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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 SAFEGUARDS
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ABSTRACT

This report to the U.S. Nuclear Regulatory Commission (NRC) presents the observations and recommendations of the Advisory Committee on Reactor Safeguards (ACRS) concerning the NRC Safety Research Program being carried out by the Office of Nuclear Regulatory Research (RES). In its evaluation of the NRC research activities, the ACRS considered the programmatic justification for the research as well as the technical approaches and progress of the work. The evaluation identifies research crucial to the NRC missions. The report also addresses the issue of long-term sustained research at the NRC. This report does not address the research being done at NRC on issues of reactor security or the threat of sabotage. The ACRS views on current work in the area of security have been reported in separate documents. Two pertinent, interdisciplinary efforts, the State-Of-the-Art Reactor Consequence Analyses (SOARCA) project and the study of sump screen blockage are not addressed in this report. These projects are actively followed by the Committee. The ACRS has been providing interim reports on the technical approach and progress of these activities.

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ABBREVIATIONS

ABR	Advanced Burner Reactor
ACRS	Advisory Committee on Reactor Safeguards
ACR-700	Advanced CANDU Reactor-700
AEOD	Office for Analysis and Evaluation of Operational Data
AMPX	A Modular Code System for Processing X-Sections
ANS	American Nuclear Society
ASP	Accident Sequence Precursor Program
ATHEANA	A Technique for Human Event Analysis
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CANDU	CANada Deuterium Uranium
CAROLFIRE	Cable Response to Live Fire
CFD	Computational Fluid Dynamics
CFR	Code of Federal Regulations
COL	Combined License
DOE	Department Of Energy
EPIX	Equipment Performance and Information Exchange System
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
FY	Fiscal Year
GNEP	Global Nuclear Energy Partnership
GSI	Generic Safety Issue
GWd/t	Giga Watt day per metric ton
HERA	Human Event Repository and Analyses
HRA	Human Reliability Analysis
HTGR	High Temperature Gas-cooled Reactor
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
IEEE	Institute of Electrical and Electronics Engineers
IRIS	International Reactor Innovative and Secure
ISI	In-Service Inspection
LBLOCA	Large-Break Loss-Of-Coolant Accident
LER	Licensee Event Report
LERF	Large Early Release Frequency
LHS	Latin Hypercube Sampling
LMR	Liquid Metal-cooled Reactor
LOCA	Loss-of-Coolant Accident
LWR	Light Water Reactor
MACCS	MELCOR Accident Consequence Code System
MOX	Mixed Oxide
NDE	Non-Destructive Examination
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NEWT	NEW Transport algorithm

ABBREVIATIONS (Cont'd)

NFPA	National Fire Protection Association
NMSS	Office of Nuclear Material Safety and Safeguards
NRC	Nuclear Regulatory Commission
NRO	Office of New Reactors
NRR	Office of Nuclear Reactor Regulation
NSIR	Office of Nuclear Security and Incident Response
OECD	Organization for Economic Cooperation and Development
PARCS	Purdue Advanced Reactor Core Simulator
PCI	Pellet Cladding Interaction
PFM	Probabilistic Fracture Mechanics
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PSHA	Probabilistic Seismic Hazard Analysis
PTS	Pressurized Thermal Shock
PUMA	Purdue University Multidimensional Integral Test Assembly
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
ROP	Reactor Oversight Process
RPV	Reactor Pressure Vessel
SCALE	Standardized Computer Analysis for Licensing Evaluation
SDP	Significance Determination Process
SER	Safety Evaluation Report
SNAP	Symbolic Nuclear Analysis Package
SNL	Sandia National Laboratories
SOARCA	State-Of-the-Art Reactor Consequence Analyses
SPAR	Standardized Plant Analysis Risk Model
SSHAC	Senior Seismic Hazard Analysis Committee
THIEF	THermally Induced Electrical Failure
TRACE	TRAC-RELAP Advanced Computational Engine
U.S.	United States
USGS	United States Geological Survey

I. INTRODUCTION

In this report, the Advisory Committee on Reactor Safeguards (ACRS) presents the results of its review and evaluation of the Nuclear Regulatory Commission (NRC) Safety Research Program. The NRC maintains a Safety Research Program to:

- Ensure its regulations and regulatory processes have sound technical bases and these bases are refined as new knowledge develops.
- Prepare for anticipated changes in the nuclear industry that could have safety implications.
- Develop improved methods to carry out its regulatory responsibilities.
- Maintain an infrastructure of expertise, facilities, analytical capabilities, and data to support regulatory decisions.

The current research program, organized by the Office of Nuclear Regulatory Research (RES), is closely coupled to specific, near-term issues to support regulatory activities and initiatives in the Offices of Nuclear Reactor Regulation (NRR), New Reactors (NRO), Nuclear Material Safety and Safeguards (NMSS), and Nuclear Security and Incident Response (NSIR). For the purposes of this report, the ongoing research has been examined in terms of the following technical disciplines:

- Digital Instrumentation and Control Systems
- Fire Safety
- Reactor Fuel
- Neutronics Analysis, Core Physics, and Criticality Safety
- Human Factors and Human Reliability
- Materials and Metallurgy
- Operational Experience
- Probabilistic Risk Assessment
- Seismic and Structural Engineering
- Severe Accidents
- Thermal Hydraulics
- Advanced Non-LWR Designs

This report does not address the research being done at NRC on issues of reactor security or the threat of sabotage. The ACRS views on current work in the area of security have been reported in separate documents. Two pertinent, interdisciplinary efforts, the State-Of-the-Art Reactor Consequence Analyses (SOARCA) project and the study of sump screen blockage are not addressed in this report. These projects are actively followed by the Committee. The ACRS has been providing interim reports on the technical approach and progress of these activities.

Chapter 2 of this report provides a synoptic account of research activities in each of the technical disciplines and highlights some of the accomplishments of the work. Additional details on each of the research areas are included in Chapters 4 through 15.

In its review of the NRC Safety Research Program, the ACRS has focused on the technical and regulatory justification for the ongoing research activities. The ACRS supports research that:

- Identifies and resolves current safety and regulatory issues.

- Provides technical bases for the resolution of foreseeable safety issues.

- Develops the capabilities of the agency to independently review risk-significant proposals and submittals by licensees and applicants.

- Supports agency initiatives, including the move toward a much greater use of risk information in the regulatory process and to evolve NRC safety regulations to be "technology neutral".

- Improves the efficiency and effectiveness of the regulatory process.

- Maintains technical expertise within the agency and associated facilities in disciplines crucial to the agency mission and that are not readily available from other sources.

As requested by the Commission in its June 22, 2007 Staff Requirements Memorandum (SRM), the ACRS examined the safety research program to identify any gaps as well as ongoing research projects that have progressed sufficiently to meet the regulatory needs so that they can be curtailed in favor of more important research activities. The Committee finds the ongoing research program well directed to address the short-term needs of the agency. The Committee has also identified research that could benefit from opportunities for collaborations with similar activities under way outside the U.S.

The long–term, sustained research at the NRC is discussed in Chapter 3. In that Chapter, the ACRS suggests that in addition to the work on specific technical issues that are anticipated to arise, a fraction of the NRC research program be devoted to developing a regulatory infrastructure for much more efficient and effective regulatory work by line organizations at NRC in 10 to 20 years.

2. GENERAL OBSERVATIONS AND RECOMMENDATIONS

In this Chapter, the ACRS highlights major components of the ongoing research activities dealing with the safety of nuclear power plants and presents its recommendations concerning these activities. Some notable generic aspects of the research activities are:

- NRC has succeeded over the last few years in its effort to tie research activities it undertakes to near-term issues being confronted by the NRC line organizations (NRO, NRR, NMSS, NSIR). About 67% of research activities support specific needs of these Offices. About 28% are mandated programs such as the Accident Sequence Precursor (ASP) program. The remainder is administrative activities. An especially useful innovation adopted in some major research activities has been the organization of Technical Advisory Groups. Members of these advisory groups consist of staff from the line organizations. Often, the close interactions of the Technical Advisory Groups with the ongoing research activities have positive impact on both the scope and the pertinence of the research.

- Current NRC research activities are positioned well to address most of the research areas identified in a recent Strategic Plan for Light Water Reactor (LWR) Research and Development prepared for the Department of Energy (DOE) and the nuclear industry, including:

  - Greater use of risk information to improve safety

  - Support the development of a regulatory process for deployment of digital instrumentation and control technology

  - Improve understanding of materials degradation and plant aging to support license extension to 80 years

  NRC research will have to confront a fourth thrust identified in the DOE/nuclear industry Strategic Plan – extension of fuel burnup to 85 GWd/t. Research the agency has completed provides a suitable basis for modifying existing regulations to accommodate new fuels and alloy claddings for extended burnup. Research will be needed to search for possible emergence of new fuel physics, cladding degradation, and fuel-clad mechanical and chemical interactions at such high levels of burnup. As noted below, lack of availability of suitable experimental facilities will be a handicap. Building upon the international collaborations developed in the course of confirmatory research for burnup extensions from ~30 GWd/t to the current regulatory limit of 62 GWd/t may be useful.

- Collaborations with other countries are being pursued in many of the research projects. The nature and depth of collaboration varies. In the case of severe accident research, there is very comprehensive collaboration on both experiments and modeling. Similarly, detailed collaboration is taking place in fuels and materials research. In other research areas, the collaborations are simple information exchange. Collaborations have the potential of leveraging resources. More importantly, collaborations expand the intellectual basis for research and provide the NRC staff with a kind of real time peer review of its research plans.

  Representatives of industry have expressed an interest in closer collaboration with NRC, including the joint development of models and computer codes. NRC has a Memorandum of Understanding with the Electric Power Research Institute (EPRI) supporting joint experimentation. The ACRS has been supportive of NRC collaboration with industry on experiments provided that NRC has an effective and early voice in the design of the tests. The ACRS has misgivings about collaboration between industry and the NRC in the development of models or computer codes and the interpretation of test results. The independent development of models has been a significant route to discovery of safety issues and safety margins. The independence of analysis is a crucial element in ensuring public confidence in the regulatory process.

- In the past, few of the research activities undertaken by the agency had detailed and updated research plans describing the objectives of the research as well as both tactics and strategy for meeting these objectives. A notable exception has been the planning of research in the area of Digital Instrumentation and Control Systems. It is noteworthy that several other research activities are beginning to document their research plans. Indeed, there is work under way to document the entire reactor safety research program. We look forward to an opportunity to review this overall research plan as well as plans for individual research activities as they are being developed.

- The continued erosion of experimental capabilities in the U.S. is affecting the NRC's research program. This loss of experimental capabilities is especially acute in the areas of fuel research and neutronics research. Domestic hot cells are aging and in short supply and test reactors suitable for technical resolution of LWR safety issues are scarce. The lack of properly scaled experimental research facilities also affects research into two-phase thermal hydraulics.

- There is a growing reliance on computational methods to deal with physical, chemical, and neutronics issues affecting the safety of nuclear power plants. Associated with this growth is a demand to have a quantitative understanding of the uncertainties associated with computational results. Many types of uncertainty may affect results of computation, including uncertainties in parameters, uncertainties in the models, and uncertainties in the completeness of the analysis. Techniques have been developed for the quantitative description of parametric uncertainties such as Monte Carlo sampling and Limited Latin Hypercube (LHS) sampling. Such methods should be included in computer codes being developed by the agency.

- The agency is vulnerable to the loss of institutional knowledge in certain technical areas where it is critical to maintain competency and even excellence. The ACRS calls attention to areas where this vulnerability is being addressed well such as neutronics research. New staff members are brought into the field and trained while more experienced staff members are available for consultation. This method is superior to simple exit interviews and like for the preservation of institutional knowledge and technical capabilities.

Major observations, conclusions, and recommendations concerning specific research activities are summarized below. Additional details on the research activities in the various technical disciplines are provided in Chapters 4 through 15.

Digital Instrumentation and Control

The use of software-based digital safety systems in nuclear plants is inevitable. Demonstration of the reliability of these systems is a challenge that merits research. A well founded research plan is in place. But, it must be recognized that the issue may not be amenable to complete technical resolution. The feasibility of technical resolution must be demonstrated by generating research products useful to the relevant line organizations over the course of the current five-year research plan.

Fire Safety

The agency has a much revitalized fire safety research effort to prepare the line organizations for evaluating licensees' transition to the risk-informed, performance-based fire protection programs that meet the requirements of 10 CFR 50.48(c), and the referenced 2001 Edition of the National Fire Protection Association (NFPA) Standard, NFPA 805. The NFPA 805 Standard requires that licensees use only fire models that are acceptable to the NRC. This Standard further requires that the fire models be verified and validated and the fire models only be applied within their limitations. Significant progress has been made in the verification and validation of fire models. Still, NRC does not have the phenomenological analysis capabilities to support realistic assessment of fire risk associated with operational events. The NRC needs to develop acceptable fire-effects models, including models of smoke transport within plants and its impacts on plant safety.

Reactor Fuel

Research has been completed on the effects of burnup on fuel and cladding behaviour under design basis accident conditions. Results of this research can be used to modify current regulations to better address new fuels and claddings that will be introduced by the industry to extend fuel burnup. As more reactors begin operation at extended power uprate, there is a need for the NRC to develop and apply analytical codes to evaluate the effects of pellet-cladding mechanical and chemical interaction on fuel integrity during abnormal operating conditions. NRC needs to maintain expertise in the area of reactor fuel and fuel cladding since further developments in this field can be expected. NRC may need to develop further its expertise in safety of fuel reprocessing facilities since such facilities may be the first products of the DOE's Global Nuclear Energy Partnership (GNEP) to be submitted for licensing.

