NUREG-1635, Vol. 7

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
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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Manuscript Completed: March 2006
Date Published: May 2006

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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). These observations and recommendations focus on that portion of the NRC research program dealing with the safety of existing nuclear reactors and advanced light water reactor designs, such as the Economic Simplified Boiling Water Reactor (ESBWR) submitted for certification. The research strategy for more advanced reactors that are not based on water reactor technology such as the Generation IV reactors being studied by the Department of Energy is also discussed. 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 ACRS also attempts to identify research that had progressed sufficiently to meet current and anticipated regulatory needs so that it could be curtailed in favor of more important activities. This report does not address research on the security of nuclear power plants. Comments on such research will be reported separately. Also, the ACRS does not comment on research activities dealing with nuclear waste issues. The Advisory Committee on Nuclear Waste (ACNW) will report on these research activities.

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ACNW	Advisory Committee on Nuclear Waste
ACR-700	Advanced CANDU Reactor-700
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASP	Accident Sequence Precursor
ATHEANA	A Technique for Human Event Analysis
ATWS	Anticipated Transients Without Scram
BWR	Boiling Water Reactor
CAMP	Code Applications and Maintenance Program
CFR	Code of Federal Regulations
CRDM	Control Rod Drive Mechanism
CSARP	Cooperative Severe Accident Research Program
DOE	Department of Energy
ECCS	Emergency Core Cooling System
EMI	Electro Magnetic Interference
EPIX	Equipment Performance and Information Exchange System
EPR	Evolutionary Power Reactor
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
FY	Fiscal Year
GDC	General Design Criterion
GSI	Generic Safety Issue
HERA	Human Event Repository and Analyses
HRA	Human Reliability Analysis
HSST	Heavy Section Steel Technology
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
ICET	Integrated Chemical Effects Tests
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
INL	Idaho National Laboratory
INPO	Institute of Nuclear Power Operations
IPEEE	Individual Plant Examination of External Events
IRIS	International Reactor Innovative and Secure
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
LANL	Los Alamos National Laboratory
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant Accident
LPSD	Low Power and Shutdown
LWR	Light Water Reactor
MACCS	MELCOR Accident Consequence Code System
MOX	Mixed Oxide

ABBREVIATIONS (Cont'd)

NDE	Non-Destructive Examination
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NIST	National Institute os Standards and Technology
NRC	Nuclear Regulatory Commission
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
PBMR	Pebble Bed Modular Reactor
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor
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
SDP	Significance Determination Process
SMIRT	Structural Mechanics in Reactor Technology
SNAP	Symbolic Nuclear Analysis Package
SPAR	Standardized Plant Analysis Risk Model
SRM	Staff Requirements Memorandum
SSHAC	Senior Seismic Hazard Analysis Committee
TRACE	TRAC-RELAP Advanced Computational Engine
UNM	University of New Mexico
U.S.	United States

1 INTRODUCTION

The Nuclear Regulatory Commission (NRC) maintains a Safety Research Program to:

- Ensure its regulations and regulatory processes have sound technical bases.
- 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 decisionmaking.

These essential missions for the research effort were defined when the NRC was established and there was limited experience with the operation of light water nuclear power plants. The need for research remains today, despite the growth of experience with existing power plants, because:

- Nuclear power plants age and encounter challenges of material degradation not anticipated when the plants were designed.
- The NRC considers applications for extending licenses, uprating the operating power levels of plants, and new plant licenses.
- Reactor fuels are used to higher levels of fuel burnup and new cladding alloys for the fuels are introduced.
- Mixed-Oxide (MOX) fuel is considered for the disposal of excess weapons-grade plutonium.
- The NRC evolves its regulations from a deterministic foundation to a risk-informed basis that makes ever greater use of best-estimate analyses to assess safety.
- New technologies including software-based digital instrumentation and control (I&C) systems are backfit into the existing nuclear power plants.
- New water reactor designs such as the ESBWR, which uses passive systems, have been submitted for certification.

There are on the horizon new power reactor concepts that are not based on the water reactor technologies used in the current fleet of power reactors. The U.S. Department of Energy is studying power reactors that use gas cooling, liquid metal cooling, and molten salt cooling. Reactors that use fast rather than thermal neutrons for fission are being studied with the intent of development. These new reactors make it important for the NRC to consider evolution of its regulatory system from one that is specific to water reactor technologies to one that is not specific to particular reactor technologies, but still lead to adequate protection of the public health and safety. This will require substantial research not only for the early development of technology-neutral regulations, but also, in the longer term, for the development of technology-specific regulatory guidance and plans for reviewing specific license applications.

In this report, the Advisory Committee on Reactor Safeguards (ACRS) presents its observations and recommendations concerning that portion of the NRC Safety Research Program devoted to regulation of existing light water reactors (LWRs) and the certification of advanced water reactor designs submitted for certification such as the ESBWR. The ACRS also makes observations on the need for research in anticipation of more advanced power reactor concepts. Observations and recommendations on research dealing with the security of existing nuclear power reactors and nuclear facilities will be provided in separate reports and are not discussed here. The ACRS does not comment on research activities dealing with nuclear waste issues. The Advisory Committee on Nuclear Waste (ACNW) will address such research separately.

In its review of the NRC Safety Research Program, the ACRS considered the programmatic justification for the research as well as the technical approach and progress of the work. The ACRS supports research that:

- Provides support to the identification and resolution of current safety and regulatory issues.
- Provides the technical basis 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 initiatives of the agency such as the development of "technology-neutral" regulatory systems.
- 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.

This review of the NRC Safety Research Program identifies some research activities that have made valuable contributions to the agency mission in the past, but now have reached the point where additional research is not needed for efficient and effective safety regulation. This review also identifies research activities that could benefit by greater collaboration with research activities elsewhere in the world, including collaboration with researchers in Asia and Europe.

