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Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program

A Report to the U.S. Nuclear Regulatory Commission

ADVISORY COMMITTEE ON REACTOR SAFEFGUARDS
with contributions from the
ADVISORY COMMITTEE ON NUCLEAR WASTE
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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FOREWORD

In 1998, the Advisory Committee on Reactor Safeguards (ACRS) submitted to the Nuclear Regulatory Commission (NRC) a comprehensive report of the NRC's Safety Research Program, NUREG-1635, Vol. 1, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," which documented the ACRS conclusions and recommendations. The ACRS continues to support the conclusions and recommendations of that report. The present report is more modest in scope and is intended to provide additional information concerning the research needed to support Commission programs, especially risk-informed regulation. Not all research programs are included in this report and our observations are often limited to certain aspects of a given program. This does not imply in any way that these programs are less important to the Commission's activities.

The primary reason the NRC carries out nuclear safety research is to provide the knowledge that constitutes the technical bases for regulatory decisions. When the Commission receives an application for licensing a new nuclear system or a major modification to an existing system, or is confronted with an unexpected situation, it often must make decisions on the basis of available knowledge. It is usually the responsibility of the NRC's Office of Nuclear Regulatory Research (RES) to ensure that the necessary information is available or to initiate research to secure the needed information. A properly structured research program can ensure that such information is available when it is needed. For instance, the Commission has sponsored multi-year research programs into the aging and deterioration of materials (e.g., heavy section steel) and components (e.g., steam generator tubes). The resultant understanding is providing invaluable information in dealing with deterioration of components in operating plants and with many of the issues associated with license renewal.

This report addresses research in 11 areas:

I. Probabilistic Risk Assessment
II. Human Factors and Human-Machine Interface
III. High-Burnup Fuel Performance and Mixed Oxide Fuel
IV. Instrumentation and Control Systems
V. Severe Accidents
VI. Fire Protection Research
VII. Thermal Hydraulics
VIII. Integrity of Steam Generators
IX. Integrity of Pressure Vessels
X. Environmentally Assisted Stress Corrosion Cracking
XI. Nuclear Waste-Related Research (Contribution by the Advisory Committee on Nuclear Waste)

The scope and depth of our reviews of these 11 areas vary considerably and are related to changes that have occurred in the past year or to our perception that additional attention is timely. The ACRS is aware of RES's reorganization and self-assessment activities currently under way, as well as its use of a systematic methodology for prioritizing research activities. The ACRS expects to review those activities at a later date.

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INTRODUCTION

The Nuclear Regulatory Commission (NRC) is undertaking a historic change in the philosophy of regulating nuclear power plants with the introduction of risk-informed regulation. The potential rewards of this approach are significant for the agency and for the licensees it regulates. It can, for example, focus attention and resources on the most important threats to nuclear safety while at the same time offering a potential reduction of the manpower and financial resources required for regulation.

The underlying basis of risk-informed regulation is quantitative risk assessment. It is important that risk-informed decisions be based on adequate knowledge, so that any change in the regulations does not compromise the safety of the public. This means that the uncertainties that were originally addressed by conservatism need clear understanding and quantification. To accomplish this, research will be needed to upgrade the appropriate base of knowledge and methods associated with probabilistic risk assessment (PRA).

The research program in PRA deals with risk assessment for nuclear power plants during normal operation. This program is only minimally adequate for dealing with risk assessment for low-power and shutdown operational modes, even though the perception is that these modes could contribute significantly to the overall plant risk. The risk from fire-initiated accidents, which is applicable to both operating and shutdown plants, is another area in which the PRA knowledge base is inadequate.

Additional work is needed on PRA model development. This report discusses research needed to support such model development in the areas of severe accidents, human factors, fire protection, and instrumentation and control systems. If the Commission is to have a comprehensive risk-informed regulatory program, there must be a larger dissemination of PRA tools and results within the agency.

Nuclear power plants are subject to a variety of degradation processes. Some of these (e.g., fatigue and uniform corrosion) were anticipated in the original design, but others (e.g., stress corrosion cracking, thinning, denting, etc.) were not. The nuclear industry has expended large resources on replacements, repairs, and inspection programs to manage this degradation. An area that is important to the safe operation of plants during their current licensing periods and for license renewal is assurance of the adequacy of aging management programs. For active systems, the maintenance rule is intended to provide this assurance. Passive components, such as reactor coolant pressure boundary, electrical cables, and reactor internals, are also subject to a variety of degradation processes and are the focus of the current license renewal process. Fortunately, study of these degradation processes has been a focus of NRC research for many years and has provided needed insights into the regulatory issues arising from these degradation processes.

The ability to conduct more sophisticated and realistic analyses of the integrity of components such as the reactor vessel and steam generator tubes offers the potential for reductions in conservatism. The NRC must be able to independently verify that any changes to regulatory criteria will not increase the probability of failure of these systems to unacceptable levels. The need for independent verification of regulatory criteria and the technical complexity of the issues require a level of technical expertise that can be maintained by the agency through the continued support of research programs in these areas. The way in which several areas of current research respond to these needs is described in more detail in the body of this report.

The ACRS believes that risk-informed regulation requires a strong research program. A strong research program will also allow the United States to maintain its influence in nuclear safety regulation throughout the world.