Neutronics Analysis, Core Physics, and Criticality Safety

NRC has adequate computational capabilities to meet many current needs for neutronics analysis, core physics, and criticality safety analysis. These capabilities have been demonstrated in the identification of reactivity transients associated with "checkerboard" voiding in the Advanced CANDU Reactor (ACR) -700. There is an adequate program to maintain expertise in these areas that must constitute a core competency of the agency. However, available computational and analysis methods are being challenged by the complexity of advanced LWR cores such as the core of the Economic Simplified Boiling Water Reactor (ESBWR). Further upgrades to the computer codes to address better these complexities are needed if the capability to perform independent assessments is to be maintained. More extensive improvements to the existing computational capabilities will be needed to address advanced reactors that use high temperature gas or liquid metal coolants and involve neutron spectra with larger epithermal and fast components. Validation of the modeling is a crucial issue. Much of the validation must now be done with legacy data. Participation in international benchmark activities certainly is some compensation for the lack of active experimental facilities in this area. Collaborations on modern validation experiments will be useful if major upgrades to computational capabilities are found necessary.

Human Factors and Human Reliability

Human factors and human reliability research continues to generate results useful to the regulatory organizations. An important benchmark of the many human reliability models is being planned. Results of this test may help define future directions of human reliability research at NRC.

Materials and Metallurgy

Based on the number and visibility of programs, materials and metallurgy is the most active area of research within the NRC. This is appropriate in light of the continuing emergence of crucial and unexpected materials degradation in aging LWRs, including the stress corrosion cracking of control rod drive mechanism nozzles at Davis-Besse and dissimilar metal weld issues that have arisen at a number of plants. Research programs are under way in five areas:

- Environmentally Assisted Cracking in LWRs
- Steam Generator Tube Integrity
- Non-destructive Examinations
- Proactive Materials Degradation Assessment
- Reactor Pressure Vessel Integrity

All of the research activities seem well founded. The studies of reactor pressure vessel integrity are enhancing the capabilities at the agency for probabilistic fracture mechanics analysis to support more realistic assessments of the behaviour of aging pressure vessels. The ACRS is especially pleased with the proactive materials degradation assessment effort to identify degradation mechanisms and vulnerable components before phenomena become manifest in plants. The ACRS is also pleased with the work being done to compare various methods for non-destructive examination (NDE) of plant components and answer the questions being raised about the reliability of visual inspections of vulnerable components.

Operational Experience

The scope and the visibility of work to analyze and evaluate operational data have dwindled dramatically since the function of then Office for Analysis and Evaluation of Operational Data (AEOD) was transferred to RES. Aside from mandated activities such as industry trends analysis and the Accident Sequence Precursor (ASP) program, the work is mostly in support of other research efforts and particular licensing efforts. No longer are publicly available, in-depth analyses of particular systems or trends being produced. The resources devoted to this activity are approaching the minimum needed to sustain a distinct work unit within the research at NRC.

Probabilistic Risk Assessment

Probabilistic Risk Assessment (PRA) research has been focused on the applications of the available technology to particular regulatory issues especially to the Reactor Oversight Process. There has been a very heavy demand for this type of support from the line organizations. There is a growing effort within the larger PRA community to improve software. NRC needs to devote some of its PRA research resources to the development of next generation risk analysis software to improve the accuracy of risk assessment and to facilitate NRC review of risk-informed applications from licensees.

Seismic and Structural Engineering

There is a resurgence of interest in the threats posed by seismic events to new power plants. The interest is driven by early site permit applications, new design certifications, and combined license (COL) applications as well as re-evaluations of seismic hazards especially in the central and eastern U.S. Further impetus for seismic research has been driven by interest in the performance-based seismic hazard standard issued by the American Society of Civil Engineers and used in the Clinton early site permit application. The research program has established a useful collaboration with Japan. An excellent study of threats from tsunamis, produced by undersea landslides, on the eastern U.S. has been completed in cooperation with the U.S. Geological Survey. Other elements of the program are well founded and will generate products useful to the line organizations at NRC.

Severe Accidents

The ACRS supports the strategy developed by the NRC staff to support regulatory decisions associated with severe accidents. This strategy involves development of computer codes and analysis and evaluation of experimental data. This approach can successfully maintain and update the modeling capabilities for severe accident analyses. The NRC approach of leveraging resources through international experimental collaborations is particularly notable. The planned program extensions and continuations of these collaborations are well worth the investment.

Thermal Hydraulics

The thermal-hydraulics research at NRC is primarily focused on the development of the TRACE computer code. TRACE development is progressing well. The ACRS looks forward to hearing results of the ongoing peer review for TRACE.

There is a continuing and even growing need to incorporate TRACE into the regulatory process to deal with power uprates, and greater use of best-estimate analyses in licensing actions and new reactor design certifications. It is important that TRACE be brought into the regulatory process as expeditiously as possible. The resources for sustained code development of TRACE appear minimal.

Experimental facilities to support and validate TRACE are aging. They do not scale well for validation of passive plant responses that are central to many of the new LWR designs being submitted for certification. NRC needs to develop a strategy for providing high-quality experimental data both to validate TRACE predictions and to support certification of advanced LWR designs.

The use of computational fluid dynamics (CFD) methods in licensee applications will continue to increase. NRC must be capable of reviewing and independently validating licensee proposals using these computational methods. NRC may be better served by joining international efforts to develop open source CFD computer codes rather than relying solely on commercial CFD codes.

Advanced Non-LWR Designs

The possibility of the need to certify or license advanced reactors using gas-cooling or liquid metal-cooling rather than conventional light-water cooling pose a conundrum for NRC and the planning of its research efforts. NRC has some experience with these alternative reactor technologies, but does not have in place the technical infrastructure that would be needed to support an effective certification review of a reactor design based on such technologies. The Office of New Reactors (NRO) is dubious of the feasibility of developing this technical infrastructure and conducting research in parallel with a certification review. The ACRS certainly agrees with the NRO on this point. It is clear that it will take some time to develop the needed technical infrastructure at NRC. What is not clear is when applicants might be able to submit a certification application with an adequate technical basis. This makes it difficult to assign priority and allocate resources for advanced non-LWR research.

The research staff has systematically identified and prioritized the technical issues associated with the certification of a gas-cooled reactor. This has been done based on an expert opinion elicitation process that identified important phenomena and ranked these phenomena with respect to safety importance and the current level of understanding. The results of the phenomena identification and ranking will help to communicate expectations for the technical information needed for a defensible certification application for a gas-cooled reactor. Especially crucial will be use of the results to define areas where analyses must be substantiated by pertinent, prototypical experimental results.

NRC should initiate activities to assess the state-of-knowledge in other advanced reactor technologies before there is further loss of the limited capability available (worldwide) in some of these technologies. This effort can provide a significant input to the development of safety requirements for some of these advanced designs. Multinational cooperation at an early stage could lead to common safety standards and a sound and more cost-effective research program

3. LONG - TERM, SUSTAINED RESEARCH AT NRC

The research program that NRC sponsors is now very closely coupled to the immediate needs of the line organizations processing licensing applications and other regulatory matters. The assessments presented in this report, overall, are indicative of a positive appraisal of both the utility and the quality of the applied research now being sponsored by NRC. In its previous biennial reports on review and evaluation of the NRC safety research program, the ACRS noted the need for long-term research not tied to the near-term issues of the regulatory process. As directed by the Commission, the staff has undertaken some efforts to identify longer-term research. The development of a long-term research plan is a considerable departure from the staff's focus in recent years on immediate regulatory needs. The ACRS has previously commented on several of the specific long-term research activities identified by the staff. By in large, the topics in the long-term research plan developed by the staff address regulatory issues that will have to be confronted eventually.

In this Chapter, the ACRS addresses the scope of longer-term research that NRC needs to consider. The Committee believes that the primary role of long-term research should be to address the technical capabilities of the agency and the way the agency conducts its regulatory and safety mission. The time horizon of interest is about 10 to 20 years. Two important themes fall within this category of long-term research. One is maintaining an infrastructure of technical expertise, facilities, and analytical capabilities. The second deals with the development of an infrastructure for much more efficient and effective regulatory work by line organizations at NRC in 10 to 20 years.

The ACRS views on the scope of the long-term research the agency needs to consider are based on a number of assumptions that the Committee has made about the future:

- Use of nuclear power will grow in the U.S. over the next twenty years.
- New reactors in the U.S. will be based on LWR technology; Sodium-cooled and gas-cooled reactors will become a part of the NRC's regulatory workload.
- NRC staffing will not grow commensurate with the growth in nuclear power.
- The average years of experience and education of NRC staff will decrease over the next ten years and then stabilize.
- Licensing and regulatory actions per plant will be about the same in the future as it is now.
- Licensee submittals will grow more complicated and technically sophisticated as licensees continue to utilize the margin that exists between current operations and regulatory limits.
- The support vendors and suppliers for the nuclear industry will grow more international in character than in the past.
- NRC will continue to risk inform its regulations and regulatory processes to achieve a greater focus on issues that are important to safety. Greater use of realistic rather than demonstrably conservative safety analyses will be encouraged.
- Fuel reprocessing and use of mixed oxide (MOX) fuel in nuclear power plants will become more economically and politically attractive.

Maintaining an infrastructure of Technical Expertise, Facilities, and Analytical Capabilities

The ACRS supports maintaining technical expertise at the agency in areas where there is limited availability of independent expertise from the academic community and the private sector. Technical expertise in these disciplines (neutronics, fuel, PRA, fission product chemistry, etc.) is needed by the agency to make technically sound regulatory decisions without imposing conservatisms that are actually unnecessary when the state-of-the-art is examined. The ACRS also calls attention to needs for access to experimental facilities and state-of-the-art independent analyses tools for the regulatory process. The state-of-the-art in science and technology is rapidly advancing. It is inevitable that some of these technologies will also be utilized by the existing fleet of reactors to improve economics and safety. These new technologies would include NDE to monitor for impending failure, advanced safety analysis tools (e.g., Multidimensional CFD codes), next generation of PRA software, innovative materials, development of high-burnup fuels (85 GWd/t and enrichment beyond 5%), etc. Although in many cases introduction of these new technologies can increase safety and efficiency, the use of new technologies can also introduce new failure modes. NRC should assess its research plans and develop a long-term program to ensure that the agency is prepared to make sound science-based safety decisions.

The topic of maintaining the infrastructure of technical expertise, facilities, and analytical capabilities to support regulatory decisions is also addressed in Chapter 2 and in the discussions of the various technical disciplines of research in Chapters 4 through 15. The proposed long-term research topics include:

- *Technical Infrastructure to Respond to Innovative Fuels*

  There is widespread expectation that the nuclear industry may push for extending burnup limits to 85 GWd/t. Certainly, a recent research strategy document prepared for DOE proposes that fuel burnups be extended to 85 GWd/t. In addition, innovative designs utilizing advanced fuel pellet and cladding materials are in the LWR development pipeline. The nuclear industry program for development of high-burnup fuel is expected to take about 10 years and involves innovative fuels with uranium enrichment above 5%.

  NRC must develop and maintain expertise in the area of reactor fuel to respond to these proposed fuels and claddings. The challenge in maintaining what amounts to an essential core competency of the agency arises because of the limited availability of expertise outside of the agency that is independent of licensees. Current manpower working for the agency in this field either directly or by contract is experienced and there is a need to groom newer generations in the field. A major impediment in meeting this need is the decline in this Country of in-pile test facilities and hot cells for examinations of irradiated fuels and cladding. Collaborations with international partners having these facilities and capable of undertaking pertinent studies will be essential for NRC to maintain an adequate level of expertise in reactor fuels.

- *Full-Height, Integrated, Scaled Test Facilities for Passive Systems*

  Based on the expected number of COL applications in the next few years, it is very likely that in 10 to 20 years some of the operating reactors would be of the advanced passive LWR designs such as AP1000 and ESBWR. It is perhaps not unlikely that some unanticipated challenges may arise with operating experience in such plants. To better understand the safety impacts of events and to maintain a high level of technical expertise, NRC should consider the value of having access to full-height, integrated, scaled test facilities that can simulate reactors with passive systems. The understanding and correlations developed in full-height test facilities can be applied to reactor systems more reliably, since they do not depend on scaling analyses required for reduced--height systems, which are difficult to develop and validate for two-phase flows. NRC should explore potential international interest in maintaining such facilities.

- *Technical Infrastructure to Support the Independent Safety Evaluation of non-LWRs, IRIS, and other Unique Designs*

  Advanced non-LWRs, IRIS, and other unique designs pose a challenge for NRC and the planning of its research efforts. The non-LWRs of interest to NRC can be categorized as liquid metal-cooled reactors (LMRs) and high temperature gas-cooled reactors (HTGRs). The Toshiba 4S (Super Safe, Small, and Simple) is a LMR. In addition, LMR technology is also being considered for the Advanced Burner Reactor (ABR) proposed under the DOE GNEP initiative. The Next Generation Nuclear Plant (NGNP) prototype will also utilize HTGR technology when the concept is developed. NRC has some experience with these alternative reactor technologies, but does not have in place the technical infrastructure that will be needed to support an effective certification review of a reactor based on such technologies.

  As noted previously, the staff has systematically identified and prioritized the technical issues associated with the certification of a gas-cooled reactor based on an expert opinion elicitation process that identified important phenomena and ranked these phenomena with respect to safety importance and the current level of understanding.

  NRC should initiate activities to assess the state of knowledge in other technologies before there is further loss of limited capability available (worldwide) in some of these technologies. The state-of-knowledge report should identify significant gaps in knowledge that might be necessary for future licensing of some of these unique designs. Specific focus should be given to understanding severe accident behavior and crucial areas of safety (e.g., reactor fuel, high-temperature materials). This effort can provide a significant input to the development of safety requirements for some of these designs. Multinational cooperation at an early stage could lead to common safety standards and a sound and more cost-effective research program

- *Technical Infrastructure to Support Safety Evaluation of Fuel Reprocessing Facilities*

  Reprocessing of nuclear fuels may be an emerging technology that merits research attention by NRC. Aqueous reprocessing of irradiated fuels has been done for many years within the nuclear weapons community using the PUREX process. Though familiar, this process has not been trouble free and there are many known hazards.