General observations and recommendations concerning NRC research activities are presented in Chapter 2. Observations and recommendations regarding research activities in specific technical disciplines are discussed in detail in Chapters 3 through 14:

- Advanced Reactor Research
- Digital Instrumentation and Control Systems
- Fire Safety Research
- Reactor Fuel Research
- Neutronics and Criticality Safety
- Human Factors and Human Reliability Research
- Materials and Metallurgy
- Operational Experience
- Probabilistic Risk Assessment
- Seismic Research
- Severe Accident Research
- Thermal-Hydraulics Research

2 GENERAL OBSERVATIONS AND RECOMMENDATIONS

The NRC Safety Research Program is largely focused on addressing near-term regulatory needs of the agency. Current activities are especially concentrated in three disciplines:

- Materials and Metallurgy
- Probabilistic Risk Assessment
- Thermal Hydraulics

This is an appropriate focus of the current NRC research activities. These activities are discussed further below along with other major aspects of the research program.

The incident at the Davis-Besse Nuclear Power Plant has emphasized, among other things, how important it is for the agency to have a better understanding of the corrosion of metallic systems in the aging fleet of currently operating nuclear power plants. Aging degradation research is necessary to ensure effective aging management for plants operating for extended periods under license renewal and to assess the effect that operation under extended power uprate conditions may have on margins against degradation. Continued challenges posed by stress corrosion cracking of steam generator tubes in pressurized water reactors (PWRs) and systems within boiling water reactor (BWR) vessels further support such focus in the research effort.

Probabilistic risk assessment (PRA) is the basic technology for the risk-informed regulatory system envisaged by the Commission. Research activities are focused now largely on the application of current PRA technology to reactor regulation through the Reactor Oversight Process (ROP). PRA insights are essential to develop and implement revisions to such central regulations as 10 CFR 50.46. They also will play a key role in the development of "technology-neutral" regulatory systems that will have applications to power reactors that are not based on the LWR technology used in the current fleet of operating plants.

The Standardized Plant Analysis Risk (SPAR) models are fundamental tools for risk-informed regulation. A stronger commitment should be made to the improvement of these models and their extension on a timely basis to include fire, seismic, and shutdown risks. The development of these capabilities for the SPAR models will not only provide a regulatory capability but will also encourage industry to more aggressively develop their own capabilities in these areas.

The quality of PRA results depends on good phenomenological models and there are important areas where such models still need further development. Approximate and often bounding risk analyses done for individual plants suggest that the risk of core damage as a result of events initiated by fires can be comparable to risks from other accidents initiated during normal operations. It is important to know if similar results would also be obtained using fire risk assessments of sophistication comparable to the risk assessments possible for normal operations. Such a finding would have ramifications on both regulatory attentions and licensee attentions to safety. The ACRS continues to believe that based on the potential risk significance of fires, fire safety research merits strong consideration in the NRC research program. The collaboration with Electric Power Research Institute (EPRI) is providing a good understanding of the current state-of-the-art methodology for fire risk assessment. This work provides a basis for determining the need for further development.

Thermal hydraulics is a fundamental feature of safety analyses of nuclear power plants. The NRC allows licensees to do either bounding or best-estimate analysis of plant thermal hydraulics for design basis accidents.

Confirmatory review of licensee analyses requires that the agency have high quality thermal-hydraulic analytical tools. Need for such tools is even greater for the analysis of advanced light water reactors that rely on passive systems to achieve safe configurations following accidents.

NRC has consolidated several models of the thermal-hydraulic transient analysis codes into a single code called TRACE. The TRACE code should be subjected to an independent technical review to assess its range of validity. The TRACE code then should be at a point at which it can be used as the primary thermal-hydraulic tool for regulatory analyses. A plan should be developed for its integration into the regulatory process. This integration will require strong support from the management of the NRC user organizations since such a change in the short run will create additional burden on the staff.

The potential for blockage of sump screens by debris dispersed into the sumps during depressurization of the reactor coolant system during an accident remains an unresolved issue. The complexity of the interactions between fibrous and particulate debris, as well as the chemical interactions that can occur among debris materials and solutes in the coolant, make predictions of blockage and consequently screen size requirements difficult. Research needed to reach a prompt resolution of this issue should receive the required resources.

International Collaboration

Reactor safety is an international undertaking. It is important that there not be great differences in safety regulations among the nations making major use of nuclear power generation. The NRC research is making good use of collaborations with other countries on reactor safety research. Much of this collaboration has been in the nature of information exchange. Such information exchanges are important and should continue to be encouraged and supported. They provide access to information and a kind of peer review that might not otherwise be obtained. However, there are other important cases where NRC has gone farther and formally partnered with other countries to leverage resources for experimental investigations of important reactor safety research issues. Such collaborations are especially noteworthy in the disciplines of reactor fuel research and in severe accident research. The combined resources of the partners in these collaborations are yielding higher quality and more extensive results than would be possible in research programs sponsored by individual countries.

Other areas of NRC research could benefit from more extensive collaborations. Such areas include fire safety research and thermal-hydraulics research. The benefits of such collaborations become more apparent as NRC moves to more realistic analyses which may require validation by costly large-scale, integral tests. Collaborations of this type may become even more important in the future as new types of reactors are proposed for certification internationally. To be effective and efficient in dealing with future challenges, NRC should look for opportunities to increase significantly collaboration with other countries. The ongoing collaborative efforts are very extensive with European countries. More collaboration with Asian countries having active nuclear power plant programs should be pursued.