SUMMARY REVIEW OF RESEARCH ACTIVITIES

For each of the 11 research areas discussed in this report, we provide some background and our views on program strengths and shortcomings, and make specific recommendations. The ACRS has not attempted in this review to be all-inclusive of either individual areas or of the nuclear safety research program. The Advisory Committee on Nuclear Waste has provided its comments and recommendations on waste-related research and technical assistance programs.

I. Probabilistic Risk Assessment

Recommendations

- RES should expand its investigations of methods for addressing uncertainties beyond just parameter uncertainty to include model uncertainty, especially in the areas of human performance, fire protection, aging, and success criteria.

- Improved PRA models must be developed for the following items:

  - Fires
  - Software-based digital systems
  - Aging of structures, systems, and components (SSCs)
  - Human performance
  - Safety culture

- PRAs should be performed for low-power and shutdown operations in an expedited manner. These should be as rigorous as the PRAs for reactors at power.

Discussion

The cornerstone of risk-informed regulation is a PRA of acceptable quality. The release of Regulatory Guides 1.174, 1.175, 1.176, 1.177, and 1.178 has focused attention on the estimation of core damage frequency (CDF) and large, early release frequency (LERF), since these are the two metrics that play a major role in risk-informed decisions.

Uncertainties due to model assumptions have been of concern in the severe accident arena for a long time. NUREG-1150 has identified one way of quantifying this uncertainty using expert judgments dealing primarily with Level 2 phenomena. This work, of course, is very important in calculating LERF. Model uncertainty has not been treated with equal rigor in the calculations of CDF (Level 1 PRA), although there are isolated instances in which this uncertainty has been addressed. An example is the timing of reactor coolant pump seal loss of coolant accident (LOCA) associated with loss of component cooling water.

There are two needs in this area at this time: (1) the major model uncertainties that exist in the estimation of CDF must be identified, and (2) the methods for quantifying these model uncertainties must be developed.

In addition, PRA models need to be improved to better treat fires, software-based digital systems, aging phenomena, human performance, and safety culture. Further details on these modeling needs are discussed elsewhere in this report.

During our discussions with the staff on the ASME proposal for risk-informed in-service inspection, it became evident that the model uncertainties associated with some aging mechanisms of SSCs could be very large. Quantifying these uncertainties is important to the evaluation of strategies for managing the risks from aging degradation of SSCs.

There is evidence from both the operating experience and the scientific literature that the organizational structure of a nuclear power plant and the prevailing safety culture are major determinants of human performance. The ACRS believes that RES should undertake an effort to identify the contribution of these issues to human performance and to the estimation of CDF and LERF. The ACRS realizes that there are sensitivities regarding the NRC's investigating "management" issues, yet, it believes that there is a need to understand the ways in which these elements may affect the CDF and LERF estimates before decisions are taken regarding regulatory action, if any.

The ACRS notes that the current effort to make the reactor oversight process risk informed lists "problem identification and corrective action processes" (clearly an organizational element) as an issue of concern to the agency. Furthermore, it is decided that "no separate and distinct assessment of licensee safety culture is needed because it is subsumed by either the Performance Indicators or baseline inspection activities" (SECY-99-007, p. 15). The technical basis for these decisions and conclusions is not evident to the ACRS. While they may be valid, the ACRS believes that a systematic investigation of these issues would provide a more convincing argument as to what a risk-informed oversight process should include.

The risk-informed regulatory guides specify that the CDF and LERF values reflect all modes of reactor operation. Plant configurations vary during low-power and shutdown operations, yet, only a few have been investigated rigorously, e.g., mid-loop operations for PWRs and Plant Operational State 5 during refueling for BWRs. Operating experience indicates that a large number of human errors occurs during low-power and shutdown operations. In a risk-informed regulatory system, it is important to have credible estimates of CDF and LERF for all plant configurations and modes of operation. This should be achieved expeditiously.

II. Human Factors and Human-Machine Interface

Recommendation

The ACRS supports the RES strategy to develop a technically justified Human Performance Plan (HPP) and urges RES to proceed aggressively in the development and implementation of the proposed approach.

Discussion

In NUREG-1635, Vol. 1, the ACRS noted a lack of a systematic approach for achieving the three goals of the mission statement of the human performance program; namely, (1) identifying human performance issues important to public health and safety; (2) increasing understanding of the causes and consequences of degraded human performance in such settings; and (3) implementing the appropriate regulatory response to such issues. The ACRS is pleased to report that the strategy presented to it on February 4, 1999 for developing a more technically defensible HPP is promising. This strategy will begin with the identification of agency needs in the field of human performance by reviewing the operating experience, contacting other organizations such as the Institute of Nuclear Power Operations, performing sensitivity studies, and comparing the preliminary findings to error classifications available in the literature. The list of human performance needs for NRC will then be prioritized by a process now being developed within RES. Requirements and a closure strategy for the priority activities will be defined, quantitatively where possible, using regulatory analysis guidelines and risk criteria described in Regulatory Guide 1.174.

The ACRS recognizes that much work needs to be done to implement this strategy. The ACRS continues to believe that a technically defensible HPP is essential to guide the agency's future activities in this very important area. The ACRS, therefore, encourages the staff to continue its efforts to develop such a HPP by following the proposed strategy. A technically defensible HPP will establish the basis for realistically quantifying the risk associated with human performance.

III. High-Burnup Fuel Performance and Mixed Oxide Fuel

Recommendations

- The ACRS finds the confirmatory research program on high burnup fuel performance to be well designed. The program can be improved by including considerations of accident source term and information needs for fuel burnups in excess of current limits.