NRC is gaining some exposure to the associated safety issues of “red oil”, hydroxylamine nitrate, ammonium nitrate and the like through its review of the construction authorization application for the DOE’s MOX Fuel Fabrication Facility.

Aqueous reprocessing of spent reactor fuels is widely thought to be the first technology that will emerge for licensing from the DOE’s GNEP initiative.

Modifications of the PUREX process – the so-called UREX processes – are being considered by DOE for its GNEP initiative. Undoubtedly, safety issues similar to those of the PUREX process and new safety issues will arise in these new processes.

Even more challenging will be the pyrometallurgical reprocessing of spent nuclear fuel also being considered by DOE as part the longer-term plans of its GNEP initiative. The large-scale implementation of pyrometallurgical reprocessing of commercial fuels has never been undertaken in the past. NRC will need to develop expertise in the safety of fuel reprocessing technologies well before licensing applications involving use of such technologies are submitted for review.

- *Technical Infrastructure to Support Materials Degradation Assessment*

Long-term research on materials degradation will be essential to anticipate and address known and emerging issues as the Nation’s LWRs extend their operation for 60 years or longer.

Research will be needed to confirm that the various materials, fabrication, and water chemistry changes that have been implemented in LWRs to mitigate known environmentally assisted degradation mechanisms will be effective for the extended life of the plants.

- *Technical Infrastructure for Independent Assessment of Multidimensional CFD Analyses*

The ACRS anticipates that licensees will increasingly employ multi-dimensional CFD capability to resolve problems which the current generation of thermal-hydraulic codes, such as TRACE, are unable to do. The NRC has limited confirmatory capability to check such submittals. Such capabilities will also undoubtedly be needed for advanced reactors, such as HTGRs, and for evaluation of new fuel designs.

The NRC currently has very little effort in the area of CFD other than application of commercial CFD codes. Although these are used in the process industry for qualitative indications of phenomena, they are validated to a much less rigorous standard than codes for nuclear use, and the source codes are not available. They appear to include a number of *ad hoc* fixes to improve stability and robustness which may affect their predictive capability for situations that cannot be studied experimentally.

The international nuclear community, on the other hand, has instituted the development of multidimensional CFD capabilities. CFD codes are being developed and validated to standards of reliability and accuracy required for use in evaluations of nuclear systems. NRC should join in this effort and provide adequate resources to allow productive participation.

- *Next Generation of PRA Software*

  The agency's main PRA tool is the SAPHIRE code that has been developed in the last twenty years and is the basis for the Standardized Plant Analysis Risk (SPAR) models that are used extensively in the Reactor Oversight Process. The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment was briefed recently by representatives from EPRI and an international group of researchers that have formed the Open PSA Initiative. Two issues with the present state of PRA software were identified. First, it is hard to check the completeness and correctness of the logic of the PRA models produced using current PRA codes because these models are code dependent. Second, there is disturbing evidence that, under certain circumstances, the probabilistic calculations may be flawed.

  There are efforts under way to develop a standard PRA representation format that would promote independence between the logic models and the individual codes. Such a standard would facilitate significantly the review of the PRA logic, a benefit of great value to the agency. The staff should participate actively in the international activities related to the development of the next generation of PRA software.

  In its 2006 report to the Commission on the NRC Safety Research Program (NUREG-1635, Vol. 7), the ACRS stated that "the staff needs to review the literature concerning Binary Decision Diagrams and evaluate the need to adopt this technology." The ACRS stated further that "the growing importance of the SAPHIRE code and the SPAR models in the regulatory process warrants such an investigation." The ACRS continues to believe that the staff should undertake such an investigation and participate actively in the international activities related to the development of the next generation of PRA software. A short-term product could be an evaluation of the potential impact of the new software on regulatory decisionmaking. The results of such an evaluation would guide the establishment of a longer-term research project in this area.

Development of a Regulatory Infrastructure for Much More Efficient and Effective Regulatory Work

The second theme within the general topic of long-term research at NRC deals with the productivity of the line organizations handling regulatory issues and the growth of nuclear power in the U.S.

The growth of nuclear power seems inevitable at this juncture based on applications for early site permits and the recent submittal of several COL applications. This presumes, however, that no major reactor accident occurs in this Country or pertinent accidents occur in other countries with well developed nuclear energy capabilities. A focus on LWR technologies also seems likely, since alternative technologies have not yet been demonstrated adequately and several modern LWR designs have been certified. More certifications of advanced LWRs are anticipated. However, the NRC will likely need to develop safety requirements and an infrastructure to deal with safety matters relating to certain non-LWR designs (e.g., LMR, HTGR) and unique LWR designs (e.g., IRIS). Constraints on the growth of the size of the NRC staff seem likely based on the political challenges to growth. Loss of experience within the staff is simply a reality of demographics. The seriousness of this loss will depend, in a significant measure, on the competition for manpower by the nuclear industry itself. Stabilization of the staff demographics may be optimistic. Even if the loss of experience is not realized, the staff will have to confront more numerous and more complex licensing actions. Growing technical sophistication and complexity of licensee submittals seems likely since the "easier" changes to utilize margin between operations and regulatory limits are being made now by the licensees. Other assumptions such as the concern over the growing use of international vendors and suppliers as well as the development of reprocessing are extrapolations of current trends.

Together, the assumptions lead to the unavoidable conclusion that the NRC staff will have to be both more productive and more technically sophisticated in the future. Risk-informing the regulatory processes can relieve some of the burdens on the staff, but increasing use of realistic analyses in the place of demonstrably bounding analyses for risk-important regulatory actions imposes additional, different burdens.

To achieve greater productivity in line organizations at NRC, the ACRS sees no alternative to the provision of superior technology to the staff in these organizations. Useful, superior technology will have to come from NRC research. But, this technology will have to be useful to the staff within line organizations that may or may not have in-depth familiarity with the bases of the technology. That is, the research will have to yield computational and simulation technology that can be routinely used without the close assistance of the developers of the technical capabilities. The ACRS foresees a time when individual members of NRC staff can access routinely such things as:

- Comprehensive PRA for individual plants that addresses all modes of operations and all initialing events.

- Comprehensive neutronics and thermal hydraulic analyses of particular nuclear steam supply systems.

- Simulations of control rooms of particular nuclear power plants.

- Expert systems that replace the current Standard Review Plan.

- Corrosion analyses, including stress corrosion cracking and probabilistic fracture mechanics of reactor pressure boundary materials and components

- Virtual reality simulations of particular plants.

- Finite element structural models.

- Fire-effects models of particular structures, systems, and components

- Human reliability analyses

- Intrusion simulations

In addition, the ACRS foresees the availability to individual staff within line organizations of "user friendly", searchable databases dealing with such things as:

- Operational events
- Component reliability data
- Failure modes and effects analyses of systems, structures, and components
- Seismic hazards
- Weather at licensee sites

The specific computational and simulation capabilities that will be needed to facilitate greater staff productivity in the processing of regulatory and the licensee applications are not the focus of ACRS. What the ACRS recommends is that some fraction of the NRC research effort be devoted to the development of tools, databases, and decision-support capabilities that will be needed in the future. A first step that might be taken in such research is a systematic examination of line organization activities to identify areas where new information technologies can be used to facilitate regulatory review. A second step might be the review of existing computational, simulation and database resources to determine how these resources might be made more “user friendly”, robust, and less dependent on technical support. A third step might be the development of strategies not to just preserve knowledge but to make the knowledge broadly available throughout the agency.

The long-term research that the agency needs to undertake to prepare itself for a future that makes vastly more use of information technology will have to be centered within RES. The research cannot be done, however, independent of the line organizations. RES has proved itself many times capable of coordinating well multidisciplinary research projects through its Action Plan structure. The ACRS feels such an organization of the research with involvement of NRR, NRO, NMSS, NSIR, and regional personnel could be effective. In any case, the communications between the research effort and the line organizations must be at the staff level if effective products welcome by the users are to be produced.

4. DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

Three issues bedevil regulations and the inevitable transition from analog to software-based, digital instrumentation and control technology for reactor safety systems. First, the development of methods for the comprehensive testing of the reliability of digital systems has not reached to a level of maturity that allows them to be used in the regulatory process. It may not even be possible to comprehensively test the reliability of such systems. Consequently, the industry (IEEE) standards that are endorsed by the NRC focus on the process of design and development of software-based digital systems. The assumption is, of course, that a good controlled production process will lead to a highly reliable product. The quality of the product is then very dependent on the quality of the requirements established for the system. The focus on process leads to the second challenge faced by NRC in dealing with digital systems. Process monitoring and reviews are very manpower intensive. They can be the limiting step in a system review. Furthermore, the focus on the development process frustrates the licensees' desire to use commercial off-the-shelf digital systems for safety functions in nuclear power plants. The third challenge faced by the NRC is how to portray the reliability of digital systems in PRAs.

Digital systems of interest to the NRC are safety systems such as the reactor protection system. By in large, the safety systems involve simple actuations and do not involve complex feedback and control functions. But, software-based digital systems offer the designer much greater flexibility and functionality than analog systems. In taking advantage of the potential of digital systems, the designer creates complexity in the circuitry, logic, and software. Interconnections among safety and control systems, some of

A Highly Integrated Control Room

which are subtle and even unintentional, create vulnerabilities to common-cause failures that are not random in nature. To a very real extent, the difficulties associated with the safety review of digital systems are burdens the nuclear industry is placing upon itself.

NRC has been well aware of all of these challenges for some time. A detailed research plan[1] has been developed to confront these issues. The staff has formed a Steering Committee consisting of senior managers to provide oversight and guidance on six key technical and regulatory issues and to interface with the industry:

- Cyber security
- Diversity and Defense In Depth
- Risk-informing Digital Instrumentation and Control
- Highly Integrated Control Room – Communications
- Highly Integrated Control Room – Human Factors
- Licensing Process Issues

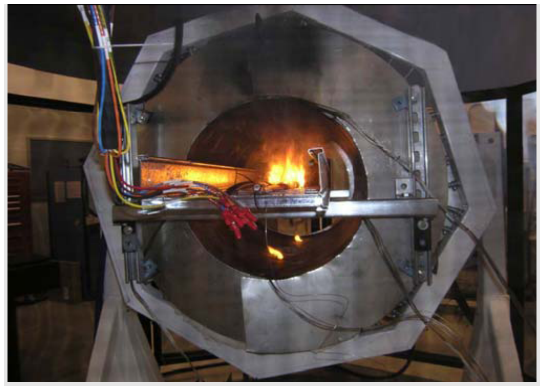
[1] See NRC Digital System Research Plan FY2005-2009

Each key issue is assigned to a Task Working Group that reports to the Steering Committee. The staff has identified specific deliverables and due dates for these deliverables.

The ACRS has been meeting frequently with the staff and providing reports to the Commission on progress in these issues on a regular basis. In general, the ACRS agrees with the staff's approach to the development of a process that will facilitate the deployment of digital instrumentation and control technology for new and operating reactors.

5. FIRE SAFETY

The fire safety research program is focused on the NRC's regulatory needs as it prepares the line organizations for evaluating licensees' transition to the risk-informed, performance-based fire protection programs that meet the requirements of 10 CFR 50.48(c), and the referenced 2001 Edition of the National Fire Protection Association (NFPA) Standard, NFPA-805. The NFPA-805 Standard requires that licensees use only fire models that are acceptable to the NRC. This Standard further requires that the fire models be verified and validated and the fire models only be applied within their limitations. There is a close coupling between the ongoing research programs and the regulatory needs.



The "small-scale" (Penlight) radiant heating test facility at SNL, shown testing two multi-conductor cables for the US NRC's CAROLFIRE program

Research projects within the program can be grouped into three technical areas:

- Fire Risk Assessment
- Fire Modeling
- Fire Testing

Each of these technical areas is discussed below.

FIRE RISK ASSESSMENT

In 2004, the NRC amended its fire protection requirements in 10 CFR 50.48(c) to permit existing reactor licensees to voluntarily adopt and maintain a fire protection program that meets the requirements of the 2001 Edition of the NFPA-805 Standard as an alternative to the existing deterministic fire protection requirements. As of January 2008, 27 sites involving 42 nuclear power plants have announced their intent to transition to NFPA-805 requirements. Full scope fire risk assessments will be performed for each of the transitioning nuclear power plant.

The NRC research efforts in the fire risk assessment area are aimed at developing standards and associated guidance to assess the quality of such fire risk assessments and providing inspectors with the tools and knowledge necessary to assess the validity of the fire protection licensing bases for the plants.

During the past few years, RES in cooperation with EPRI has taken some important steps to consolidate the fire PRA research and development activities. This effort has led to the publication of NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Plants," which provides a structured framework for the overall fire risk assessment, along with recommended practices to address specific aspects of the analysis. Current RES activities in the fire risk assessment area include support for projects aimed at:

- Development of ANS standards for fire PRAs

- Supporting NRR in the implementation of 10 CFR 50.48(c) through participation in NFPA-805 pilot plant visits.

- Development and delivery, in collaboration with EPRI, of training on NUREG/CR-6850 and conducting User's Group meetings to provide joint interpretations of fire PRA issues such as frequently asked questions (FAQs) as well as issues arising from non-pilot-plant application of NUREG/CR-6850.

FIRE MODELING

Fire models are the phenomenological basis for fire risk analysis. 10 CFR 50.48(c), and the referenced 2001 NFPA 805 Standard requires that "only fire models that are acceptable to the authority having jurisdiction shall be used in fire modeling calculations." NFPA-805 Standard also requires that the fire models be verified and validated, and be applied only within their domains of validity. These requirements of NFPA-805 create, then, the agency need to review, assess, and validate fire models.

