolability of a degraded reactor core, resulting from a severe reactor accident, was investigated through a scale model with spaced rubble beds.

In the light of recent experience primarily stemming from the events at TMI-2, the importance of a variety of properly designed and operated and of accurately interpreted scale-model experiments cannot be overemphasized.

Large-Scale Demonstrations

In several cases (where small-scale tests are insufficient because of uncertainties in scaling laws and procedures, or where plant tests are impractical) large- or full-scale tests under controlled conditions are necessary.

Extensive missile testing has been conducted over a number of years with the rocket sled facility at Sandia Laboratories. Since the tornado missile tests on which reports were presented last year, a series of impact tests with turbine missiles have been conducted. A companion paper in this session will elaborate on technical details of these tests. Suffice it to mention here the following two outcomes: (a) The turbine missile tests impinging under various conditions on turbine casings have demonstrated that a large amount of the missile energy (65 to 100%) is absorbed by the casing in the breakthrough process, thus considerably reducing the threat of the missile to the containment; (b) A series of tests featuring actual-size turbine fragments impinging on full-thickness, reinforced-concrete wall segments have shown the capacity of the latter to withstand the impact without loss of containment function, even under severe overspeed conditions. Empirical formulae presently used to predict scabbing and wall perforation were shown to be quite conservative.

Another example of large-scale effort is the intensive valve-testing program undertaken to resolve questions of primary system valve performance. Spring-loaded safety valves, pilot-actuated safety valves, and power-operated relief valves will be tested to demonstrate their capability to operate satisfactorily under steam inlet conditions as well as under a range of subcooled water and transition (steam-to-water) flows. The program responds to regulatory document NUREG 0578 Section 2.1.2 and includes, in addition to testing, an analytic activity and a technical support program. This is a dedicated, intensive R&D program, separately funded and heavily involving operating utilities. Due to the stringent schedule imposed by NUREG 0578, the tests have been divided among three facilities, namely Combustion Engineering, Wyle, and Marshall of Duke Power Company. Preliminary test results from the Marshall facility of relief valves under steam-flow are encouraging, showing that, for the most part, the valves have performed (opened and closed) as expected.

Realism in Accident Effects and Consequences

The main challenge posed to the R&D community by the potential of nuclear accidents is to narrow to realistic bands the potential range of accident effects and consequences, and to afford validated and reliable tools for risk assessment (Figure 9).

While the future course of degraded core studies is not totally clear, some combination of experimental and analytic effort will proceed. Given a set of assumed conditions, the magnitude of the effort can be easily circumscribed on a technical basis; however, nontechnical considerations will certainly have an effect. If technical answers to specific issues are required, the procurement of data from highly visible demonstration tests as benchmarks to the analytic efforts would be prudent, if not necessary. The question might be, "What test size?" This decision will definitely be affected by value/impact considerations and the availability of resources.

Small-scale tests to emphasize common elements of a core melting accident

- Coolability of deep debris beds
- Interaction of molten materials with water

Large-scale tests

- Hydrogen combustion and management
- Effect of turbulence, sprays and ignition methods

Analytical work

- Containment over-pressurization
- Containment failure
- Fission product release
- MARCH/CORRAL code evaluation and improvement

Figure 9. Realism in Accident Effects and Consequences

A number of degraded core studies are complete, underway, or tentatively planned to address phenomena ranging from flow blockage and fuel failure to severely degraded core configurations, hydrogen generation, combustion and management, and the potential for interaction with containment. Small-scale experiments for investigating basic phenomena can be combined with large-scale tests for integral effects and with modeling and code evaluation efforts. The small-scale experiments emphasize common elements of a core melting accident such as the coolability of deep debris beds and the interaction of molten materials with water in typical LWR geometries. Large-scale tests are now scheduled primarily in the hydrogen combustion and management area to study the effects of turbulence, sprays, and controlled ignition methods. The analytic work will concentrate on phenomena that may lead to containment overpressurization, failure, and fission product release. The code MARCH/CORRAL will be evaluated and improved.

A reevaluation of containment integrity and effectiveness can be made with the data and analyses from the degraded core studies. These investigations are expected to renew confidence that most current containment designs are conservative with respect to maximum conditions expected in an accident. They could also lead to improved design methods based on physical observations more than on highly conservative assumptions.

The overall risks accruing from potential nuclear reactor accidents can now be quantified through risk assessment methodologies incorporating a large number of assumptions and numerical parameters. Both the methodologies and data used in probabilistic analysis and consequence evaluation can stand considerable improvement, if they are to be used increasingly in reactor safety evaluations. Considerable effort has been and is being expended by EPRI to improve and to validate probabilistic risk assessment, as discussed in a companion paper in this session. One means to accomplish this validation is to apply available in-plant and ex-plant consequence models to actual situations such as the SL-1 accident and its radioactive releases (8).

The Goal of Quantifying and Improving Safety

The overall thrust of the various safety R&D activities aims at two main points: to accurately quantify existing margins of safety and to provide guidelines and means of improving plant safety and minimizing overall risk. Two closely related efforts at EPRI aim directly at these two goals (Figure 10): the Power Shape Monitoring System (PSMS), on which considerable effort has been expended over the

QUANTIFYING AND IMPROVING SAFETY

Power shape monitoring system (PSMS)

- Monitors and predicts core power distributions
- Predicts fuel performance
- Provides greater operating margins
- Provides more efficient load maneuvers

Disturbance analysis and surveillance system (DASS)

- Verifies, analyses, integrates and assigns priorities to plant information
- Assists operator to improve plant safety and availability

Figure 10. Quantifying and Improving Safety: PSMS and DASS

past few years, and the Disturbance Analysis and Surveillance System (DASS), on which more will be said in a later presentation.