- RES should conduct a phenomena identification and ranking process for mixed oxide fuel (MOX) to ascertain potential research and regulatory needs.

Discussion

The staff has decided that the licensees will be responsible for providing all of the data and analyses needed to approve requested extensions of fuel burnup beyond the current limit of 62 GWd/t. It is very likely that the nuclear industry will make requests for burnups beyond this limit. Indeed, the industry now has under way an extensive research program to support such requests. The ACRS has agreed with this regulatory approach and suggested that the staff define the kinds of data and analyses that will be needed to evaluate requests for extended fuel burnups.

As a step in the definition of the information needed, RES plans to use expert opinion to develop a phenomena identification and ranking of issues that are likely to arise in the evaluation of an application for extended fuel burnup. This expert judgment effort will attempt to define the important physical and chemical phenomena, identify what is known about these phenomena for high burnup fuel, and ascertain the nature of data and analyses concerning the phenomena that will be needed to approve applications for extended fuel burnup. The ACRS believes that this is an excellent way to approach the problem.

The current NRC research on high burnup fuel does not address some key technical issues that arise concerning the effects of burnup. One such issue is the effect of high burnup on the releases of fission products under accident conditions. The NRC has issued a revised source term for the estimation of fission product releases during design basis accidents. This revised source term was developed based on tests and analyses of lower burnup fuel. Correlations of the fission product release when fuel cladding is breached, the so-called gap release, based on this data may seriously underestimate the gap release from high burnup fuel. The ACRS has not heard of plans by the NRC staff to conduct research to make the revised accident source term applicable to high burnup fuel.

Another issue that will eventually come to the NRC is the use of MOX in commercial nuclear power plants. The U.S. Department of Energy (DOE) plans to utilize MOX in commercial reactors as a way to dispose of excess weapons grade plutonium. Existing regulations for commercial nuclear power plants have not anticipated the use of such fuel. There has been some experience worldwide with the use of MOX, but little of this experience addresses conditions outside the range of normal operations. Contentions are being made in some quarters that such use of MOX will pose

qualitatively different threats in the event of a reactor accident. The technical data to confirm or refute these contentions are not now available. While not a priority issue now, such concerns may well arise as the DOE efforts mature. The ACRS anticipates that experimental efforts of a fairly long duration will be needed to address issues concerning the behavior of MOX under accident conditions. The ACRS believes that RES should conduct a phenomena identification and ranking process for MOX to ascertain potential research and regulatory needs for MOX. The phenomena identification and ranking process will position the agency to respond promptly to an application by a licensee to use MOX.

IV. Instrumentation and Control Systems

Recommendations

- The staff should complete research on the effects of lightning on instrumentation and control (I&C) systems, specifically digital I&C systems, and continue research on the review of software-based digital system specifications and requirements.

- The staff should assess whether "hot shorts" arising during a fire produce significant electromagnetic interference (EMI) and, if so, whether current guidance on electromagnetic interference/radio frequency interference (EMI/RFI) is adequate.

- The staff should continue its current research on methods for estimating failure probabilities of systems that include embedded software digital components for use in PRAs.

- The staff should complete a review of advanced measurement methods and their potential to improve measurement of safety-related variables in nuclear power plants.

Discussion

Digital I&C Systems:

The introduction of digital I&C systems in nuclear power plants offers many advantages; it also presents new technical issues. These issues can be categorized as follows: (1) internal and external stressors that may cause changes or failures and (2) quality and reliability of the software inherent in most digital systems.

NUREG/CR-6579 provides a risk screening of external stressors on digital systems and indicates that humidity, EMI from lightning, and smoke can be potentially risk significant. NUREG/CR-6406 reports that a combination of high temperature and high relative humidity is a condition that could affect the performance of digital systems. These reports, as well as NUREG/CR-6543, conclude that, by using accepted design methods, quality assurance, qualification, and/or commercial dedication, along with surveillance and testing, the risk from these stressors can be reduced to levels acceptable for safety-related systems. Completion of regulatory guidance for these external stressors along with the recently issued regulatory guide on EMI/RFI, would provide guidance on all important external stressors on digital systems except for the effects of lightning.

A risk-significant issue related to fire protection in nuclear plants is the potential impact of a fire on I&C circuits, specifically "hot shorts" (shorts between two or more signals or control wires connected to different circuits). If the hot short includes a high-voltage circuit, it could be a source of EMI. The ACRS believes that the staff should assess whether there is EMI caused by "hot shorts" and if there is, whether current guidance on EMI/RFI is adequate.

Software, which is inherent in most microprocessor-based digital systems, is the basis for many of the advantages of digital systems. Software-based systems, however, have attributes that require them to be developed with a different and more thorough attention to detail and quality assurance than is required to develop hardware.

Software is a set of instructions for the operation of hardware. If those instructions are written with no errors and if they are designed to operate the hardware correctly for all anticipated conditions, which include all foreseen and unforeseen external conditions and all circumstances inherent to the hardware which the software is designed to operate, there will never be a failure. Software, however, does fail, and failures are most often attributed to errors in system and/or software specifications and requirements, and to a lesser extent to errors in writing software.

The accepted software engineering method for minimizing errors related to requirements and specifications and to programming errors is to develop software in a highly structured and disciplined process, which calls for internal testing to ensure the product of each stage in the development process meets the requirements specified for that stage. The final test is an assessment of the final product to see if it meets system design requirements. Unfortunately, this approach is not able to ensure that errors in developing the requirements for complex systems are always reduced to a level acceptable for some safety critical applications, such as safety-related applications in nuclear power plants. Current regulatory guidance for digital system design provided in Chapter 7 of the Standard Review Plan is based on these software engineering methods.