Support for Future Licensing Activities

There has been a recent resurgence in interest in the use of nuclear reactors for the generation of electrical power. Innovative reactor designs are being suggested to sustain uranium resources and to generate electrical energy at much greater efficiency. The U.S. Department of Energy is studying very high temperature gas reactors, supercritical water reactors, sodium-cooled reactors, lead-bismuth cooled reactors, and molten salt cooled reactors. Some of these reactors will use fast neutrons rather than moderated neutrons for fission. These reactors use technologies quite different than those used for the currently operating fleet of reactors. The current regulatory framework is not well suited for the licensing, regulation, or monitoring of such different reactor technologies. Several years ago, it appeared that a substantial portion of NRC resources might need to be devoted to the development of the capabilities to address these very advanced reactor technologies. Today, this is not the case. NRC advanced reactor research resources are focused on addressing issues associated with advanced water reactors such as the ESBWR and the EPR.

This seems to be an appropriate use of NRC's limited resources for advanced reactor safety research. Very advanced reactor concepts have not reached a sufficient state of development that productive use of regulatory research resources can be made. However, work should continue on the development of a technology-neutral framework for regulation, although the development of technology- specific guides can be delayed until it is clearer which alternate reactor technologies will be of the greatest interest.

Development of the framework is not only important for the licensing of non-light-water reactors, but also may provide insights that are useful in developing a more efficient regulatory program for advanced reactors of all types.

There are some indications that certifications may be sought for advanced designs with minimal experimental study of plant response under accident conditions. NRC needs to provide clear guidance on its expectations for the experimental validation of computer models used in the licensing of advanced reactors that do not use familiar technologies. Development of such guidance is an area of advanced reactor research that can be pursued at relatively low cost, but which can play an important role in timely and efficient licensing of advanced reactors with new technologies.

Opportunities for Independent Research

In recent years, a strong effort has been made to ensure that NRC research is supportive of the needs of the line organizations. Focusing NRC research entirely on the immediate needs of the line organizations does, however, entail an important risk. This focus reduces the opportunities for independent thought by the research staff and the opportunities to conduct research that could make more dramatic improvements in the regulatory process, for example, in the tools that support it at a time when there is a rapid increase in workload. The risk is magnified by the diversion of so much research talent to address issues of security of nuclear facilities. There is the further risk of a loss of prestige in the research program focused as it is on issues of implementation. This could eventually lead to a loss in the credibility of the technical basis that underlies regulatory decisions.

It is important that NRC research stay abreast of technological developments that can enhance safety. Areas where developments in the larger technical community can be important to the NRC include reactor fuels, corrosion and materials degradation, man-machine interfaces, technologies for monitoring component performance, inspection techniques, and virtual facility inspections. Where NRC can adopt or adapt developments in other industries, safety can be improved and the efficiencies of NRC reviews enhanced.

One mechanism for RES to interact with the larger technical community is by sharing its own research plans. This has been done for research into digital instrumentation and control. Investigators did creditable reviews of the state-of-the-art, presented them at appropriate professional society meetings as a kind of public peer review, and developed from these state-of-the-art reviews a research plan that is well directed to address agency needs. Sharing research plans with a larger technical community is a strategy that would benefit other NRC research activities. Such interactions also help provide visibility for and help sustain the prestige of the NRC research program.

A Vision for the Future

Nuclear energy will remain an important and perhaps growing component in the mix of energy generating technologies used in this Country. There is the potential that many new reactors could be built in the next 15 to 20 years. It is unlikely that agency resources of either manpower or funds will experience a similar growth. Indeed, the experience level of the NRC staff is likely to decrease due to retirements just when the new plant licensing activities accelerate. A portion of the research program needs to be devoted to the development of a regulatory infrastructure for regulatory work in the next 20 years that supports a staff with less experience dealing with more tasks. Computerization will be undoubtedly an important element of such an infrastructure. The ACRS can foresee, for example, a time when regulatory staff have routine access to superior analysis tools for systems analysis, phenomenological analysis, and risk assessment. Development of such validated and verified tools for routine use by non-specialists will require a research program that is not tied exclusively to the near-term issues of the regulatory process. Appropriate attention will have to be paid to the agency's analytical tools, its access to facilities, and its ability to provide recently recruited staff with a sound understanding of past safety decisions. Availability of good infrastructure will enhance safety and allow for much more efficient and effective NRC review of new reactor designs and licensing applications based on realistic evaluations of safety.

Observations and Recommendations on Specific Research Activities

NRC research has made substantial progress since the last ACRS report, NUREG-1635, Vol. 6, on the research program. This progress has occurred despite the diversion of substantial research talent in the agency to address issues of reactor security that are not reviewed here. Notable accomplishments of the research program in recent years include:

- Multidisciplinary review of pressurized thermal shock criteria
- Performance of high-burnup fuel during reactivity transients
- Embrittlement of zirconium alloy cladding when taken to high burnup.

The ACRS applauds these high technical quality research accomplishments. The ACRS is, however, disappointed at the pace with which these important research results are being used to modify regulations.

Other major observations and recommendations concerning the NRC research activities are summarized below and also discussed in more detail in individual Chapters.

<u>Advanced Reactor Research</u>

Highest priority should be given to those research activities that support the ESBWR design certification process. The importance of tasks associated with the ACR-700 or a related design with higher power depends on whether the certification review for such a reactor is resumed.

Digital Instrumentation and Control Systems

Software-based digital electronic systems are inevitable for both current and more advanced design nuclear power plants. The staff has developed a research plan that addresses the challenges associated with the use of digital technology that will face the agency in the next five years.