Monitoring the status of the reactor core can lead to a refined knowledge of its thermal margins. An on-line core monitoring system (PSMS) installed at the Oyster Creek BWR plant has demonstrated the usefulness of the system particularly during feedwater transients. Plant monitor readings refined with core physics codes have resulted in point-heat ratings and thermal limits to an accuracy of a few percent.

Considerable progress has been made in the development of a DASS as a tool useful not only in increasing plant factor and availability, but also in ensuring plant safety. Considerable attention has been given in the past year to safety status monitoring and the analysis of "small-break" transients, a topic on which a companion paper is given in this session. Although this topic involves multiple faults in the plant, primary emphasis was given on the steam generator as a heat sink and the core as a heat source. The modeling of the LWR plants is proceeding by building individual simulation models for plant components and by attempting to describe the system to the operator by interconnected and interacting modules.

The intended result of this effort, mounted in collaboration with the Department of Energy, is to provide a rapid and user-adapted computational tool for plant operational transients. Initial PWR models are being developed and will be benchmarked against existing elaborate codes (e.g., RETRAN) and existing plant data. Improving and focusing the operator's understanding of plant status, particularly under transient conditions, will result in better visibility and controllability of plant operations, which imply improved safety in nuclear plants.

Another example of safety quantification and improvement is EPRI's involvement in quantifying the probabilities of an anticipated transient without scram (ATWS), a subject of long-standing concern. The EPRI program has contributed to quantifying the relative probabilities in an ATWS event and to understanding the relative merits of various proposed fixes. Efforts continue to establish a widely accepted value/impact methodology (Figure 11) for application in this and similar situations. By making informed judgments on technical merits and risks, the appropriate authorities can ensure that funds are expended with maximum efficacy and maximum benefit to the public.

Development of a widely accepted value/impact methodology to

- Assess the relative merits of various research and development strategies
- Assess the merits of alternative, proposed technical fixes to safety concerns (e.g. ATWS)
- Ensure funds are expended with maximum efficacy and benefit to the public

Figure 11. Quantifying and Improving Safety: Value/Impact Methodology

The Collaborative Character of Safety

A synergistic approach is necessary to achieve a desired level of safety in nuclear plants. Solid design, proper operating and maintenance procedures, a rigorous but rational and enlightened regulatory oversight, and sound management practices are all indispensable. Although technical R&D efforts can provide valuable inputs to all these factors, they do not always contribute directly to fulfilling formal safety requirements and goals. Consequently, other organizations have an important role to play in the process of transferring and applying the research results (Figure 12).

The Nuclear Safety Analysis Center (NSAC), also under EPRI but with a separate status and funding, was initiated in the aftermath of TMI-2. NSAC now continues its function mainly as a technical evaluation, communication, and liaison organization capable of a fast response to current problems and emergency situations. Its main topical areas are the collection and analysis of significant event reports; the preparation of "what-if" studies to determine the margins at TMI for increasingly severe additional failures; and the communication and coordination function.

Also supported by the electric power industry, the Institute for Nuclear Power Operations (INPO), on the other hand, is an independent organization. Its function is to set and to implement plant operational standards in the safety area and to encourage and promote high dedication to operational safety.

Also important to the overall success of the safety research effort is EPRI's cooperation with a number of foreign organizations. These groups primarily represent R&D arms of the utility industry abroad to exchange information and to cooperate in other efforts. Examples of such exchanges are CEGB of Great Britain, EDF of France, BMFT of the Federal Republic of Germany, CRIEPI of Japan, CISE of Italy, and others.

Summary and Conclusions

The EPRI safety R&D program has been guided by a set of principles to operate effectively within the constraints of limited resources and to respond to the needs of the electric power industry. Many concerns that surfaced in the post-TMI atmosphere had been topics of EPRI investigation before that event, including human factors research, disturbance analysis, small-break events, and integrated system response methodology under transient conditions.

Role of NSAC

Functions

- Technical evaluation
- Technical communication
- Liaison with industry groups
- Fast response to current problems and emergencies

Topical areas

- Significant event reports
- "What-if" studies
- Communication and coordination

Role of INPO

Functions

- Set standards of operational safety
- Conduct plant-by-plant evaluations and help implement standards
- Promote high dedication to operational safety

Figure 12. The Collaborative Character of Safety

Although we at EPRI perceive no need for dramatic shifts in our general R&D direction and emphasis, flexibility must be a continuing part of the program. A look into the future suggests the following trends:

a. A definition of R&D needs to support conclusion of the current debate on Class 9 accidents by either mechanistic or probabilistic approaches or a combination of both.

b. The increased use of data collected at operating plants.

c. The analysis and interpretation of selective large-scale demonstrations, notably in the area of seismic data collection and design methods.

d. An emphasis on faster, more efficient, and more reliable analytic methods and codes in the areas of thermohydraulics, fuels and materials behavior, structural integrity, and fluid/structure interaction.

e. An improved knowledge and predictive capability in fission product behavior, transport, and attenuation.

EPRI recognizes the requirements in all these areas and intends to structure its program to provide the needed technical data and analyses for problem resolution. The program will thus be responsive to both the formal and substantive requirements placed upon the electric power industry. The fruit of EPRI's research, as well as of other organizations in the field, in the final analysis will be judged by the degree to which it responds to real and perceived safety problems and enhances the safe and efficient operation of nucler power plants.

Notes and References

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