RES and EPRI sponsored a collaborative project for verification and validation (V&V) of selected fire models used by the nuclear industry. The results of this project have been documented in NUREG-1824 (EPRI 1011999) "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." This research fulfills NRR's current needs in this area, inasmuch as it provides the necessary V&V for implementing NFPA-805 Standard and supports the credibility of fire models used in the SDP.

NRC does not have adequate models of fire effects, including the effects of smoke that may be transported and deposited substantial distances from the fire locations. A first step taken by the staff in the development of fire effects modeling has been the conduct of a Phenomena Identification and Ranking Table (PIRT) process. This process is used to identify the important phenomena that arise during fires and judge whether current fire models adequately simulate or characterize these phenomena. Future work will also develop probability distributions for the uncertainty associated with each model; these distributions will be useful for the fire risk assessments.

FIRE TESTING

Confirmatory testing is a critical element of the fire safety research program. The ability to determine risk due to fire damage of instrument, control, and power cables in nuclear power plants has been a concern for many years. The issue of cable hot-shorts (including spurious actuation) has been a source of uncertainty for the NRC and licensees. In order to better understand this issue, Nuclear Energy Institute (NEI) and EPRI jointly conducted a series of cable fire damage tests that were witnessed by the NRC staff in 2001. Data from these tests, as well data from previous tests available in literature, led the NRC to issue Regulatory Issue Summary (RIS) 2004-003, Revision 1, "Risk-Informed Approach for Post-Fire Safe Shutdown Circuit Inspections." This RIS provides guidance for NRC inspectors in deciding which causes of fire-induced hot-shorts are important to safety and should be considered during inspections. The RIS specifically describes four categories of concerns that should be considered during inspections (called 'Bin-1 items'). Other issues whose importance still needed to be determined were also described in the RIS and are referred to as 'Bin-2 items'. These included inter-cable shorting for thermoset cables, inter-cable shorting between thermo-plastic and thermo-set cables, configurations involving three or more cables, multiple spurious actuations in control circuits, and prolonged fire-induced hot-shorts that could impair the ability of a plant to achieve hot shutdown. NRC identified the need for empirical testing to provide additional data in certain cable configurations to support further development of guidance and modeling capabilities.

To improve understanding of the Bin-2 items, tests were conducted in 2006 and 2007 to examine cable response to live fire. Both small-scale and intermediate-scale open-burn fire tests were conducted. The test results are documented in two volumes of a draft report (NUREG/CR-6931) prepared by Sandia National Laboratories (SNL). The first volume, “Cable Response to Live Fire (CAROLFIRE) Volume 1: General Test Descriptions and the Analysis of Circuit Response Data,” contains the small-scale and intermediate-scale test results for electric circuit failures (i.e., "hot shorts" data). The second volume, “Cable Response to Live Fire (CAROLFIRE) Volume 2: Cable Fire Response Data for Fire Model Improvement,” contains thermal test data from these same experiments aimed at improving cable fire response models. As a part of CAROLFIRE project, a simple model was developed to predict thermally induced electrical failure when a given interior region of the cable reaches an empirically determined threshold temperature. The description and validation of the Thermally Induced Electrical Failure (THIEF) model are reported in volume 3 of NUREG/CR- 6931, “Cable Response to Live Fire (CAROLFIRE) Volume 3: Modeling.”

Efforts in the fire safety research area include the support of fire incident records exchange. Data from fire events at foreign nuclear power plants are collected and analyzed to provide insights into their causes, means of prevention and consequence mitigation, and their applicability to U.S. power plants. The effort also provides guidance on the quantification of frequencies based on experience gained from pilot application of NUREG/CR-6850.

6. REACTOR FUEL

For nuclear reactors, the fuel integrity is one of the most important safety considerations. The fuel and its metallic cladding are the first barriers in the defense in depth against inadvertent release of radionuclides.

Oxidation of a foreign niobium-bearing cladding alloy under LOCA conditions

Over the last 25 years, nuclear reactor fuel has evolved from a specialty product of individual, high-technology vendors to a commodity product. Indeed, it is not uncommon now for commercial reactors to use fuel from different suppliers. The commodity nature of fuel has affected research on fuel behavior especially in off-normal circumstances. Vendor focus is now on incremental operational improvements that have the potential to capture larger fractions of market share. This focus is especially evident in the development of new cladding alloys such as the M5 alloy in which niobium replaces tin as the main alloying agent. Small alloying changes can dramatically affect cladding performance under accident conditions as is evident in comparing the performance of the M5 alloy and the nominally same composition alloy, E110.

The suitability of new fuel-clad systems for use in power reactors is assessed typically by analyses and results obtained with lead-test assemblies located within normally operating reactor cores. Seldom do either vendors or reactor operators validate predictions of fuel performance during design basis accident conditions with experimental data. Experimental data on the performance of fuel and cladding systems under severe accident conditions are never provided by licensees. Severe accidents, of course, pose the bulk of the risk to the public health and safety. Experimental data obtained in research sponsored by the NRC have shown that cladding interactions with the fuel plays a central role in the rate and extent of core degradation and fission product release under severe accident conditions.

The NRC staff and its contractors are in the process of completing experimental studies of high-burnup fuel and cladding behavior under design basis accident conditions. The staff has carried out in-pile tests of fuel behavior during reactivity insertion events and out-of-pile tests of fuel behavior under design basis loss of coolant conditions. These experimental investigations have been conducted using an impressive combination of national and international collaborations. Results of the research have led to well considered proposals for changes in the regulations.

The proposed changes would make the regulations more realistic and could decrease burden on both the staff and licensees, especially as new fuel and cladding systems are proposed. Implementation of these proposed changes to the regulations has been slow. Acceptance of the changes to the regulations now awaits results of additional testing that could be regarded as confirmatory in nature.

The staff has also completed revisions of its fuel performance computer codes FRAPCON and FRAPTRAN. These computer codes are used to independently confirm analyses done by the vendors and other licensees. The modifications allow the computer codes to be used to evaluate fuels taken to burnups of up to 62 GWd/t and to evaluate the performance of MOX (plutonia-urania) fuels. MOX fuels will be used in the Catawba reactor as part of a DOE program to isotopically dilute excess weapons-grade plutonium. (The NRC staff is also examining the severe accident behavior of high-burnup and MOX fuels in its severe accident research program. See chapter 13 on Severe Accidents.)

The upgraded fuel performance models do appear to meet most, near-term agency needs. However, the staff does not have the analytical capability to assess the risk of Pellet Cladding Interaction (PCI) fuel failures during anticipated operational occurrences. This fuel failure mechanism can compromise fuel integrity as plants continue to uprate power, increase fuel burnup, and introduce new fuel designs. The staff should develop its analytic capabilities in the near term to assess the fuel vulnerability to PCI during operational transients. There is a wealth of experimental data on the PCI phenomenon that can be used to develop and validate a PCI failure model.

Advances in simulation under way in the national laboratories are making it possible to examine fuel performance in vastly more detail than is done with either FRAPCON or FRAPTRAN. Whether such detail is needed will depend critically on what efforts are made by licensees to extend fuel burnups beyond the current regulatory limit of 62 GWd/t and the amount of experimental data provided to support these proposed changes to regulatory limits. There is widespread expectation that the nuclear industry may push for extending burnup limits to 85 GWd/t. Certainly, a recent research strategy document prepared for DOE proposes that fuel burnups be extended to 85 GWd/t. There appears to be some confidence within the nuclear industry that such extensions of fuel burnup can be done by extrapolating the currently available bases of fuel performance data and models. Emergence of new physics complicated such extrapolations of fuel performance for burnups beyond 40 GWd/t. This necessitated the experimental research on fuel behavior under accident conditions that the NRC is currently completing. It is not evident that no new phenomena will arise in connection with the extrapolation of fuel burnup to 85 GWd/t. Consequently, there will be a continuing need for the agency to independently evaluate the safety of proposed changes in the nature and burnup limits of reactor fuels.

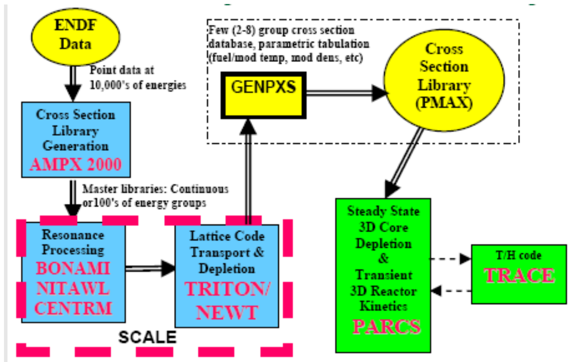
Lead-test assemblies of MOX fuels have recently emerged from their first cycle of irradiation in the Catawba reactor. These MOX fuels are being tested as part of a DOE program to dispose of excess weapons-grade plutonium by using it as reactor fuel. MOX fuels have never before been used in U.S. commercial nuclear power plants though they are being used in foreign reactors. It appears that the staff does not have plans for any research examinations of these novel fuels either after the first cycle of irradiation or after subsequent cycles of irradiation. In light of the limited NRC experience with MOX fuels, the ACRS recommends that there be a research program to follow closely the post-irradiation examination of the lead-test assemblies planned by DOE.

Because of license extensions, power uprates, and the prospect of additional new reactors, it is anticipated that the vendors will introduce new fuel-cladding systems. NRC must develop and maintain expertise in the area of reactor fuel to respond to these proposed fuels and claddings. The challenge in maintaining what amounts to an essential core competency of the agency arises because of the limited availability of expertise outside of the agency that is independent of licensees. Current manpower working for the agency in this field either directly or by contract is experienced and there is a need to groom newer generations in the field. A major difficulty in doing so is the decline in this Country of in-pile test facilities and hot cells for examinations of irradiated fuels and cladding. Collaborations with international partners having these facilities and capable of undertaking pertinent studies may well be essential for NRC to maintain an adequate level of expertise in reactor fuels.

Reprocessing of nuclear fuels may be an emerging technology that merits research attention by NRC. Aqueous reprocessing of irradiated fuels has been done for many years within the nuclear weapons community using the PUREX process. Though familiar, this process has not been trouble free and there are many known hazards. NRC has gained some exposure to the associated safety issues of “red oil”, hydroxylamine nitrate, ammonium nitrate and the like through its review of the construction authorization application for the DOE’s MOX Fuel Fabrication Facility. Modifications of the PUREX process – the so-called UREX processes – are being considered by DOE for its GNEP initiative. Undoubtedly, safety issues similar to those of the PUREX process and new safety issues will arise in these new processes. Even more challenging will be the pyrometallurgical reprocessing of spent nuclear fuel also being considered by DOE as part the longer-term plans of its GNEP initiative. The large-scale implementation of pyrometallurgical reprocessing of commercial fuels has never been undertaken in the world. NRC will need to develop expertise in the safety of fuel reprocessing technologies well before licensing applications involving use of such technologies are submitted for review. Aqueous reprocessing of spent reactor fuels is widely thought to be the first technology that will emerge for licensing from the DOE’s Global Nuclear Energy Partnership initiative.

7. NEUTRONICS ANALYSIS, CORE PHYSICS, AND CRITICALITY

Both the public and other government agencies expect the NRC to have expertise in the fields of neutronics analysis and criticality safety that is at or very near the state-of-the-art. Neutronics analysis is, of course, basic to the safe design and operation of nuclear power plants. Shielding analysis and criticality safety are essential for the safe use of special nuclear materials. Shielding analysis and criticality safety will become more important to the regulatory decisions especially if the Nation elects to move to the processing of spent nuclear fuel.



NRC Nuclear Analysis Codes for Reactor Physics

The fundamental phenomenologies of neutronics, shielding analysis, and criticality safety are well established, particularly for issues that arise in connection with the current fleet of operating nuclear reactors and nuclear facilities. The current NRC research in these fields focuses on treatment of improved data and the development of computational capabilities to address diverse circumstances in greater detail. Examples of issues where improved capabilities have been needed recently include:

- Burnup credit for spent nuclear fuel
- Neutronics of reactor cores with MOX (plutonia and urania) fuels for the isotopic dilution of weapons-grade plutonium
- Nuclear effects of "checkerboard" voiding of the ACR-700 core

The NRC staff has formed effective partnerships with investigators at Oak Ridge National Laboratory and Purdue University to establish high-quality computational capabilities to support agency needs in the areas of neutronics, shielding, and criticality safety. Computational capabilities available to the staff include:

- **AMPX** (A Modular Code System for Processing X-sections)

  The AMPX code system has been upgraded recently to permit fuller use of the recent ENDF/B-VI database as well as European and Japanese databases.

- **SCALE 5** (Standardized Computer Analysis for Licensing Evaluation)

  This modular set of codes includes:

  - ORIGEN-S for radionuclide depletion analysis
  - TRITON (Transport Rigor Implemented with Time-dependent Operation for Neutronic depletion)
  - NEWT (NEW Transport algorithm)

- **PARCS** (Purdue Advanced Reactor Core Simulator)

  Coupled with the TRACE thermal-hydraulics code, PARCS provides threedimensional, multiple group reactor core neutronics analysis.

These computational capabilities appear to be adequate for most of the current agency needs, including core load analyses and the certification of modern LWR designs utilizing fuel to burnup levels less than 62 GWd/t. Though the neutronics analysis codes have capabilities, they are being challenged by the growing complexity and asymmetry of new generations of LWR core and fuel designs. Also, the codes do not have capabilities to routinely assess uncertainties associated with calculated results. The demand for rigorous uncertainty analyses is growing within all fields where computer calculations are used to support safety analyses and regulatory decisions. As licensees seek to use ever more of the margins between current operations and regulatory limits through the use of ever more realistic safety analyses, the need for uncertainty analysis will grow.