The ACRS has recently reviewed and reported favorably on the research plan for digital systems. The ACRS was impressed by the technical quality in the development of the research plan, the scope and content of the plan, and the prioritization of activities in the plan. The ACRS recommends a number of improvements to an already quality research plan, including addition of an explicit element to the plan to study the acceptability of international standards in comparison to Institute of Electrical and Electronics Engineers (IEEE) standards for meeting regulatory requirements concerning digital instrumentation and control systems. This study will be an important element of efforts to develop a multi-national design approval process.

Fire Safety Research

There have been a number of important accomplishments by NRC research in the area of fire protection since the last ACRS report on NRC safety research program in 2004. Fire safety research continues to merit emphasis in the NRC research program.

RES, in cooperation with EPRI has taken some important steps to consolidate the fire PRA research and development activities, conducted over the past few years, into a single state-of-the-art methodology for fire risk assessment.

There are a variety of methods that can be used to model the progression of fires. Some of these have been used in fire protection programs for non-nuclear facilities for many years. The ranges of applicability of these methods have not been well studied or documented. In cooperation with EPRI, a program is in progress to verify and validate a set of fire progression modeling tools. The accuracies of these tools are being examined for different fire conditions and applications by comparison with benchmark tests.

RES has worked closely with the U.S. industry in undertaking generic fire risk research activities. Fire risk is, however, an issue of world-wide concern. RES has not aggressively sought collaborations with the international community to advance NRC capabilities for fire risk assessment. Collaborations with other countries especially in experimental studies may be essential to leverage resources of all partners sufficiently to achieve fire risk assessment capabilities commensurate with what can now be done for risk from normal plant operations.

Reactor Fuel Research

The NRC research on reactor fuel has been concentrated in recent years on the confirmation of regulatory decisions that allow licensees to take light water reactor fuels to burnups of nominally 62 GWd/t. The research on high-burnup fuel is reaching a substantial level of maturity. Some major confirmatory experiments remain to be done - notably experiments on reactivity insertion to be done in a water loop at the CABRI reactor. Since the last ACRS report on NRC safety research program, plans for these experiments have been revised so the experiments which are part of an international collaborative effort now better meet the agency needs. It is important that this work that is so well coordinated both with agency needs and with international partners be taken to completion. Still major findings of the research effort can be reduced to regulatory practice now. This reduction to regulatory practice needs to be initiated and pursued aggressively.

It is evident that high-burnup fuel research will soon achieve results that are adequate for agency needs. The NRC has made clear that it will expect the nuclear industry to provide necessary safety analyses and experimental data should the industry want to take fuel to burnups that exceed the current regulatory maximum. NRC needs to make these expectations more explicit, particularly its expectations for the experimental data needed to support the analyses of high-burnup fuel behavior under accident conditions.

Neutronics and Criticality Safety

The neutronics and criticality safety research program is small but appears adequate to ensure that the NRC has capabilities to meet immediately foreseen regulatory needs.

In the future, more innovative core designs for advanced reactors may be submitted to the NRC. Confirmatory analyses of reactor core physics will be an essential part of the regulatory process for these advanced reactors. The capabilities now available to the NRC in the area of core physics may well be stretched. It will be useful to the agency to understand these future needs. If long-term development activities are identified, such as those that might be needed for analysis of the PBMR, additional research may be needed in this area.

Human Factors and Human Reliability Research

As new reactor designs, likely dependent on a higher degree of automation than the current fleet, are introduced, the need for revised guidance and tools for the NRC staff in human factors and human reliability analysis will increase. RES has initiated a project to develop regulatory guidance and analytical techniques to review human factors for advanced nuclear power plants. The ACRS views this five-year project essential for preparing the staff in reviewing advanced reactor designs.

The quantification of human reliability continues to be a challenge in risk assessments. Human reliability modeling introduces large uncertainties in probabilistic risk assessments. The NRC staff needs guidance in its review of the human reliability models used by the industry in risk-informed licensing applications. Progress has been made with the publication of NUREG-1792, “Good Practices for Implementing Human Reliability Analysis (HRA).” Still, further guidance is needed for reviewers of licensing applications.

Materials and Metallurgy

The NRC is investing heavily in the better understanding of materials degradation issues in the currently operating fleet of nuclear power plants. Such investment is justified in view of significant agency regulatory activities that aging degradation research supports.

The current program is well focused on improving the agency’s ability to independently evaluate licensees’ efforts to prevent, detect, and mitigate environmentally assisted stress corrosion cracking.

The nuclear industry and the NRC have often been surprised by unexpected material degradation problems. As a result, they have responded to such problems in a reactive mode which has proven to be inefficient. The Proactive Materials Degradation Assessment project is an effort to identify potential material degradation problems before they manifest in operating nuclear power plants. The ACRS admires the vision of this undertaking and supports its continuation. The ACRS looks forward to reviewing the initial results of this ongoing effort soon and learning whether the admirable goal of this project is, in fact, feasible.

RES needs to reevaluate the need for continued research into heavy section steel components. This research may be justified if there is a clear need for NRC to develop its capabilities in the area of probabilistic fracture mechanics (PFM) so that it can evaluate licensees' applications. If this is the case, the research needs to be clearly focused on this objective and not the research that the industry should perform to meet its responsibilities to ensure reactor pressure vessel integrity. It appears now, however, that it is NRC that is advancing the state-of-the-art and making available information that allows licensees to reduce conservatism in their analyses.

Operational Experience

The ACRS is supportive of the research activities in the area of operational experience and recommends that these activities be continued. In light of the limited resources allocated to these tasks, RES has done a commendable job in producing outputs in well-documented and thorough fashion. Tasks that are currently in the 2005 research plan related to operational experience should remain funded and should be continued for the foreseeable future.