In its June 1997 letter, the ACRS recommended that the Commission issue the update of Chapter 7 of the Standard Review Plan also recommended that "for systems that contain digital components, the effect of software failure should be included in the probabilistic risk assessment (PRA)." This recommendation was consistent with one of the National Research Council Study recommendations that "The USNRC should develop methods for estimating failure probabilities of digital systems, including commercial off-the-shelf (COTS) equipment for use in PRAs."

The staff currently has research in progress to improve the technical basis for review of software-based digital systems with emphasis on assessment of the adequacy of system and software specifications and requirements and research to develop the technical basis for estimating failure probabilities of digital systems for use in PRAs. The ACRS believes that this effort must be continued and, if necessary for timely completion, be increased.

Sensors that measure the physical variables that are used by safety and control systems have not changed significantly since the design of the first nuclear plants more than 40 years ago. New measurement methods that integrate new sensor technology, such as optical-based sensors and embedded microelectronics that perform on-line calibration, diagnostics, testing and compensation for external effects, provide opportunities for increased measurement accuracy with lower uncertainty. These new measurement methods represent the next I&C area where new technology will have a significant impact on nuclear plant reliability, safety, and efficiency.

The NRC staff should have the information necessary to make prompt regulatory judgments concerning licensee applications to use advanced sensors in nuclear power plants. The ACRS believes that the staff should complete a review of advanced measurement methods and their potential to improve measurement of key safety-related variables in nuclear power plants. This review should include an assessment of NUREG/CR5501, which addresses a variety of advanced sensors for making better measurements in nuclear power plants.

V. Severe Accidents

Recommendation

To meet its needs in the severe accident area, the agency should strengthen its participation in international research activities that are already in existence (e.g., high burn-up fuel behavior, aerosol behavior, fuel-coolant interactions, and thermal hydraulics). Such participation in international research is necessary because of the very high cost of performing research in this area.

Discussion

The current NRC plans appear to be to essentially closeout research in the severe accident area while providing only a minimum level of effort for maintaining the suite of NRC severe accident codes and continuing with minimum (already committed) participation in international cooperative programs. With this "minimum" strategy, the ACRS is concerned that information relating to severe accidents needed to support the transition to a risk-informed regulatory system will not be available.

As the agency moves toward a risk-informed regulatory system, there will be increased tension between the level of uncertainty in the risk determinations and the need for appropriate defense-in-depth to ensure that there is sufficient margin with respect to undue risk. The ACRS believes that the current level of uncertainty in the risk assessments caused by severe accident issues is so large that the risk-informed regulations inevitably will have to remain highly conservative and include substantial (and somewhat arbitrary) defense-in-depth provisions.

The determination of these uncertainties can be sharpened and their magnitudes substantially reduced with additional severe accident research in the following selected areas:

- Fission product release from high burnup fuel
- Effects of mixed oxide fuel on core-melt behavior
- Release of fission products from molten fuel pools in the absence of sparging
- Hydrogen mixing and distribution in containment

VI. Fire Protection Research

Recommendation

RES should develop a definitive plan to enhance the agency's capabilities regarding fire risk assessment methods.

Discussion

Fire protection requirements for nuclear power plants have been developed on a deterministic basis in the past. Indeed, licensees have indicated a relative comfort with the deterministic fire protection requirements, even though complying with these requirements has imposed significant costs on the nuclear industry. Certainly, this comfort on the part of licensees has slowed the use of quantitative methods to estimate risk to the public from fire-initiated accidents at nuclear power plants. Where quantitative methods have been used in the individual plant examination of external events (IPEEE) and in a few external events PRA, findings indicate that fire initiated accidents can be significant contributors to the risk posed by a nuclear power plant. These findings suggest strongly that it will be necessary to routinely estimate risks from fire in a risk-informed regulatory framework.

The fire risk assessment methods available to the NRC are not developed to a level comparable to the methods for assessing risk during normal operations. The fire risk assessment methods and insights from analyses using these methods have not been disseminated throughout the agency in a way comparable to what has been done for risks from internal events. There is not within the NRC an activity for assessing fire events comparable to the Accident Sequence Precursor program that is used for evaluating other types of nuclear power plant events. Senior Reactor Analysts in the regions do not have well-developed methods for assessing fire risks.

There is now a research activity in NRC to improve the fire risk assessment methods. This research has begun to address a selected few of the many deficiencies previously identified in the fire risk assessment technologies. There is not, however, a well-developed plan to show that these research activities will, in fact, yield the kinds of tools that the agency will need in its move toward risk-informed regulation. There is not a demonstration that research to correct deficiencies not now being studied can, indeed, be deferred to later years and still meet the aspirations the Commission has for refining the regulatory process. Absence of a definitive plan for the development, application, and dissemination of fire risk assessment methods within the agency can be attributed in no small part to the slow process by which insights from the IPEEE submittals are being developed. The needs for adequate capabilities to assess fire risks are becoming more painful to the agency as it proceeds towards making NRC programs for enforcement, inspection, and plant assessment risk informed. Difficulties have been encountered in the risk ranking of violation of fire protection requirements. Decisions on changes to the fire protection core inspection as a result of the Fire Protection Functional Inspection Pilot Studies will be greatly facilitated by good technologies for fire risk assessment. One can easily imagine that decisions on applications for inservice testing and online maintenance could be facilitated by reliable fire risk assessment methods.