Validation of code predictions with experimental data is quite limited. Validations are done now using legacy data obtained, typically, 20 to 30 years ago. Code-to-code comparisons must increasingly be used to provide validations of analyses. Neutronics data suitable for more definitive validation will be crucial should licensees seek to extend fuel burnup beyond the current regulatory limit of 62 GWd/t. In light of the limited and declining nuclear research capabilities in the U.S., collaborations with other countries may be important for the development of new validation data.

Rather substantial improvements in neutronics computational capabilities will be required for analysis of gas-cooled reactors where epithermal (0.1 to 1.0 Mev) portions of the neutron spectrum are more important and graphite geometry effects are complicated. The NRC staff has outlined well the anticipated needs in this area and has identified several ongoing benchmark activities within the international community. Because of the very lengthy lead times associated with computer code upgrades, it will be useful for NRC to participate in these international benchmark activities.

Computational capabilities available for criticality safety assessments appear adequate for safety review of the MOX Fuel Fabrication Facility and current licensee activities. None of these are finding it necessary to press much beyond current expectations with respect to criticality safety. Criticality analysis may be more demanding for safety analysis of commercial fuel reprocessing systems such as the aqueous UREX process being considered in the GNEP initiative. This modification of the familiar PUREX process will entail new designs and materials that have not been evaluated for criticality issues. An alternative fuel reprocessing method being considered in the GNEP initiative involves pyrometallurgy. Large scale implementation of pyrometallurical processes for nuclear fuels have not been done in the past and the criticality analysis of such processes may entail substantial advances in the current state-of -the-art.

Adequate computational capabilities are, of course, essential to the agency's work. Manpower skilled in the use of the computer codes and knowledgeable concerning the important aspects of neutronics, radiation shielding, and criticality safety issues are even more important. It is essential that the NRC have available to it knowledgeable and experienced manpower in these areas and that this manpower maintains awareness of findings in the fields beyond just the improvement of computational capabilities. Steps taken by NRC management to bolster the manpower within research assigned to the areas of neutronics, shielding, and criticality appear adequate. The challenge the agency has is to keep manpower current with the states-of-the-art in these fields.

8. HUMAN FACTORS AND HUMAN RELIABILITY

Human performance plays a critical role in the safe operation of nuclear plants. Operating experience shows that human performance issues have been contributors to many accidents and unsafe conditions experienced by the current generation of operating plants. Human performance issues are likely to continue to have an important impact on plant safety at existing reactors and for new designs. The staff needs to be able to evaluate the treatment of operator action in risk-informed licensing applications that include the quantification of human reliability under accident conditions. The staff will also need revised guidance and methods to review new reactor designs that are likely to depend on a higher degree of automation than current designs. Therefore, it is essential that the NRC maintain an expertise in the areas of human factors and human reliability analysis (HRA).

Human Factors Research

Human factors research at NRC recently has been focused on support of the regulatory process, regulatory applications, and advanced reactors.

A research project now completed has resulted in the inclusion of safety culture reviews in the inspection process. Attributes of safety culture have been developed and identified as components of the Reactor Oversight Process "crosscutting issues". Detailed guidelines for independent assessments of safety culture have been developed, and inspectors are being trained in their implementation. Inspector feedback is generally positive. This is likely to lead to significant improvements in the NRC inspection program and in the Reactor Oversight Process.

The project entitled "Impact of Operator Workload on Human Performance" is focused on improving the regulatory process. This effort is intended to assess the impact of operator overload on performance. The plan includes the development of licensing requirements as well as inspector guidance and techniques for reviewing the impact of workload on operator performance and plant safety. This is an important project that could strengthen the regulatory process and deserves support both for current reactors and advanced reactors. The success and continuation of this project depends on the cooperation of the industry and this cooperation may not be forthcoming.

The MOX Fuel Fabrication Facility being constructed at Savannah River is highly dependent on operator action during both normal operation and upset conditions. A novel, computer-based permissive system has been proposed for enhancing the quality of operator actions at the facility. A research project is in place to provide human factors review and inputs to the safety evaluation of the fabrication facility. Because of the dependency of the MOX Fuel Fabrication Facility on operator action, this human factors review is likely to be a critical input to the SER and deserves to be supported.

Advanced reactor designs are introducing much greater automation than current designs. Digital instrumentation and control systems and new human-system interfaces are likely to be incorporated into the new designs. These new features are likely to affect human performance in different ways in new designs than in current designs. There is some concern that errors of commission by operators may become an issue for the advanced designs. A project entitled "Human Factors of Advanced Reactors" is intended to develop regulatory guidance and analytical techniques for the review of human factors issues associated with new and advanced reactor designs. The ACRS views this project as essential to prepare the staff for its review of new and advanced reactor designs.

Current research in the human factors area includes a continuing international collaborative research program at the Halden project. As described later in this Chapter, the Halden Project will host a HRA benchmark exercise that the ACRS views as very important. The ACRS continues to be supportive of this collaborative program and recommends continued NRC participation.

Human Reliability Analysis

Human reliability analysis remains one of the important frontiers of PRA. The quantification of human reliability continues to be a challenge in risk assessments. Many techniques have been developed, but analysts can obtain widely different results for human failure probabilities. In mid 1980s, the Ispra Joint Research Center of the European Commission organized a benchmark exercise where many teams used a number of HRA models available at the time to estimate the probability of the operating crew not responding correctly to a transient. The results produced by the teams using the same HRA model differed by orders of magnitude. The results produced by a single team using a number of HRA models also differed by orders of magnitude.

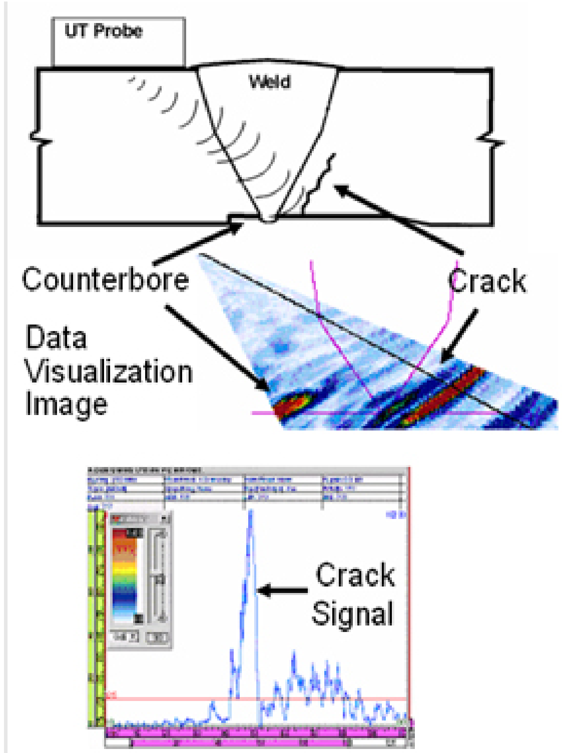
Although these results are fairly old now, not much progress has been made to improve this situation since that exercise was performed. Many techniques have been developed and are being used by the industry to support risk-informed licensing applications. Even within the NRC, the staff uses multiple approaches to HRA for actions following an initiating event. A systematic comparison of such diverse methods and their results has not been performed.

Recently, the Commission directed ACRS to "work with the staff and external stakeholders to evaluate the different HRA models in an effort to propose either a single model for the agency to use or guidance on which model(s) should be used in specific circumstances". The staff and EPRI are in the process of developing a plan that is intended to lead to an integrated approach to evaluate various HRA models. The objective should be to develop a common understanding of the relative importance of factors affecting human performance and ways in which they could be integrated into analyses. Under a project entitled, "HRA Method Benchmarking Using Simulator Data," the staff is organizing an HRA Empirical Study to perform model-to-model comparison to assess the strengths and weaknesses of HRA models. Various operator crews will run scenarios similar to those appearing in PRAs at the simulator in Halden, Norway. Teams of analysts will then analyze the human actions appearing in these scenarios using different HRA models. The results will provide insights on the validity of the assumptions that the teams made and on how the models were applied.

The HRA Empirical Study by itself will probably not be sufficient to develop meaningful quantitative estimates of error probabilities. Additional evidence needs to be collected from operating experience. The research project entitled "Human Event Repository and Analysis (HERA)" continues to support the effort to collect and analyze human performance data. These data can be used to enhance the insights gained from the Empirical Study. Continued support of HERA is important to improve the ability to develop meaningful quantitative estimates of the probability of errors and to reduce the large uncertainties associated with the modeling of human performance.

Human reliability modeling introduces large uncertainties in PRAs. The staff needs guidance in its review of the HRA models used by the industry in licensing applications. A Technique for Human Event Analysis (ATHEANA) is the main NRC tool to estimate human reliability. It is claimed that ATHEANA is even capable of estimating reliability with respect to errors of commission. These are particularly of interest for modern designs that rely heavily on passive safety systems and do not require operator intervention in the event of abnormal events. ATHEANA has been tested in recent applications to fire safety and to pressurized thermal shock. The research project entitled "HRA Application and ATHEANA Maintenance" will support evaluation of current HRA models, the development of an ATHEANA user's guide, and the development of NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator manual Actions in Response to Fire". This effort is clearly needed to provide guidance to the staff in reviewing licensing applications and to demonstrate the strengths and capabilities of ATHEANA. ATHEANA is a state-of-the-art model but is complicated to use. Applications of ATHEANA will ultimately determine if the benefits of its use outweigh the disadvantages introduced by its complexity. The research project on "HRA Methodology for Fire Analysis" will provide further application and demonstration of the effectiveness of ATHEANA.

9. MATERIALS AND METALLURGY



Schematic view of flaw detection at far side of weld using phased array ultrasonic (PA-UT) technique

Based on the number and visibility of programs, materials and metallurgy is the most active area of research within the NRC. This is appropriate in light of the efforts required by the agency to address continuing emergence of unexpected materials degradation in aging LWRs and to monitor the effectiveness of aging management programs. As plants age, known degradation mechanisms will continue to affect components important to safety, and new degradation mechanisms may develop. The agency must develop the technical capabilities to minimize the risk of such surprises and to assess the effectiveness of the industry initiatives to deal with materials degradation.

Current materials and metallurgy research activities are grouped in five areas:

- Environmentally Assisted Cracking in LWRs
- Steam Generator Tube Integrity
- Non-destructive Examinations
- Proactive Materials Degradation Assessment
- Reactor Pressure Vessel Integrity

The research activities in the five areas are appropriate and address key materials issues. The results of these research activities will improve the agency's ability to independently evaluate licensees' efforts to prevent or mitigate environmentally assisted stress corrosion cracking and other environmental degradation mechanisms. RES is making excellent use of domestic and international cooperative programs to accelerate progress, reduce costs, and resolve key issues related to the detection, understanding, and mitigation of materials degradation phenomena. These include the Program for the Inspection of Nickel Alloy Components, the Stress Corrosion Cracking and Cable Aging Program, the OECD Pipe Failure Data Exchange, the Halden Reactor Project, and the Cooperative irradiation-assisted stress corrosion cracking (IASCC) Program (CIR II).

While the research being performed is laudable, materials research planning documents should be improved. It is difficult to determine the priorities, relationships, and schedule for completion of the various tasks within the five principal research areas from documents made available to the ACRS.

Environmentally Assisted Cracking in LWRs

Environmentally assisted cracking is a complex phenomenon influenced by applied and residual stresses, water chemistry, radiation exposure, material composition and microstructure, and fabrication history. Unexpected cracking continues to arise in nuclear power plants as metal components age and radiation exposure increases. In recent years, IASCC has occurred in components internal to the vessels of boiling water reactors (BWRs) and pressurized water reactors (PWRs). Stress corrosion cracking of reactor vessel penetrations and dissimilar metal welds has occurred in both reactor types. Although the industry has implemented improved materials, fabrication processes, and water chemistries to prevent and mitigate these degradation mechanisms, they continue to occur in unmitigated components.

Environmentally assisted cracking can lead to serious secondary damage as observed on the Davis-Besse control rod drive mechanism nozzles. It is clear that the NRC staff must maintain capabilities to evaluate licensees' analyses of the active degradation phenomena in their plants, and the effectiveness of implemented or proposed mitigation methods. The research projects now under way are designed to ensure that the NRC has the necessary technical understanding of the root causes of the various environmental degradation phenomena, their underlying mechanisms, and the long-term reliability of mitigation methods. The effectiveness of stress corrosion cracking mitigation methods used in BWRs and PWRs should be confirmed by long-term and aggressive confirmatory testing. Some methods may lose effectiveness over time, while others (individually or in combination) may demonstrate the ability to protect critical components from stress corrosion cracking for the life of the plant.

Environmentally assisted cracking of reactor materials is an international concern. The project entitled "CIR-II Cooperative Agreement" involves NRC collaboration with the international community to develop a mechanistic understanding and a predictive model of IASCC. This understanding is required to ensure that current mitigation methods will remain effective as plants age and, possibly, to identify more effective countermeasures.

The project on "Environmentally Assisted Cracking of LWRs" includes tasks addressing IASCC and primary water stress corrosion cracking (PWSCC) of nickel based alloys. Researchers at the Halden Reactor are measuring IASCC initiation and growth of relevant materials in both BWR and PWR environments. This effort includes tests of neutron-irradiated specimens to improve the understanding of IASCC initiation and the influence of stress relaxation on crack growth and arrest. It also provides data on the performance of electrochemical potential probes and monitoring techniques in radiation environments. This work is essential and should be continued.

The project on "Investigation of Stress Corrosion Cracking in Selected Materials" is intended to develop a better understanding of the PWSCC mechanism affecting PWRs. Understanding the root cause and underlying mechanisms of this phenomenon is essential for effective long-term mitigation.

Research addressing environmental effects on fatigue of steels used in LWRs has been completed. NUREG-6909 Rev.1 and Regulatory Guide 1.207 have been issued providing designers and regulators with quantitative adjustment factors to account for the effects of environment on fatigue life of reactor materials.