Probabilistic Risk Assessment

Altogether the scope and the number of activities in the NRC's PRA research program is quite impressive. The ACRS cautions, however, that NRC should not allow its work in such a crucial technology as risk assessment to become totally devoted to the support of line activities. Methods development is still important. As an example, the ACRS notes that considerable research is being reported in the literature regarding Binary Decision Diagrams as tools for solving large fault tress without resort to cutoff frequencies as is now done. The staff needs to review the literature concerning Binary Decision Diagrams and evaluate the need to adopt this technology. The growing importance of the SAPHIRE code and the SPAR models in the regulatory process warrants such an investigation.

Seismic Research

Seismic hazard analysis and structural response are not areas where NRC must maintain state-of-the-art expertise. Such expertise is available to the NRC on a contractual basis. As ACRS noted in its previous report on NRC safety research program, research activities at the agency can be confined to support needed updates to regulatory guides and collaborative work with the international community to stay abreast of developments in other countries. The current research program is, indeed, largely focused on needs of the regulatory process and a few important international collaborations.

Severe Accident Research

The ACRS is very supportive of the strategy NRC has developed to maintain and update its capabilities for severe accident analyses. The leveraging of resources through international collaborative experimental research is especially important. The planned extensions and continuations of current collaborations are well worth the investment.

Thermal-Hydraulics Research

Highest priority should be given to the integration of TRACE code into the regulatory process. As this integration progresses, the research staff can continue its efforts to improve and further develop TRACE on a "time available" basis. The ACRS is concerned now that efforts to improve TRACE lack prioritization and defensible organization. Prioritization of technical improvements might be aided substantially by commissioning a detailed peer review of TRACE. To do this, the staff will have to have available code documentation of outstanding scope and quality. Such high quality code documentation will also be needed if the code is to become part of the regulatory process. Code documentation, then, is a task that ought to take precedence in the thermal- hydraulic research effort.

3 ADVANCED REACTOR RESEARCH

The agency is already engaged in various activities related to a number of new plant designs, including ESBWR, PBMR, IRIS, and ACR-700. The staff has begun its review of ESBWR design certification application. It is anticipated that requests for design certification reviews will be received for EPR, and PBMR. Of these, the ESBWR, ACR-700, IRIS, and EPR can be certified in all likelihood under the current requirements in 10 CFR Part 52 using the design basis accidents as they are now defined. Nevertheless, there will be the need for NRC to verify the thermal-hydraulic assessments made by the applicants for the various designs. This will require review and approval of the computer codes that were used by the applicants for assessing the design basis accidents. Confirmatory analyses will require that design-specific versions of the computer codes TRACE and CONTAIN be available to the staff for audit calculations and independent assessment of separate effects and integral system experiments. Highest priority should be given to those research activities that make such tools available for the ESBWR design certification review. This includes tasks Y6857, Y6898, N6018, and Y6804. The importance of tasks associated with the ACR-700 or a related design with higher power, Y6831, Y6812, Y6899, Y6489, Y6748 and Y6933, depends on whether the certification process for such a reactor is resumed.

Certification reviews for designs such as the PBMR and the 4S that do not use water reactor technology will be more challenging. Although significant efforts were undertaken in the past to license such non-LWR designs under the current regulatory system designed for light water reactors, it would be far more appropriate, effective, and efficient to have the "technology-neutral-framework" for certification of such designs. For timely application to these reactor types (and possibly even more unusual designs in later years), the development of the technology-neutral framework needs to be given high priority and provided sufficient resources to complete the job in 2006 and to allow two years for rulemaking. High priority, then, should be given to the tasks N6205 and Y6487 that will develop a technology-neutral framework for the regulation of advanced nuclear power plants.

The Commission has expressed a desire for "enhanced safety" for new reactor designs. To ensure that new designs have reached enhanced levels of safety, the NRC will require each of the applicants for design certification to submit a full-scope PRA with consideration of uncertainties. The staff must be prepared to review these PRAs, to validate the results and to compare the results with acceptance criteria for "enhanced safety." This evaluation will include undoubtably a complete Level-2 evaluation of accident source terms since LERF (large early release frequency) will no longer be an appropriate safety metric. To review and independently assess the Level-2 analyses of source terms, the regulatory organizations will need design-specific versions of the MELCOR computer code. There is, then, the potential need to develop MELCOR versions specific for the PBMR and 4S designs. Development of such code versions will take time. Second priority should be given then to tasks K6703, Y6801, and Y6619. Again, the importance of developing an accident progression model for ACR-700 depends on resumption of its certification process.

Table 1. Advanced Reactor Research Activities

Job Code	Title	Comment
Y6857	***ESBWR Input Deck Development***	Analysis of DBAs in ESBWR using the TRACE code; This project should have high priority.
Y6898	***ESBWR Design Certification Report***	Support for review of PRA for ESBWR; This project should have high priority.
N6018	***Separate Effects Experiments***	Separate effects tests in support of TRACE model development for ESBWR; This project should have high priority
Y6804	***ESBWR Containment Support***	Analysis of experiments with CONTAIN and MELCOR. This is a high priority task for ESBWR design certification review.
Y6489	***PRA for ACR-700***	Support for review of ACR-700 PRA. This project can be deferred until certification application becomes active again.
Y6899	***ACR-700 Design Certification Support***	Support for review of PRA for ACR-700. This project can be deferred until the certification application becomes active again.
Y6748	***Review ACR-700 Support***	Support for thermal hydraulics review of ACR-700. This project can be deferred until the certification application becomes active again.
Y6831	***Methods Development for ACR-700***	TRAC code upgrades needed for ACR-700 certification calculations. This project can be deferred until the certification application becomes active again.
Y6812	***ACR-700 Input Model Development***	Develop RELAP5 and TRAC-M input models for ACR-700. This project can be deferred until the certification application becomes active again.
Y6933	***Evaluate Severe Accident Phenomena in ACR-700***	Analysis of risk dominant sequences for ACR-700. This project can be deferred until the certification application becomes active again.
K6703	***Coop. Agreement with Center for Advanced Nuclear Energy Systems***	Improve NRC's knowledge and information on advanced reactors. This project is useful but can have a second level priority.
Y6619	***Advanced Reactor PRA Development***	Develop knowledge needed to review advanced reactor PRAs. Second priority work for non-LWR design certifications.