VII. Thermal Hydraulics

Recommendations

- The NRC needs to maintain a capability to run thermal-hydraulic codes and to evaluate code predictions to support future regulatory decisions.

- Code consolidation now under way should lead to efficient focus on one set of technical enhancements and considerable savings in maintaining one code instead of three or four different ones. Emphasis should be on improvement, rather than compromise that preserves several poor features from each previous code. Figures of merit, such as gains in run time, robustness, accuracy, and fidelity, need to be demonstrated by clear "before" and "after" comparisons.

- The NRC should identify and address those features of codes that need modification in order to supply realistic, rather than conservative, evaluations of accident scenarios. Ways need to be developed to quantify uncertainties in code predictions.

- More specific justification of new experiments and the expected payoff from their results is needed. New experiment efforts should be judged by how they complement past results and how they improve the quality of specific capabilities in the thermal-hydraulic codes.

- The NRC should continue to encourage new ideas that offer substantial benefits for improving accuracy, coherence, and robustness of its thermal-hydraulic codes. Anticipatory research that is planned is very modest for an area of such fundamental importance to the realistic prediction of accident scenarios in which present methods of prediction have significant limitations.

Discussion

As the nuclear industry is challenged to perform more competitively in the future, in response to deregulation and low costs of alternative electrical generation, the NRC will be asked to approve power uprates and relaxation of other technical constraints that have the potential to reduce real margins of safety. The Commission will need assurance that these decisions can be made with confidence. Since the consequences of accidents depend on thermal-hydraulic phenomena and there is a sparse data base of experience with accidents, except for the Three Mile Island event and a handful of lesser transients with no major consequences, reliance is placed on the predictions of "best-estimate" thermal-hydraulic computer codes.

Nuclear accidents involve multiphase flows of steam, water, and noncondensable gases. In extreme cases, other materials may melt, vaporize, or be entrained, contributing to the complex mixture to be analyzed.

The thermal hydraulics of multiphase flow is far from being a precise science. It is not based on exact equations. Rather it uses "engineering" methods that evolve as more complete formulations, empirical results, and theoretical insights are achieved. Present codes contain many assumptions and coefficients that do not have an extensive validation for reactor conditions and may even be "wrong" in an academic sense if they yield an incorrect answer to a simple, well-defined problem. The codes must be carefully compared with representative data because they may not provide accurate predictions for situations not considered when the codes were developed.

The original confirmation of the adequacy of codes for Appendix K calculations was a long process, relying on extensive experiments (e.g., Loss of Fluid Test) which NRC is unlikely to repeat. When doubt existed, conservative assumptions were adopted in the codes. To assess the risk associated with the changes in plant configuration or operating procedures that burden-reduction measures require, code predictions that are closer to reality are needed. In addition to the "best estimate" of what is likely to happen, the uncertainty in these code predictions emanating from model assumptions and parameter values must be quantified.

The validation of present codes for best-estimate use is problematical because their many assumptions are not "realistic." The assumptions were made as expedients when codes were developed decades ago. A mature level of technical knowledge, sophistication, and tenacity is required in order to discern how far the codes can be trusted, when they need improvement, and when recourse must be made to experiments. Code review and validation is a major task that must be done thoroughly, otherwise faults may remain hidden until misleading predictions are revealed in a regulatory context.

The ACRS perceives the need for "best estimate" code review and validation as being insufficiently appreciated, and even discouraged, by the NRC management. The eventual check is a thorough time-consuming investigation by the ACRS -- far too late in the process to ensure efficient use of agency resources. Our experience with reviewing codes for AP600 and preliminary experience with codes presently under review suggests that the ACRS and its consultants will find many deficiencies related to physical modeling and code documentation. These deficiencies should be discovered by the NRC staff reviewers who are in a much better position to delve into the coding details.

Key technical issues may require resolution by RES's thermal-hydraulic research group which has lost several FTEs during the past year.

Recently, the NRC has taken steps to improve the effectiveness and logic of its regulatory processes. Real improvements in public safety may be elusive if the technical bases for modeling accident phenomena are neglected. Vendors may be expected to restrain the enthusiasm with which they probe the limitations in their codes unless the NRC leads the way. The NRC has an obligation to public safety to develop and exercise its own capability to check prediction methods and results. Consultants and contractors may help with specific expertise, but the staff must be knowledgeable and insistent enough to ensure critical evaluation of sufficient quality. To do this, the NRC staff needs experience with running its own "best-estimate" thermal-hydraulic code.

Formerly, the NRC supported the development of a number of light water reactor thermal-hydraulics codes (e.g., TRAC-PWR, TRAC-BWR, RELAP), with most of the development and applications work being done at the national laboratories. The thermal-hydraulics program is now focused on the formulation of a single thermal-hydraulics and neutron transport code. Although some outside contractor development help will be needed, the goal which the ACRS supports is to achieve the in-house code capability needed for the evaluation of future license applications.

This research program is more than a code consolidation effort. It will also correct some of the identified weaknesses in the previous thermal-hydraulics codes. For example, in the course of the AP600 design certification review the NRC staff and the ACRS uncovered many deficiencies in the existing suite of thermal-hydraulics codes and databases for the application of these codes to new designs.