Steam Generator Tube Integrity

Rupture of steam generator tubes in PWRs can lead to accidents that allow radioactive materials released from the core to bypass the reactor containment and enter directly into the environment. Severe accidents involving containment bypass can be risk dominant in some PWRs. Degradation has been observed on both primary and secondary sides of steam generator tubes, and many different phenomena have been observed including: general corrosion and wastage, denting, crevice corrosion, pitting, intergranular attack, and stress corrosion. Careful water chemistry control by licensees as well as the introduction of new tubing materials and improved designs in replacement steam generators had mitigated most of these degradation mechanisms. Stress corrosion cracking is now the dominant threat to the integrity of steam generator tubes.

NRC research is concentrated on the assessment and improvement of non-destructive crack detection methods, the development of analytical models to predict initiation and growth of stress corrosion cracks, and the development of analytical models to predict leak rates and rupture of degraded steam generator tubes.

The reliability of current manual eddy current testing methods used in in-service inspection is being assessed by comparing the results of past human inspector round robin tests on mockups with known flaws with automated eddy current testing inspections.

A variety of advanced NDE and signal analysis techniques are being evaluated for inspecting original or repaired steam generator tubes.

An analytical model based on improved understanding of the mechanisms of initiation and propagation of stress corrosion cracks in nickel based alloys, and the influence of crevice conditions will be developed and validated experimentally.

An analytical model to predict leak rates or rupture of degraded steam generator tubes under normal or postulated accident conditions will be developed and validated experimentally.

Non-Destructive Examinations

Various NDE methods are relied upon to monitor the integrity of reactor coolant systems. These include ultrasonic testing, eddy current testing, penetrant testing, radiographic testing, and visual testing. The reliability and effectiveness of these methods can vary considerably depending on component geometry, materials, and types of defects. There are two primary projects addressing NDE technology.

The project on “Evaluation of Reliability of NDE Techniques” is focused on quantifying the reliability of NDE techniques used in power plant in-service inspection programs. This task was initiated in 2007 to evaluate the accuracy and reliability of the NDE methods and to provide recommendations to the staff to improve the effectiveness and adequacy of in-service inspection programs. This research, which is expected to be completed in 2012, covers the entire spectrum of NDE techniques used in reactor construction, in-service inspection, and repairs. In particular, the effectiveness of in-service inspection techniques on materials with coarse-grained microstructures such as cast austenitic stainless steels, dissimilar metal welds, overlays, and claddings will be evaluated. Because conventional ultrasonic testing is ineffective for cast materials, promising advanced ultrasonic testing methods, including phased array and synthetic aperture focusing techniques are being evaluated.

The project on “Cooperative Activities Reactor Coolant System Pressure Boundary Components” is a shorter-term effort planned for completion 2008. It is focused on comparative non-destructive and destructive examinations of control rod drive mechanism nozzles and J-groove welds recovered from the North Anna 2 reactor vessel head. Cracking of vessel head penetrations has called into question the adequacy of visual testing methods routinely used during in-service inspection of these components. This research is testing real components containing real defects under ideal laboratory conditions to assess the effectiveness of various NDE techniques, and to provide detailed information on flaws in J-groove welds. Research completed on one penetration has revealed that eddy current testing was the most effective technique for detecting through-wall PWSCC. However, complexities in component geometry and PWSCC cracks made interpretation difficult and there was little overlap in flaw detection using different NDE techniques. Results of the in-service inspection of plants, although not in full agreement with the results of laboratory tests, did provide sufficient information to identify the penetration as needing repair. Future research will gather additional NDE reliability data by destructively examining 11 additional crack like indications found using eddy current testing techniques. Based on work completed to date, guidance has been provided to licensees to perform more rigorous inspections, and work will be completed in 2008 to support NRR rulemaking.

These projects are responsive to the NRC's needs and should be continued.

Proactive Materials Degradation Assessment

The nuclear industry and the NRC have often been surprised by unexpected material degradation events. The project "Proactive Material Degradation Assessment' is an NRC initiative to identify materials and systems in LWRs where degradation can reasonably be expected to occur in the future. With such knowledge, current inspection and monitoring programs at plants could be reviewed and modified as needed to provide earlier identification of incipient degradation before it affects plant safety.

The staff has completed phase 1 of the project. A comprehensive assessment of the likelihood and safety significance of possible environmental degradation mechanisms has been completed for approximately 1900 BWR and PWR components, and NUREG/CR-6923 documenting this work has been issued. Phase 2 of the project will establish agreements with industry and international organizations to define research tasks addressing the identified issues of greatest concern. The Zorita Internals Research Program is such an activity which will examine high fluence core internal components which have operated for over 30 years. This research will provide valuable data on the potential for IASCC in PWR environments.

This program should be augmented by a long-term experimental effort to confirm or correct the most safety-significant predictions produced in the proactive assessment. In particular, research should confirm that the various materials, fabrication and water chemistry changes that have been implemented in LWRs to mitigate known environmentally assisted degradation mechanisms will be effective for the life of the plants. In addition, potential materials degradation phenomena identified by the Proactive Materials Degradation Assessment Program should be confirmed experimentally in order to justify proactive regulatory actions and to avoid unexpected failures in operating plants.

The ACRS admires the vision of this Proactive Materials Degradation Assessment program and supports its continuation.

Reactor Pressure Vessel Integrity

The integrity of the reactor pressure vessels has been studied for decades. Licensee obligations to ensure the structural integrity of the reactor pressure vessel during both routine operations and postulated upset conditions, are codified in three general design criteria (GDC 14, GDC 30, and GDC 31) as well as in 10 CFR 50.61 and the appendices G and H to 10 CFR Part 50. The technical bases for these requirements were largely established in the1980s.

Significant progress has been made in completing and closing six research projects on reactor pressure vessel integrity. This completed research has led to updates in several NRC regulatory documents as well as ASME and ASTM codes and standards. Revisions in progress to Pressurized Thermal Shock (PTS) screening criterion in the PTS rule and the associated regulatory guides and Appendices G and H to 10 CFR Part 50 are likely to provide great benefit by removing excessive conservatisms in current requirements and allowing longer life of reactor pressure vessels.

The staff is now focusing pressure vessel research to incorporate probabilistic fracture mechanics in analytical models and to improve understanding of the embrittlement mechanisms controlling the properties of current and future pressure vessels.

The Integrated Component Integrity task will concentrate on the development and validation of a modular tool to perform probabilistic assessments of any reactor pressure boundary component, Success in this endeavor will provide the NRC with a flexible computational tool enabling efficient analyses of new materials degradation phenomena such as those observed at Davis Besse and Wolf Creek. It will also provide a systematic basis for risk-informed assessment of any pressure boundary component in new and advanced reactors.

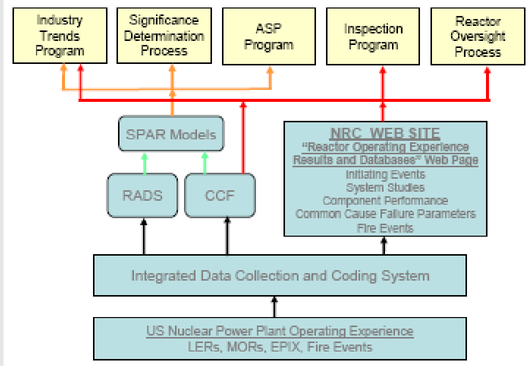
The Integrated Embrittlement task will focus on the development and validation of a methodology to systematically identify and quantify embrittlement mechanisms before they occur in service. Specific issues to be addressed include embrittlement predictions at high fluence, and the influence of nickel content, flux effects, and attenuation effects on mechanical properties of current and new alloys of reactor pressure vessels.

10. OPERATIONAL EXPERIENCE

Operating experience provides an invaluable source of information that is used by the NRC to identify safety significant weaknesses in plant design, operation, or equipment. An important use of this information is to assess the overall state of reactor safety and to determine the effectiveness of the regulatory process.

Two of the most important ongoing research activities associated with operational experience are the Accident Sequence Precursor (ASP) Program, and the Industry Trends Program Support. The ASP is used to monitor the agency's performance against the NRC Strategic Plan goals to ensure the industry is maintaining appropriate levels of safety. The Industry Trends Program Support provides trends and other data used to enhance inspection activities, evaluate risk-informed applications, evaluate the need for changes to the regulatory requirements, and provide the public with operating trends developed independent of licensee-sponsored assessments. These two programs provide important information necessary to maintain a strong and effective regulatory process.

Two projects intended to improve the efficiency and accuracy of the NRC's significance assessments of findings and events at nuclear power plants are "Procedure Development for External Events" and "SDP/ASP Standardization." The procedure for external events will provide generic guidance for the calculation of risk from external events (fire and seismic) and evaluation of findings in ASP and SDP assessments. The SDP/ASP Standardization effort will attempt to make the procedures used in the SDP and ASP programs consistent.



Uses of Operational Data and Analyses in NRC Regulatory Programs

This project will develop analysis guidelines for operating events during low-power/shutdown conditions and calculation of large, early release frequency (LERF) for all event types for use in SDP and ASP. Providing consistent guidance for the two programs and including more guidance for fire, seismic, and low-power/shutdown conditions should enhance the efficiency and accuracy of the staff's evaluations.

One of the key uses of operational experience is in the assessment and improvement of regulatory effectiveness. In the research project to "Assess/Improve Regulatory Effectiveness," the operating experience will be used as one of the tools to determine whether regulatory changes and other NRC actions have accomplished their intended objectives. The effectiveness reviews and subsequent actions will support implementation of high level performance based guidelines, develop decisionmaking methodologies for NRC activities, and develop a technical basis to identify and reduce unnecessary regulatory burdens. Effectiveness reviews are important in maintaining an efficient and effective regulatory process.

The scope and the visibility of work to analyze and evaluate operational data have dwindled dramatically since the function of then AEOD was transferred to RES. Aside from mandated activities such as ASP Program and Industry Trends Program Support, the work is mostly in support of other research efforts and particular licensing efforts. No longer are publicly available, in depth analyses of particular systems or trends being produced. The resources devoted to this activity are approaching the minimum needed to sustain a distinct work unit within the overall NRC research program.

11. PROBABILISTIC RISK ASSESSMENT

In this era of risk-informed regulation, NRC must have state-of-the-art PRA capabilities. Certainly, NRC has been responsible for the development of methods in wide-spread use today. In recent years, as NRC revises both its regulations and its regulatory processes to be risk informed, much of the research work has been in the area of applications. Development of methods has not been a priority. The extensive use of risk information by both the industry and the staff in regulatory decisionmaking, the reviews of new reactor designs, and articles published in the literature have identified areas where methodological advances are needed.

There are many computer codes used by the PRA community to perform the necessary calculations. The industry uses codes such as CAFTA, RISKMAN, and RiskSpectrum. The agency's main tool is the SAPHIRE code that has been developed in the last 20 years and is the basis for the SPAR models that are used extensively in the Reactor Oversight Process. The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment was briefed recently by representatives from EPRI and an international group of researchers that have formed the Open PSA Initiative. Two issues with the present state of PRA software were identified. First, it is hard to check the completeness and correctness of the logic of the PRA models produced using these codes because these models are code dependent. Second, there is disturbing evidence that, under certain circumstances, the probabilistic calculations may be flawed.

The ACRS was informed that there are efforts under way to develop a standard PRA representation format that would promote independence between the logic models and the individual codes. Such a standard would facilitate significantly the review of the PRA logic, a benefit of great value to the agency.

Pending the implementation of a fully integrated PRA representation standard, the agency should explore other methods to improve the transparency and efficiency of its PRA review process. Regulatory activities involve increasingly extensive reviews of probabilistic analyses for risk-informed applications at existing plants, and PRAs that support new reactor licensing efforts. Substantial improvements in staff review efficiency, understanding of the plant-specific PRA models, and reductions in the number of requests for additional information may be achieved through implementation of "PRA viewer" technology. This technology is currently available in some commercial PRA software platforms such as RiskSpectrum and RISKMAN. This technology allows a reviewer to examine all parts of a PRA including the event tree model, the fault tree models, the logic rules, data, results, as well as uncertainty and sensitivity analyses. The reviewer cannot change the actual models or data, and cannot independently re-quantify the PRA results. However, all facets of the actual model used to perform a submitted analysis are directly available at the reviewer's desktop. Information of this detail should allow a capable reviewer to confirm quickly key technical elements of the models and data without the need for reference to cumbersome summary reports or time-consuming requests for additional supporting details. The agency should explore the feasibility of developing this technology, including necessary protocols to control the exchange of application-specific PRA models and data.

Most current-generation codes produce results such as CDF, LERF, and importance measures using a set of accident sequences that is left over after a truncation process in which sequences with frequencies below a user-defined cutoff value have been deleted from the model. Recent studies have indicated that the choice of the cutoff value may affect the results significantly, especially the risk achievement worth which is used extensively in risk-informed decisionmaking by the agency. New calculation methods, such as those based on Binary Decision Diagrams and EPRI's Direct Probability Calculation (DPM™), are exploring the possibility of eliminating the approximations in current methods.

In its 2006 report to the Commission on the NRC Safety Research Program (NUREG-1635, Vol. 7), the ACRS stated that "the staff needs to review the literature concerning Binary Decision Diagrams and evaluate the need to adopt this technology." The ACRS stated further that "the growing importance of the SAPHIRE code and the SPAR models in the regulatory process warrants such an investigation." The ACRS continues to believe that the staff should undertake such an investigation and participate actively in the international activities related to the development of the next generation of PRA software. A short-term product could be an evaluation of the potential impact of the new software on regulatory decisionmaking. The results of such an evaluation would guide the establishment of a longer-term research project in this area.