Table 1. Advanced Reactor Research Activities (Continued)

Job Code	Title	Comment
Y6801	***Advanced Reactor/Severe Accident Code Development***	Develop a version of MELCOR code for advanced reactors. This project can have a second level priority.
Y6755	***Materials Evaluations for Advanced LWR Reactors***	Research materials engineering issues for advanced LWRs especially effect of coolant environment on fatigue and in-service inspection and monitoring. This project can have a second level priority.
N6205	***Assistance for Development of a Regulatory Structure for New Plant Licensing***	Development of a technology-neutral regulatory framework. This project should have high priority.
Y6487	***Advanced Reactor Regulatory Framework Development***	Development of a regulatory framework for advanced reactors. This project should have high priority.
Y6741	***Environmental Effect on Containment***	Develop understanding of the properties of concrete in high temperature gas cooled reactors. This project can have a low priority.

4 DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

Software-based digital electronic systems are inevitable for both current and advanced design nuclear power plants. Already such software-based digital electronics appear ever more frequently in systems for plant control. Eventually, they will appear in safety systems. The reliability of digital systems especially when using commercial, "off the shelf" hardware and software has become an issue because they cannot be comprehensively tested. The quality of the requirements for the software cannot be assessed fully through testing. Quality in the software-based systems is achieved through the control of the process of software development. Particular attention has to be given to the requirements for the system software. Failure to specify adequate requirements has often been found to be the root cause of digital system failures. Review and approval of licensee applications to incorporate software-based digital systems in its facility is, then, time-consuming for both the regulator and the licensee. New failure modes that arise in digital systems need to be recognized. Such failures can depend on the operational state of the system at the time of failure. Indeed, testing and maintenance as well as normal operations of digital systems can create the opportunities for their own unique kinds of failures.

Security of digital systems has become a major concern and there needs to be regulatory guidance and acceptance criteria for the security aspects of digital systems. Codes, Standards, and regulations must prompt the designer of digital safety systems to avoid system communications outside of the controlled areas of the plant and the use of wireless technology must be carefully evaluated to prevent interception, interdiction, or interference in communications to digital systems.

Current licensing guidelines provide information on what to review in digital systems. They do not necessarily provide sufficient guidance on how to review submittals or the acceptance criteria to apply. The NRC staff needs a firm technical basis for deciding when review of submittals is adequate and when confirmatory analyses are necessary. The situation will get worse with time. Digital systems in nuclear power plants are expected to become more numerous. The complexity of these systems will increase. There is the potential for the consolidation of what are now discrete analog safety systems into a single digital system. At the same time, there is interest both within the agency and on the part of licensees to adopt risk-informed techniques for the review of digital software systems. NRC lacks the technical basis to support risk-informed reviews of digital systems. Currently, the ability to model the reliability of software-based digital systems in PRAs is very limited. Without quantitative risk information, a much less defensible, qualitative, "graded approach" to the review of digital systems is likely to emerge.

If the use of digital protection systems and control systems becomes as widespread as now predicted, review of digital systems as part of ITAAC (Inspections, Tests, Analyses, and Acceptance Criteria) may eventually become a burdensome, time-consuming aspect of the licensing process. Methods and tools to facilitate confirmation that "as built" systems conform to accepted designs are going to be needed. As use of digital systems becomes more extensive in nuclear facilities, NRC may find it necessary to reconsider its current positions on defense-in-depth and diversity in instrumentation and control systems.

The nuclear industry is not a major user of digital technology relative to many other industries. Yet, the consequences of failure of digital systems in nuclear power plants are likely to be less acceptable to the public than are failures of such systems in other industries even when consequences are significant. Greater rigor in the review of digital systems is necessary for nuclear applications of these systems. It is expected then that NRC will have to “blaze new paths” in this area through research. In particular, the usual industrial practice of separately considering hardware and software reliabilities may not be adequate for nuclear systems and a more integrated or systems approach may be needed.

The staff has developed a research program plan that addresses these challenges that will face the agency in the next five years. Critical reviews of the state of the art in several areas were completed, documented and presented before audiences in professional societies. Recommendations made to the NRC by independent bodies, including the National Academy of Sciences were considered in the development of the plan. Inputs from the program offices at NRC (NRR, NSIR, and NMSS) were also obtained. The research plan is well directed toward meeting the agency needs and is intended to provide:

- Improved technical guidance for review of digital systems
- Technical support for developing improved acceptance criteria for assessing the safety and security of the systems
- Tools and methodologies for improved review of digital systems
- Technical bases for including models of digital systems in PRAs

The research plan has six major elements:

- Systems aspects of digital technology
- Risk assessment of digital systems
- Emerging digital technology with application to nuclear facilities
- Software quality assurance
- Security aspects of digital systems
- Advanced nuclear power plant digital systems

Within each of these major elements of the plan, there are a number of subelement. The staff has prioritized work on the subelement basis. Now, the major focus of the work is on collection of data on the failure modes of digital systems, including international experience with digital system failures, software quality assurance, environmental stressors on digital systems, modeling digital systems in PRAs and cyber security of digital systems. Within the general element of emerging digital technologies applicable to nuclear facilities, attentions are on system diagnosis, prognosis and on-line monitoring as well as wireless technology. Research on digital systems for advanced nuclear power plants was given a low priority. Perhaps, future new orders for advanced plants (AP1000, ESBWR, etc.) may create new regulatory demands and cause this priority to be re-evaluated.