While the ACRS endorses this short-term strategy of code consolidation and improvement, it is concerned that the NRC may be too easily satisfied with the latest evolution of what it has called "archaic codes" that are "becoming obsolete and are difficult to use and maintain." Only recently has RES begun to thoroughly investigate the details of the TRAC codes, discovering unrealistic physical models, unexplained coefficients and undocumented changes that have been patched together over many years. These codes may require a quantum leap in quality improvement if they are to form a reliable basis for "realistic" or "best estimate" calculations that should provide opportunity for relaxing earlier conservative assumptions (they may also reveal if the assumptions were really conservative when many phenomena interact during a long scenario). Moreover, future codes must provide a quantitative measure of uncertainty, which is not now possible. Additionally, risk assessment will require that sensitivity studies be made of the effects of assumptions or empirical coefficients. This will require codes that run much faster than they do currently.

When existing codes have been successfully consolidated, they will still be based on decades-old methods while the state of the art has not remained static. NRC should plan a strategy for identifying computational needs of the future and incorporating improved numerical methods and physical modeling into its expertise in order to achieve more confidence and precision in predicting transients and accidents. In the commercial arena, computational fluid dynamics (CFD), including multiphase flow, is a very active and competitive area of development with new features continually emerging. While the small NRC-sponsored program at Rensselaer Polytechnic Institute provides a window on a particular academic development effort in multi phase CFD, it should be supplemented by a broader assessment of possible benefits from adopting capabilities from other sources.

The work on interfacial area modeling, centered on Purdue University, is a rare example of long-range research. The objective is to replace the present computationally awkward set of flow regimes and correlations with a unified predictive approach. While this is an ambitious and unproven endeavor, there are large potential benefits in terms of code simplicity, stability, speed, and structural cohesion. The ACRS regrets that this is the only promising new idea currently being pursued in response to the need, already mentioned, for considerable improvements in code quality, which should still be possible in this relatively immature branch of science.

Since practical two-phase thermal-hydraulic phenomena cannot now, and probably never will, be predicted from first principles, there is an ongoing need for an experimental reality check on all analyses and assumptions used for predicting the course of nuclear accidents. Separate-effects tests are needed in order to improve modules in the codes and, eventually, to reach a stage where complete systems perhaps need not be tested every time a design is changed, as is now necessary because of uncertainty in combined uncertainties. NRC should better articulate the specific needs for new tests and the payoff in terms of improved predictions and regulatory consequences. The primary goal should be to support NRC's own code development, with the expectation that industry will bear the burden of providing new data for evaluating its codes.

NRC has maintained medium-scale test facilities at Oregon State University (APEX) and Purdue University (PUMA), primarily to support evaluation of new reactor concepts. They are being adapted to answer a few unresolved technical questions, such as phase separation at tees. A thorough and comprehensive evaluation needs to be made about how many such questions remain and a decision made whether these facilities are the best place to answer them. However, we have qualms about the NRC giving up too many experimental capabilities out of short-term considerations, since we anticipate that future issues will emerge that require experimental resolution. Rebuilding facilities from scratch will involve delays and costs.

On a related issue, the ACRS is concerned lest the wealth of data generated in the heyday of thermal-hydraulic research decades ago become lost, inaccessible or unappreciated, leading to unnecessary duplication of effort. For example, a major program has recently been initiated at Pennsylvania State University to reinvestigate PWR core reflood, as a result of deficiencies found in existing codes. It appears to repeat the objectives of older FLECHT tests. The ACRS would like to see what new information is being generated by this new program and how the anticipated results will quantitatively improve predictions relevant to safety decisions.

VIII. Integrity of Steam Generators

Recommendation

RES should provide additional data, methods, and models to evaluate the technical credibility of licensee programs on steam generator integrity.

Discussion

Steam generators have been historically the most troublesome components in pressurized water reactors. Industry efforts have been largely successful in managing degradation due to wastage, pitting, and denting; however, fretting, stress corrosion cracking (SCC), and intergranular attack have proved more difficult to manage. The replacement cost for steam generators is on the order of $100M plus the cost of replacement power. Even for those plants which have not yet had to replace their steam generators; inspection, monitoring, and repair are expensive. Thus, there is substantial industry interest in operating with degraded steam generators, and a significant fraction of the plants will probably not replace their steam generators during their operational life.

Highly prescriptive requirements have been used by NRC to regulate the integrity of steam generators. This regulatory process ensures safety because a "conservative" approach is taken when sufficient information is not available. A performance-based approach, better focused on the types of the degradation actually occurring in steam generators, has the potential for reduced burden in steam generator tube plugging requirements with minimal changes in plant risk. The staff and the industry are developing performance-based regulatory guidance for the assessment of the integrity of steam generator tubes. Steam generator tubing is unique in that it has an important containment function as well as being by far the largest component of the reactor pressure boundary. The NRC must have the capability to independently evaluate the technical credibility of licensee programs to meet the performance criteria being developed.

The objectives of the NRC research are to provide the data, correlations, and models needed to permit the NRC to assess the validity of the programs proposed by industry to manage the degradation of steam generators. The focus of the program is on: (1) assessment of personnel, procedures, and equipment used for inservice inspection (ISI); (2) validation and assessment of correlations and models for evaluating the integrity of and leakage from degraded steam generator tubes under design basis and severe accident conditions; and (3) validation and assessment of correlations and models for predicting degradation generation and progression in steam generator tubes.

IX. Integrity of Pressure Vessels

Recommendation

RES should continue its efforts at verification of current regulatory criteria and maintain technical expertise in this area to respond to anticipated industry initiatives.