The quantification of uncertainties is an essential element of risk-informed decisionmaking. The uncertainties in the values of input parameters to the PRA are usually handled well and there are many software packages that propagate these uncertainties to the output quantities. Uncertainties in the models themselves due to questionable or plausible alternative assumptions are still not included routinely in PRAs. Regulatory Guide 1.174, Revision 1, contains a discussion of these uncertainties but does not provide acceptable methods for handling them. There is a need for specific guidance on what we mean by model uncertainties, how to identify them, and how to quantify or manage them in decisionmaking.

Regulatory Guide 1.174, Rev. 1, states that the impact of alternative model assumptions can be investigated through sensitivity studies. Of course, if all such studies lead to a conclusion that the decision is insensitive to them, the issue of model uncertainty goes away. If, however, some studies have a significant impact on the decision, the analysts would have to evaluate the uncertainties associated with the underlying assumptions and factor them in the decisionmaking process. Such studies are rarely, if ever, done. The NRC staff should provide guidance on what a sensitivity study should entail and how its results ought to be used.

A systematic approach to the evaluation of the predictive capability of mechanistic models is being developed for compartment fire models. Preliminary results are documented in NUREG-1824. The staff is to be commended for this effort which should provide insights useful to the general handling of model uncertainties.

The uncertainty issues discussed above should be investigated in the context of regulatory decisionmaking. The ultimate objective is not the development of the most rigorous quantification method but, rather, the management of uncertainty in the integrated decisionmaking process. The ACRS views this to be the proper context for these investigations.

The development of efficient computational tools for the propagation of uncertainties in mechanistic models (usually computer codes) is desirable. This will be particularly important when the need arises to evaluate the unreliability of passive systems in advanced reactor designs. If, for example, one wishes to use Monte Carlo simulation to propagate these uncertainties, the long running times (e.g., hours) of the codes inhibit the rigorous estimation of output uncertainties. These issues have been encountered in other contexts such as the NUREG-1150 study and the performance assessments for nuclear waste repositories. The staff should take advantage of these past efforts.

The development of risk information for regulatory decisionmaking requires often the elicitation of expert opinions. A recent important example is the evaluation of the frequencies of LOCAs of various sizes in the context of the efforts to risk-inform 10 CFR 50.46 (NUREG-1829). Another important study (jointly done with DOE and EPRI and reviewed by a National Research Council Committee) focused on expert opinion elicitation in the assessment of seismic risk (NUREG/CR-6372). Of course, the first studies to formalize the use of expert opinions in nuclear plant risk assessments are contained in NUREG-1150 and its supporting documents. Numerous other examples involving expert opinion utilization can be found within the agency including examples in performance assessments for waste repositories, development of phenomena identification and ranking tables, and the ATHEANA model for human reliability assessment. Few of these studies build on previous NRC-sponsored approaches. The staff should combine the best attributes of these studies and develop a systematic methodology for the elicitation and processing of expert opinions that should be used agency-wide. If there is a need for parts of the general methodology to be different for different applications, these differences should be identified.

PRA issues related to HRA and digital instrumentation and control are discussed in Chapters 4 and 8 of this report.

12. SEISMIC AND STRUCTURAL ENGINEERING

Seismic safety in the design, construction, and operation of nuclear facilities has been an ongoing and evolving issue since shortly after the inception of civilian nuclear facilities. Probabilistic risk analyses show that seismic events can be important contributors to the overall risk of both current and advanced nuclear reactors.

Initially, seismic hazards were addressed by "deterministic" regulation and guidance. In the early- and mid-1990s, however, the NRC began to evolve its siting and design review processes toward the use of a "probabilistic" seismic hazard approach that could better address both aleatory and epistemic uncertainties. Probabilistic seismic hazard analyses (PSHAs) are performed by the NRC in order to independently validate seismic hazard levels provided in nuclear power plant applications. These assessments have revealed differences between the results of NRC analyses and those performed by industry. These differences are attributed to differences in seismic source characterization, ground motion prediction, and approaches to the PSHA process. Research is planned to provide the technical basis for resolving these differences and developing appropriate guidance.

Seismic source characterization is a key area for the seismic research program because it is a major contributor to the uncertainty in the calculation of seismic hazards, particularly in geographical areas that tend to have rare (although possibly large) events and have limited seismic instrumentation such as the central and eastern U.S.

Current research related to seismic source characterization includes a study of the maximum magnitude (M_{max}) appropriate for seismic sources in the central and eastern U.S. and further characterization of the East Tennessee Seismic Zone (ETSZ). The ETSZ issue stems from the review of early site permit application of Vogtle. Work is also planned to enhance the understanding of the seismic processes and characteristics of the New Madrid and Charleston earthquake source zone. Much of this work involves cooperative efforts between the NRC and the United States Geological Survey (USGS). These research efforts are well leveraged and important for providing the data needed by the staff for an independent review of the seismic source parameters used by applicants.

The prediction of ground motions for a given magnitude and distance has always constituted a significant source of uncertainty in seismic hazard results. Uncertainty in these relationships can lead to discrepancies in the hazard levels. A very successful collaborative effort, known as the "Next-Generation Attenuation Relationship" project or the "NGA-West" project produced a set of consensus relationships that are now viewed as the state-of-the-art. Follow-on work is planned on an "NGA-East" project. One result of the revaluation of seismic hazards for rock sites has been predicted motions in the high-frequency range that exceed the enhanced spectrum in Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants ," used for some certified nuclear power plant designs. The effects of these motions depend on the ground motion, i.e., the degree to which two points in the soil are moving in phase at each frequency. Methods for incorporating incoherency effects into seismic design of nuclear facilities to address the effects of high-frequency motion have been proposed by the industry. Although significant work has been undertaken to review the proposed methodology (as part of the development of Regulatory Guide 1.208, "A Performance-

Based Approach to Define the Site-Specific Earthquake Ground Motion," and updates to the related Standard Review Plan sections), a more comprehensive validation program is warranted.

In an effort to standardize approaches to PSHA, the NRC sponsored the research documented in NUREG/CR-6372, "Senior Seismic Hazard Analysis Committee (SSHAC) Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts" (referred to as the "SSHAC guidelines"). This document provides guidance for performing PSHA using four different levels of complexity depending on project needs.

Although the SSHAC guidelines provide a framework for PSHAs, the document does not describe in detail how to conduct PSHAs. Subsequent to the publication of NUREG/CR-6372, practical experience in conducting PSHAs in accordance with the SSHAC guidelines has been gained at Yucca Mountain, in the Swiss PEGASOS Project, and in the EPRI CEUS Ground Motion Project. This experience has not been captured in a form that could benefit an organization conducting or reviewing a major PSHA effort. Development of additional guidance based on this experience is planned.

Recently, the NRC and the nuclear industry have accepted a risk-informed, performance-based approach to the definition of site-specific earthquake motion. The American Society of Civil Engineers (ASCE) has provided some of the technical basis for the performance-based approach through its Standard ASCE 43-05, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities." Although the staff has developed Regulatory Guide 1.208, for the performance-based approach to be acceptable, additional research is planned to complete the development of the technical bases for reviewing a performance-based approach.

The Sumatran earthquake in December 2004 (magnitude ~9) and the associated devastating Indian Ocean tsunami focused considerable attention on structures and facilities that are sited on or close to the coastline. The intensity of an extreme tsunami event could exceed known historical events considered in the design bases of nuclear power plant structures or other nuclear facilities located close to the coastline. In the past, design of coastal facilities did not explicitly address sources known as "submarine landslides," which can trigger extreme tsunami events. The NRC is currently working with the National Oceanic and Atmospheric Administration (NOAA) and the USGS to review the existing state-of-knowledge for the tsunami hazard assessment, mitigation, and landslide mechanics. This collaboration has produced an excellent assessment of the state-of-knowledge concerning submarine landslides that could affect future nuclear power plant sites on the east coast of the U.S.

The NRC routinely participates in cooperative research activities with the International Atomic Energy Agency (IAEA), the Organization for Economic Cooperation and Development (OECD) and its Nuclear Energy Agency (NEA), and other organizations to foster the exchange of data and analysis results. In addition to ongoing activities, the recent earthquake at the Japanese Kashiwazaki-Kariwa (K-K) Nuclear Power Plant provides a unique case history and an opportunity to assess the accuracy of analytical tools that are typically used in seismic design of nuclear power plants and other nuclear facilities.

In August 1999, the NRC signed a 5-year collaborative agreement with the Japanese Agency for Natural Resources and Energy, Nuclear Power Engineering Corporation (NUPEC). The collaboration was extended for an additional 5 years (expiring in August 2009) with the successor to NUPEC, the Japan Nuclear Energy Safety Organization (JNES).

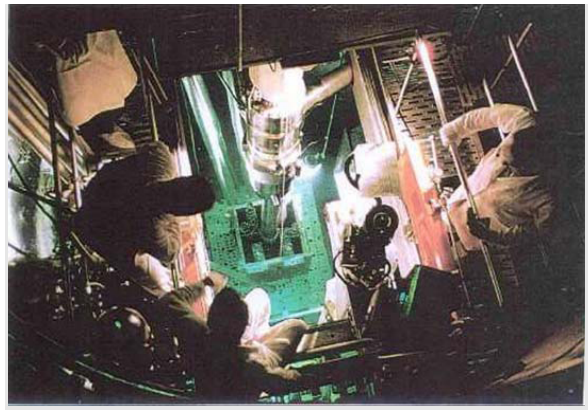
The research conducted and in-kind information exchanged under this collaborative agreement leverages NRC resources to obtain earthquake test data for scale-model structures, multi-axial shear walls, piping, and equipment fragility tests. Collaboration with JNES has proven to be an economical way for the NRC to obtain the results of large-scale test programs that are not available anywhere else in the world. In addition, this research program provides opportunities to interact and exchange information with other Japanese organizations, thereby ensuring that the NRC staff remains cognizant of all ongoing seismic research in Japan.

13. SEVERE ACCIDENTS

In the last three decades, the NRC has conducted substantial research in severe accident phenomena through experiments, data analysis and evaluations, as well as the development of an integrated, systems-level computer code (MELCOR) for use in analyzing severe accident progression. The NRC requires such expertise and analysis capabilities to help support regulatory decisions for current and future nuclear power plants as well as to help the staff in its transition to a more risk-informed regulatory framework.

Support of regulatory needs implies sufficient understanding of severe reactor accidents to estimate whether risks reach a level to threaten adequate protection of the health and safety of the public. The NRC invested heavily in severe accident research to achieve the needed understanding. Once this need was met, the NRC curtailed its investments in severe accident research to needed levels for on-going analysis and risk-informed activities.

Severe accident research is continuing internationally, with substantial experimental programs in the Pacific Rim (Japan and Korea) as well as Europe (France and Germany). The NRC has developed an effective strategy to maintain its leadership in severe accident analytical capabilities as well as to maintain its state-of-the-art knowledge and understanding of key severe accident phenomena by collaborations in international research programs. The knowledge gained from NRC's past experimental work and current ongoing international efforts are systematically incorporated into MELCOR accident analysis code. In addition, the NRC staff has entered into a number of international cooperative experimental research programs to obtain key data for validating the MELCOR code as well as improving its accuracy and realism.



PHEBUS-FP

MELCOR Code Development and Usage

The MELCOR code was originally developed as a PRA tool, and thus, models the reactor coolant system and safety systems as well as containment systems in a less detailed manner than more mechanistic thermal-hydraulic and fuel rod models. A major current activity is to consolidate the key physical models and capabilities of more detailed severe accident codes into MELCOR. This effort will provide an efficient state-of-the-art code for severe accident analyses. In addition, the MELCOR code has been adopted by a number of international nuclear safety organizations (e.g., IBRAE-Russia, PSI – Switzerland) and universities (e.g., Purdue, Texas A&M) for use in severe accident analyses. Finally, the MELCOR code and its fission product source term transport and consequences model (MACCS) is being used as an integral part of the SOARCA project. MACCS is a widely accepted tool used for consequence analysis and its continued development and support is important to the agency for both safety and security regulatory applications. The ACRS is supportive of this effort.

Collaborative Severe Accident Experimental Programs

Collaborative severe accident research programs that the NRC has joined are making good progress and some of the key accomplishments are noted below.

ARTIST: This program involves an experimental study being completed at the Paul Scherrer Institute (PSI) in Switzerland. The objective is to measure the aerosol removal on the secondary side of steam generators during accidents in a PWR, given that containment bypass occurs due to a steam generator tube failure. Such bypass accidents are often risk dominant for PWRs. Based on initial results, the key areas where additional data are needed are: gas jet behavior across the tube bank; particle resuspension as well as particle inertial impaction, and turbulent deposition under certain flow conditions.

MASCA: The MASCA experimental program as well as RASPLAV, its predecessor, was begun in Russia at the Kurchatov Institute to understand the behavior of reactor core debris in the lower plenum of the reactor pressure vessel. The technical objective is to determine if the core debris can be retained within the reactor pressure vessel with water flooding on the outside of the vessel. The major experiments have been completed. Results of the experiments are now being incorporated into the MELCOR computer code. There is an interest in maintaining capabilities for follow-on experiments that may be needed for advanced reactor designs. No commitment to such a follow-on program has been made.

OECD-MCCI: This program is an international collaborative experimental study being conducted at the Argonne National Laboratory and is focused on investigating the heat transfer to an overlying water layer to cool molten core materials interacting with concrete and its viability for long-term coolability. The MCCI project has completed a second phase of experimentation and consists of two experimental efforts: Small-scale Water Ingression and Crust Strength Tests and Core Concrete Interaction Tests.

The Small-Scale Water Ingression and Crust Strength tests focus on quantifying the heat loss to an overlying water layer by mechanisms other than conduction-limited heat transfer, and to measure the strength of the crust formed during flooding of the melt. The Core Concrete Interaction tests are focused on resolving uncertainties in axial versus lateral power splits and respective concrete ablation rates. The tests attempt to replicate as closely as possible conditions at plant scale and provide data to verify and validate predictive codes. These tests allow water flooding after partial ablation to obtain debris coolability data at later stages in the MCCI process. Results of the tests need to be incorporated into the MELCOR computer code.