The ACRS has recently reviewed and reported favorably on the research plan for digital systems. The ACRS was impressed by the technical quality in the development of the research plan, the scope and content of the plan, and the prioritization of activities in the plan. Indeed, it would help better understanding of other research programs if they were also based on such thorough planning efforts. The ACRS recommends the following to further improve an already quality research plan:

- The plan is currently focused very much on the software aspects of digital systems. Eventually, the research will have to be expanded to recognize the entire system of interest. Though the focus on software is appropriate now, the plan should reflect the need for expansion in scope in the longer term.

- There should be an explicit element of the plan to study the acceptability of international standards in comparison to IEEE standards (such as IEC 60780 in comparison to IEEE 323) for meeting regulatory requirements concerning digital instrumentation and control systems. This study will be an important element of efforts to develop a multi-national design approval process.

- As data on digital system failures are collected and analyzed, the research staff should prepare episodic papers or presentations to professional societies of their interpretations and "lessons learned" for peer review by the larger digital system reliability community.

Table 2. Research Activities in Digital Instrumentation and Control Systems

Job Code	Title	Comment
N6116	***Secure Network Design Techniques***	Develop technical guidance for mitigating cyber vulnerabilities in secure networks
N6095	***Assignment Robert Edwards***	Support analysis of digital systems failures and consequences
Y6962	***Emerging Technologies***	Conduct periodic surveys of the state of the art for a wide range of technology issues in the I&C field
Y6873	***International Cooperative Research Program on Digital I&C***	Search for opportunities to collaborate in the safety assessment of digital systems
N6010	***COMPSYS***	OECD/NEA international program to develop data-base on digital systems failures
K6472	***Risk Importance of Digital Systems***	Develop methods to include digital systems in PRAs
Y6332	***Digital Systems Risk***	Develop a PRA method for modeling failures of digital I&C systems.
Y6591	***Software Reliability Code Measurements***	Large-scale validation of NRC methodology for predicting software reliability in digital systems
N6080	***Interactions with Industry on Standards***	Development of standards on EMI/RFI
Y6475	***Wireless***	Confirmatory research on effects of wireless communications
N6113	***Security of Digital Platforms***	Study in laboratory digital systems generically qualified for nuclear safety applications
N6114	***Site-specific Protocol Analysis***	Study power plant implementation of digital systems generically qualified for nuclear safety applications
N6124	***Digital System Dependability Performance***	Qualify safety of a digital system using a process developed in NRC research
W6851	***Review Guidance for Lightning***	Support for response to public comments on draft regulatory guide; Program completed.
Y6924	***SPACE Engineering Workstation for Review of TXC Applications***	Evaluate the use of the RETRAN tool for review of TELEPERM-based digital instrumentation and control upgrades
Y6349	***Halden Environmentally Assisted Cracking*** (The title of this program is amazingly misleading!)	Despite the name this is research on COS operating experience, ranking software engineering practices and testing digital reliability assessment methods

5 FIRE SAFETY RESEARCH

The fire safety research program can be divided into three technical areas:

- Fire Risk Assessment
- Fire Modeling
- Fire Testing

Each of these areas is discussed below.

Fire Risk Assessment: The nuclear industry has made substantial progress over the past thirty years in the development and standardization of internal events risk assessment. Progress in the development of the methods of fire risk assessment has been much slower. Only a few nuclear power plants currently have full-scope fire risk assessments. The requirements placed by the NRC on the industry for performing Individual Plant Examinations of External Events (IPEEE) permitted the use of simplified and qualitative techniques. Most analyses of fire risk at nuclear power plants were performed with these less quantitative techniques..

As the NRC moves from deterministic regulations to risk-informed and performance-based regulations, the need for quality risk information increases greatly. It is expected that many nuclear power plants will transition from their current fire protection programs 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 National Fire Protection Association (NFPA) standard, NFPA 805, “Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Stations.” This is only possible if a full-scope fire risk assessment is performed for each transitioning nuclear power plant. NRC will need appropriate standard to assess the quality of such fire risk assessments and inspectors will need tools

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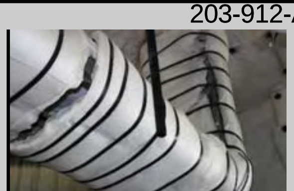

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Confirmatory Testing of Hemyc/MT Fire Barriers

The Hemyc and MT electrical raceway fire barrier systems are used in a number of plants to provide a fire barrier between two trains of safe shutdown equipment within a fire area. In the performance of fire protection inspections at nuclear plants, questions raised regarding the fire resistance capability of these systems. NRC conducted confirmatory testing of Hemyc and MT fire barriers at the Omega Point Test Facility in 2005. All of the configuration tested failed to meet acceptance criteria. A Generic Letter was issued requiring licensees to identify where Hemyc and MT fire barriers are used in their plants and to provide a plan and schedule for corrective actions.

and the knowledge to assess the validity of changes to the licensing basis made at the plants.

RES in cooperation with EPRI has taken some important steps to consolidate the fire PRA research and development activities, conducted over the past few years, into a single state-of-the-art methodology for fire risk assessment. In 2005, the final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," was issued. This document provides a structured framework for the overall fire risk assessment as well as specific recommended practices to address key aspects of the analysis. While the primary objective of the project was to consolidate state-of-the-art methods, in many areas the newly documented methods represent a significant advancement over those previously documented. Although some utilities have used parts of the improved approach, no utility has completed a fire risk assessment using the methodology and submitted the assessment for critical peer review.

Areas of fire risk analysis where further development in methodology is needed have been recognized by RES. These include spurious equipment actuations, post-fire human reliability analysis, aging effects, and low power and shutdown fire risk.