Discussion

The Reactor Vessel Integrity Program has already provided great insight into the regulatory questions arising from vessel embrittlement. The generic analyses in NUREG/CR–6023 and the submittals from the owners groups demonstrate that all PWRs and BWRs in the U.S. should have adequate upper shelf toughness at least through the end of their current licensing period. The recent change from "crack arrest" toughness to "crack initiation" toughness for the calculation of pressure temperature limits for reactor vessels will significantly decrease heatup constraints for operating plants. A screening limit for pressurized thermal shock (PTS) of PWR vessels has been developed. Almost all U.S. plants will be well below this screening limit through their current licensing period, although PTS concerns could pose difficulties for a significant number of plants considering license extension.

Issues related to the embrittlement of reactor pressure vessels (RPVs) will continue to arise both for operating reactors during their current licensing period and in the context of license extension. Current regulatory requirements for pressure vessels are believed to ensure vessel integrity. The conservatisms in these requirements, however, impose significant regulatory burdens, operational constraints, and large economic costs on the industry. These conservatisms could be detrimental to safety because they may distort the allocation of resources. They may also increase the exposure of personnel performing unnecessary inspections and may unnecessarily restrict operational flexibility during plant heatup and cooldown operations.

Although reductions in conservatism in the assessment of vessel integrity can and should be made, the NRC must be able to independently verify that any changes to regulatory criteria will not increase the probability of vessel failure to unacceptable levels. The ACRS feels that this need for independent verification of regulatory criteria and the technical complexity of the issues require a level of technical expertise that can be maintained by the agency through the continued support of a significant research program in this area. The budget for the program has been significantly reduced from FY 98 to FY 99. Further reductions may be forced by competition for resources by other high priority research needs, but a core program will need to be maintained. Work should continue on the "regulatory applications" of this research in the development of a revised PTS rule and the updating of Regulatory Guide 1.99, Revision 2. Work should continue on the basic processes of embrittlement and fracture to assess potential nonconservatism in the present understanding.

X. Environmentally Assisted Stress Corrosion Cracking

Recommendation

A cooperative research program by industry and NRC on environmentally assisted cracking (EAC) is needed to ensure the integrity of the reactor pressure boundary and reactor internals.

Discussion

Environmentally assisted cracking of reactor structural materials has been a recurrent problem in operating reactors since the mid-1970's. The initial focus was on Boiling Water Reactor (BWR) pipe cracking problems. Since that time, the emphasis has shifted to other problems in EAC such as Irradiation Assisted Stress Corrosion Cracking (IASCC). The industry, largely through the BWR Vessel and Internal Project Program, is developing inspection, mitigation, and repair/replacement programs to address these problems. Because cracking depends on a complex interaction between materials, fabrication processes, irradiation conditions, and water chemistry, there are still substantial uncertainties in crack growth rates which must be reflected in inspection requirements and in integrity assessments. NRC and industry research in this area have recently provided the basis for the staff to accept a reduction of a factor of two in the crack growth rates used for integrity assessments of defected stainless steel components, which reduces the need for mid-cycle inspections in many cases.

The ACRS reviewed the current research on the EAC which deals with three issues of EAC: fatigue of reactor structural materials, IASCC, and EAC of nickel based alloys and weld metals. This includes work done by NRC contractors and participation in the Cooperative IASCC Research (CIR) Program, which is managed by the Electric Power Research Institute (EPRI) and which includes contributions from vendors, utilities, the NRC, and regulatory authorities in Europe and Japan. The overall objective of the program is to provide data and physical models to assess environmentally assisted degradation of primary pressure boundary components in light water reactors. In addition, crack growth models developed by industry and submitted for evaluation and approval by the NRC are being studied. The findings from this research program will enable the staff to make independent evaluations of industry assessments of the operability of degraded components. The ongoing work may lead to additional inspection relief. The work is also strongly leveraged through the access it provides to the information being developed in programs in Europe and Japan.

XI. Nuclear Waste-Related Research

1999 UPDATE ON THE REVIEW OF THE NRC SAFETY RESEARCH PROGRAM CONCERNING WASTE-RELATED WORK

As reported last year (NUREG-1635, Vol. 1), the Advisory Committee on Nuclear Waste (ACNW) is charged with review of safety research in waste management. There is a relatively small ($2.3 million a year in Radionuclide Transport and $1.3 million a year in Radiation Protection and Health Effects) research program in the Office of Nuclear Regulatory Research (RES) in this area. The Office of Nuclear Material Safety and Safeguards (NMSS) contracts with the Center for Nuclear Waste Regulatory Analyses (CNWRA) for technical assistance ($13.1 million a year), much of which is, in essence, research related to the Yucca Mountain program. Thus, the ACNW also considers work at the CNWRA to be included in its review.

The RES staff made a presentation to the ACNW at its 106th meeting in February 1999. Although we submitted a series of questions to NMSS, we received no formal answers to them, nor were any members of the staff available to make presentations at our meetings primarily because the viability assessment review consumed most of the staff's time during this period. Our comments on the program at the CNWRA are based on reading the Technical Program Description for the CNWRA. The ACNW staff did meet with several scientists from NMSS and the CNWRA to discuss some of the questions that we posed. We are not currently in a position to comment on the adequacy of the CNWRA to provide all of the technical support necessary for review of a license application from the Department of Energy (DOE). We plan to visit the CNWRA in the near future.