PHEBUS-FP: This program consists of five large-scale, in-pile integrated tests of: fuel degradation; fission product release; radionuclide transport through a model of a reactor coolant system; and aerosol behavior in a model containment. These tests have been designed to allow validation of core degradation and fission product release and transport models. Although these tests have been completed in 2006, data analysis and associated computer simulations are still on-going. These experiments have proved to be important in providing physical insights on fission product behavior, validating MELCOR modeling of radionuclide release from degrading reactor fuel, and validating the alternative source term used in 10 CFR Part 100 plant analyses. NRC has also joined a second phase of the program that will conduct separate-effects tests to better understand key source term phenomena. For example, this program will address the containment chemistry of radioactive iodine, as well as investigate fission product release from mixed-oxide fuels and high-burnup fuels under accident conditions.

The ACRS encourages the strategy that the NRC staff has developed to support regulatory decisions for severe accidents via computer code development and experimental data analysis and evaluation. This approach can successfully maintain and update its modeling capabilities for severe accident analyses. The planned program extensions and continuations of these collaborations are well worth the investment. This type of collaboration in experimental research could be emulated in other NRC research areas such as thermal-hydraulics and digital instrumentation and control.

14. THERMAL HYDRAULICS

Thermal-hydraulics research aims at improved prediction of pressures, temperatures, and flow rates in nuclear systems and containments during normal operating and accident conditions. Such predictions require understanding and modeling of flow and heat transfer in complex geometries. The flows may involve more than one phase, for example, vapor-liquid or liquid-solid flows. Of particular importance in the regulatory process is prediction of thermal-hydraulic conditions such as fuel clad temperatures and containment pressures in anticipated transients and postulated accidents. Independent evaluation of licensee's thermal-hydraulic analyses has long been a key element in NRC design certification process.

Historically, early thermal-hydraulic analyses of nuclear power plants employed very conservative bounding assumptions, assuring large safety margins with regard to allowable temperatures and pressures over a wide range of accident and operating conditions. With time, the database of experimental findings and the confidence in analytical predictions have grown, allowing the NRC to consider submittals from licensees employing best-estimate thermal-hydraulic analyses together with estimates of uncertainties. As these analyses have grown ever more sophisticated, it has been necessary for the NRC to continue development of state-of-the-art thermal-hydraulic computational tools and more sophisticated understanding of important thermal-hydraulic phenomena. To this end, the NRC maintains competence in the thermal-hydraulics field and capability to conduct confirmatory analysis through its research program.

OECD/NEA ROSA Program

In 2005, under the auspices of the OECD-NEA, the NRC joined 19 other international organizations in an agreement with the Japan Atomic Energy Agency (JAEA) to conduct thermal-hydraulic experiments using the ROSA-V configuration of the Large Scale Test Facility (LSTF), a full-height, 1/48 volume-scaled model of a four-loop pressurized water reactor.

The major elements of the current NRC thermal-hydraulics research program fall into three general areas:

- The TRACE computer code development and validation
- Experimental studies of thermal-hydraulic phenomena

- Thermal hydraulics of sump screen blockage

There are also smaller efforts directed towards maintenance of legacy codes such as RELAP5, participation in international benchmarking activities for thermal-hydraulic sub-channel codes, and addressing BWR ECCS suction strainer concerns arising from possible gas entrainment (GSI-193).

The main features of the first two elements of the thermal-hydraulics research program are discussed below. The issue of sump screen blockage is being actively followed by the Committee. The ACRS has been providing reports on the technical approach and progress of research in this area.

TRACE Computer Code Development and Validation

In the mid-1990s, a prudent decision was made that the several primary reactor system thermal-hydraulic codes that were in use at that time be consolidated into a single code. The several codes included RELAP5 (for LOCA), TRACP (for PWR LOCA), TRACB (for BWR LOCA), and RAMONA (for BWR stability).

The models, correlations, and solution methodologies in these codes did not reflect the state-of-the-art and required in-depth review and modification. It was also recognized that they had been designed at a time when computer capabilities were limited and included many structural aspects, such as memory management, that were no longer needed and increased the cost of continued code maintenance and development. The availability of graphical user interfaces and their wide acceptance also suggested the desirability of incorporating similar capability into the NRC codes. All these considerations led to extensive code consolidation, model improvements and implementation efforts, culminating in the development and validation of the TRACE computer code.

TRACE is intended to serve as the main tool for the confirmatory analyses of a broad range of thermal-hydraulic problems for current and future reactor designs. It has the potential to offer significantly enhanced capabilities for state-of-the-art analyses of thermal-hydraulic issues. Several important technical issues, such as core stability and ATWS behavior, involve coupling between neutronics and thermal hydraulics requiring that TRACE be properly coupled to neutronics codes like PARCS, an activity that is still ongoing. TRACE also has the capability to interface with the CONTAIN code for containment response analysis as well as with other computational tools, including MATHCAD. Applications include analyses in support of certification of new reactor designs and the regulatory review of power uprates for currently operating reactors.

Much of the work in the period under review here has gone towards documentation, validation, and the peer review of TRACE. All these are current work items, and high priority is being given to completing these tasks expeditiously. Efforts are also being made to use TRACE in support of the reviewing the ESBWR and EPR design certification applications. These designs have some novel thermal-hydraulic features that may challenge TRACE capabilities.

The staff has responded positively to the previous ACRS recommendations related to TRACE, and the required activities are now under way, including initiation of the peer review. It remains to be seen whether the work will be completed on a schedule that enables incorporation of the TRACE code into the regulatory process, particularly with regard to reviewing several power uprates and design certification applications currently under consideration.

Experimental Studies of Thermal-hydraulic Phenomena

Thermal-hydraulic phenomena involved in normal and accident conditions for LWRs are complex, and often involve the difficult-to-model flow of two-phase mixtures (steam and water). Predictions from computer codes of such phenomena need extensive experimental validation, and there are many effects, such as those involving multidimensional flows in complex geometries where large-scale experiments are the primary means of confirming the validity of these predictions. In view of this, NRC-RES has maintained several large, complex experimental facilities:

- APEX facility at Oregon State University for PWR-related problems
- PUMA facility at Purdue University for BWR-related problems
- RBHT facility at Penn State University for PWR emergency core cooling problems

In parallel, the NRC has collaborated with international groups in undertaking experiments in facilities abroad, notably the Japanese ROSA facility.

APEX is a medium-sized, reduced-height test facility that has been used in the certification of the AP600 and the AP1000 reactor designs. In the period under review, it has been used to assess the predictive capability of thermal-hydraulic codes for heat transfer in steam generators under partially voided primary system conditions, particularly reflux condensation in the steam generator. The facility is also being used to study condensation on the primary side of steam generators. This is of interest for the EPR design, which incorporates controlled secondary-side cool-down for pressure reduction in small-break LOCAs.

Since APEX is a reduced-height facility, properly scaled natural convection tests are difficult to define. Consequently, the NRC is participating in the OECD ROSA program to study steam generator behavior during depressurization and in natural circulation conditions. The full-height ROSA facility is able to provide both integral and separate-effects experimental thermal-hydraulic data. NRC is collaborating in OECD/NEA programs that also cover a range of other important thermal-hydraulic phenomena, such as thermal stratification in horizontal legs during emergency core coolant injection, the effects of water hammer phenomena, and small-break LOCA without scram. Such collaborative efforts are to be commended, as they take advantage of facilities that are of a scale and capability that do not currently exist in the U.S. Furthermore, they draw on the expertise of international partners, who have continued to maintain a very high level of capability in thermal-hydraulics field.

The PUMA facility is also of reduced height, and is being used to perform integral LOCA tests of interest for the ESBWR design. In particular, the beyond-design-basis passive safety system response of the ESBWR design is to be evaluated and provide data for TRACE assessment.

In addition, an exploratory research program to develop so-called “closure relationships” for the evolution of interfacial area in two-phase flows is being undertaken at Purdue University. It is expected that when the data encompass the range of flow regimes expected in two-phase flows, then a model of interfacial area evolution will be incorporated into the TRACE code, potentially improving its accuracy and reliability in the next-generation version. Results from this program have been slow in coming and the strategy for utilizing them in TRACE still remains to be elucidated.

The RBHT facility at Penn State University was developed to address issues related to emergency core cooling, including the development of a better understanding of reflood and rewetting in realistic, bundled geometries. Currently, this facility is being used to provide data for the development of spacer grid models for the TRACE computer code. The effects on heat transfer of both entrained drops and non-condensable gases are also being determined.

It is clearly desirable that the NRC maintain capability to assess thermal-hydraulic phenomena in both existing and future reactor designs. In view of this, maintenance of research programs in U.S. experimental facilities is justified. Participation in international programs utilizing large-scale facilities abroad is also to be encouraged. The advantage of using the full-height international facilities is that the understanding and correlations developed can be applied to reactor systems more reliably and do not depend on scaling analyses required for reduced height systems, which are difficult to develop and validate for two-phase flows.

The accelerated efforts to document, validate, and peer review TRACE and enable its incorporation for confirmatory analysis in the regulatory process is laudable. The review of several BWR extended power uprate applications require coupled neutronic/thermal-hydraulic confirmatory analyses, both for operational and ATWS instability evaluations. These will prove to be a challenge in the near future. Furthermore, future design certifications will require the capability to finely nodalize riser structures in steam generators and chimneys to capture oscillatory phenomena and refluxing which will necessitate excellent numerical stability and computational efficiency.

In this regard, it should be noted that large-scale parallel computer clusters are becoming available at low cost. Undoubtedly, licensees and vendors will take advantage of such capabilities using their own codes. It is therefore desirable that the NRC also institute long-term programs to take advantage of massively parallel distributed memory systems to speed run times and allow finer resolution in the numerical models used for confirmatory analysis. The development of codes to be scalable on large, parallel machines is therefore highly encouraged.

The ACRS anticipates that licensees will increasingly employ multidimensional computational fluid dynamic (CFD) capability to resolve problems which the current generation of thermal-hydraulic codes, such as TRACE, are unable to do. Multidimensional effects are important in downcomers, plena, and in large pipes in the vicinity of discharge locations, e.g. automatic depressurization systems. The NRC currently has very little effort in the area of CFD other than some testing of commercial CFD codes. Although these are used in the process industry for qualitative indications of phenomena, they are validated to a much less rigorous standard than codes for nuclear use, and the source codes are not available. They appear to include a number of *ad hoc* fixes to improve stability and robustness which may affect their predictive capability for situations that cannot be studied experimentally. NRC has proposed in the NPHASE effort to develop a CFD model suitable for regulatory applications. The ACRS questions the feasibility of developing such a code within the resource constraints of the agency. The international nuclear community, on the other hand, has instituted the development of multidimensional CFD capabilities. CFD codes are being developed and validated to standards of reliability and accuracy required for use in nuclear systems evaluations. NRC should join in this effort and provide adequate resources to allow productive participation.

CFD analyses are already being submitted for a number of issues such as particle drop out during containment debris transport following loss-of-coolant accidents. The NRC has no confirmatory capability to check such submittals. Such capabilities will also undoubtedly be needed for advanced reactors, such as high-temperature gas-cooled reactors, and for evaluation of new fuel designs.

The experimental facilities being maintained by the NRC are aging and require many assumptions regarding scaling to interpret the data obtained due to their reduced-height design. To some extent, this problem can be alleviated by participation in international programs in which full-height facilities are utilized. NRC participation in such programs is therefore to be encouraged and should be increased in the future.

15. ADVANCED NON-LWR DESIGNS

Advanced reactors using gas-cooling or liquid metal-cooling rather than conventional light water cooling pose a conundrum for NRC and the planning of its research efforts. The non-LWRs of interest to NRC can be categorized as liquid metal-cooled reactors (LMRs) and high temperature gas-cooled reactors (HTGRs). The Toshiba 4S (Super Safe, Small, and Simple) is a LMR. In addition, LMR technology is being considered for the Advanced Burner Reactor (ABR) proposed under the DOE GNEP initiative. The Next Generation Nuclear Plant (NGNP) prototype will also utilize HTGR technology when the concept is developed. NRC has some experience with these alternative reactor technologies, but does not have in place the technical infrastructure that will be needed to support an effective certification review of a reactor based on such technologies. NRC is currently considering how to address the evolving demands related to non-LWRs, IRIS, and other unique designs. The Office of New reactors (NRO) is dubious of the feasibility of developing this technical infrastructure and conducting research in parallel with a certification review. The ACRS certainly agrees with the NRO on this point. It is clear that it will take some time to develop the needed technical infrastructure at NRC. What is not clear is when applicants might be able to submit a certification application with an adequate technical basis. This makes it difficult to assign priority and allocate resources for advanced non-LWR research.

The research staff has identified well the technical issues associated with the certification of a gas-cooled reactor. They have done this based on an expert opinion elicitation process that identified important phenomena and ranked these phenomena with respect to safety importance and the current level of understanding. It will be important for NRC to use results of the phenomena identification and ranking to communicate expectations for a defensible certification application for a gas-cooled reactor. Especially crucial will be defining areas where analyses must be substantiated by pertinent, prototypical experimental results.

NRC should initiate activities to assess the state of knowledge in other technologies before there is further loss of limited available capability (worldwide) in some of these technologies. The state–of-knowledge report should identify significant gaps in knowledge that might be necessary for future licensing of some of these unique designs. Specific focus should be given to understanding severe accident behavior and crucial areas of safety (e.g., reactor fuel, high temperature materials). This effort can provide a significant input to the development of safety requirements for some of these designs. Multinational cooperation at an early stage could lead to common safety standards and a sound and more cost-effective research program

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