Office of Nuclear Regulatory Research

Our main recommendation last year was that RES adopt a way to prioritize the research that is sponsored. With only modest funds available for waste-related research, the only way to have an effective program is to focus on critical issues. We understand that RES is considering the use of an Analytical Hierarchy Process to aid it in establishing research priorities. We also understand that RES is engaged in a self-assessment that may affect future directions with respect to prioritization of research programs. We look forward to hearing about the revisions to the current research program that result from these efforts.

We have become concerned that the waste-related research program within RES may be too small to accomplish what the NRC may need. This is not to say that specific projects being supported are not valuable; indeed, the program is supporting some excellent individuals and important work. We do recommend, however, that definitive goals for the program be set and that funding commensurate with the goals be provided. Perhaps this will be a product of the self-assessment process. A program that is too small to be effective can be continued for the short term, but it should not be allowed to consume resources with little likelihood of producing important results for the regulatory program in the long term.

Office of Nuclear Material Safety and Safeguards — CNWRA

In the continuing interactions that the ACNW has with the staff on the Yucca Mountain project, we have gained considerable insight on the nature and quality of the work being performed in high-level waste. Our conclusion is that much important work is being accomplished by the staff on performance assessment, corrosion, geochemistry, and so forth. We are generally favorably impressed with specific projects of the ongoing work. Nevertheless, we have concerns about the process for prioritizing work.

The ACNW was told last year that results from the performance assessments of Yucca Mountain were the major drivers for prioritizing work in the CNWRA program. On the basis of this information, the ACNW recommended last year that NMSS continue to use Total Performance Assessment (TPA) code results to guide its work. Our review of the program at the CNWRA for this year leads us to question whether NMSS truly is doing what it says. We list two pieces of evidence below that lead to our concern that the overall program is not being driven by high-level strategic concerns.

1. The CNWRA work for the coming year includes substantial effort related to Igneous Activity. We have seen no convincing argument that such work should receive significant support. In fact, the TPA results that we have seen led us to suggest more than a year ago that the program in Igneous Activity be phased out.

2. The ACNW wrote a letter on the Engineered Barrier System (EBS) on September 9, 1998. In it we recommended that consideration be given to enhanced research (i.e., technical assistance) in a few areas related to near-field chemistry and corrosion. (Increasing emphasis by DOE on the EBS, along with TPA results, suggests that these areas should have high priority.) The EDO response said that there was limited flexibility in the program and that the ACNW should indicate what items might be cut to make room for new, high-priority research. Again, this response suggests that the status quo is the baseline for the program.

We recommend that NMSS adopt a procedure for prioritizing work at the CNWRA using results from the TPA code as a primary driver and that decisions be based on the prioritization. In particular, we reiterate our recommendation that work on Igneous Activity be ended unless a credible case can be made for its continuation.

We also reiterate our recommendation of last year that the CNWRA make use of external experts as it takes on work. This advice may be especially important in the area of engineering, where we have not seen clear indication that either NMSS or the CNWRA has sufficient depth in design engineering to allow full evaluation of DOE's repository system for Yucca Mountain.

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APPENDIX A: ACRONYMS

ACNW	Advisory Committee on Nuclear Waste
ACRS	Advisory Committee on Reactor Safeguards
APEX	Advanced Plant Experiment
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CFD	Computational Fluid Dynamics
CIR	Cooperative IASCC Research
CNWRA	Center for Nuclear Waste Regulatory Analyses
COTS	Commercial Off The Shelf
DOE	Department of Energy
EAC	Environmentally Assisted Cracking
EBS	Engineered Barrier System
EDO	Executive Director for Operations
EMI	Electromagnetic Interference
EPRI	Electric Power Research Institute
FLECHT	Full-Length Emergency Cooling Heat Transfer
FTE	Full-Time Equivalent
GWd/t	Giga Watt Days/Ton
HPP	Human Performance Plan
IASCC	Irradiation Assisted Stress Corrosion Cracking
I&C	Instrumentation and Control
IPEEE	Individual Plant Examination of External Events
ISI	Inservice Inspection
LERF	Large, Early Release Frequency
LOCA	Loss of Coolant Accident
LOFT	Loss of Fluid Test
LP/SD	Low Power/Shutdown
LWR	Light Water Reactor
MOX	Mixed Oxide Fuel
NMSS	Office of Nuclear Material Safety and Safeguards
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
PTS	Pressurized Thermal Shock
PUMA	Purdue University Multidimensional Integral Test Assembly
PWR	Pressurized Water Reactor
RELAP	Reactor Leak and Analysis Program
RES	Office of Nuclear Regulatory Research
RFI	Radio Frequency Interference
RPV	Reactor Pressure Vessel

SCC	Stress Corrosion Cracking
SECY	Office of the Secretary of the Commission
SSCs	Systems, Structures, and Components
TPA	Total Performance Assessment
TRAC	Transient Reactor Code Analysis
VIP	Vessel Integrity Program

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In 1998, the Advisory Committee on Reactor Safeguards (ACRS) submitted to the Nuclear Regulatory Commission (NRC) a comprehensive report of the NRC's Safety Research Program, NUREG-1635, Vol. 1, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," which documented the ACRS conclusions and recommendations. The ACRS continues to support the conclusions and recommendations of that report. The present report is more modest in scope and is intended to provide additional information concerning the research needed to support Commission programs, especially risk-informed regulation. Not all research programs are included in this report and our observations are often limited to certain aspects of a given program.

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