Fire Modeling: Deterministic criteria for fire protection are typically very conservative in their treatment of fire progression. Fire risk assessment, on the other hand, requires a realistic assessment of fire progression. There are a variety of methods that can be used to model the progression of fires. Some of these have been used in fire protection programs for non-nuclear facilities for many years. The ranges of applicability of these methods have not been well studied or documented. In cooperation with EPRI, a Project (Y6688) is in progress to verify and validate a set of fire progression modeling tools. The accuracies of these tools are being examined for different fire conditions and applications by comparison with benchmark tests performed by National Institute of Standards and Technology (NIST) The phenomena identification and ranking table (PIRT) process is being used by RES to identify potential limitations of the fire progression modeling tools. Preliminary draft of multi volumes NUREG-1826, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," was issued for Public Comment in January 2006.

Fire Testing: Confirmatory testing is another critical element of the fire safety program. During the past year, tests were performed at the Omega Point Test Facility on the Hemyc and MT electrical raceway fire barrier systems (see side column). The test results indicated that these fire barrier systems are not capable of satisfying regulatory requirements. It is somewhat distressing that confirmatory testing of these fire barriers did not occur until sixteen years after problems were identified with a similar fire barrier material, Thermo-lag, and five years after inspection teams raised specific concerns about the Hemyc and MT fire barriers. The results of these tests provide further evidence of the continuing value of NRC's confirmatory testing program.

There have been a number of important accomplishment by NRC research in the area of fire protection since the last ACRS report on NRC safety research program in 2004. Fire safety research continues to merit emphasis in the NRC research program. Approximate, and often bounding risk analysis, performed for individual plants indicate that the risk of core damage from fire- initiated events is comparable to or greater than the risk from other accidents initiated during normal operations. It is important to know whether the same conclusion would be drawn if fire risk assessments were performed using tools of comparable sophistication as those used for assessing risk of accidents initiated by internal events. Conclusions based on more realistic fire risk assessments could have ramifications on both regulatory attention and licensee attention to safety. In the interim, risk-informed regulatory decisions are being made with an incomplete understanding of the impact of fire on risk.

RES has worked closely with the U.S. industry in undertaking generic fire risk research activities. Fire risk is, however, an issue of world-wide concern. France, for example, has recently initiated a fire research program in a multi-volume test facility. RES has not aggressively sought collaborations with the international community to advance NRC capabilities for fire risk assessment. Collaborations with other countries especially in experimental studies may be essential to leverage resources of all partners sufficiently to achieve fire risk assessment capabilities commensurate with what can now be done for risk from normal plant operations.

Table 3. Fire Safety Research Activities

Job Code	Title	Comment
N6107	***10 CFR 50.48C - related Technical Activities***	Develop fire PRA methods, tools, and data. Perform demonstration studies. This is a collaborative effort between NRC and EPRI.
N6108	***Fire Risk Assessment and Risk Applications***	Improve fire PRA approaches. Develop test plan to address spurious equipment actuation issues.
N6134	***LPSD Level 1 & Fire Risk Standard***	Supports NRC staff in the development of industry standards.
Y6651	***Effects of Switchgear Aging on Energetic Faults***	Assess the aging of medium voltage switch gear as it affects the potential for energetic electrical faults. Such faults are thought to contribute significantly to fire initiation. The work addresses how aging affects fire risk.
Y6688	***Fire Model Benchmarking and Validation***	Benchmark fire model computer codes against fire experiments performed by NIST. Such validation is necessary to ensure that appropriate tools are used for regulatory applications.
Y6817	***Fire Protective Wrap Performance Testing***	Test Hemyc and MT fire wrap materials. These important tests conducted in 2005 showed there to be significant issues associated with these fire barrier materials.
Y6955	***Fire Incident Records Exchange***	Collect and analyze international fire events data. This is a long-term collaborative effort with OECD.

6 REACTOR FUEL RESEARCH

Reactor fuel is an important element of safety technology. NRC must maintain expertise in the area of reactor fuel because of both the importance to safety and because of the limited availability of expertise outside the agency that is independent of licensees. Research is an important vehicle for maintaining expertise in reactor fuel. NRC research on reactor fuel during normal operations and design basis accidents has been concentrated in recent years on the confirmation of regulatory decisions that allow licensees to take light water reactor fuels to burnups of nominally 62 GWd/t. This research has largely resolved the issue of the vulnerability of high-burnup fuel and cladding to reactivity transients though some confirmatory tests need to be completed. Research results will allow regulatory changes to better reflect the degraded capacity of high-burnup fuel to sustain reactivity insertion events.

The reactor fuel research has remained quite productive as examinations of high-burnup fuel behavior under loss-of-coolant accidents have been initiated. An important discovery has been the synergistic effect on clad ductility of hydrogen absorption during normal operation and steam oxidation of the cladding during an accident. Based on the research, revised embrittlement criteria have been developed that could be incorporated into 10 CFR 50.46.

The research on high-burnup fuel is reaching a substantial level of maturity. Some major confirmatory experiments remain to be done - notably experiments on reactivity insertion to be done in a water loop at the CABRI reactor. Plans for these experiments have been revised since our last report on reactor fuels research so the experiments which are part of an international collaborative effort now better meet agency needs. It is important that this work that is so well coordinated both with agency needs and with international partners be taken to completion. Still, major findings of the research effort can be reduced to regulatory practice now. This reduction to regulatory practice needs to be initiated and pursued aggressively.

It is evident that high-burnup fuel research will soon achieve results that are adequate for agency needs. The NRC has made clear that it will expect the nuclear industry to provide necessary safety analyses and experimental data should the industry want to take fuel to burnups that exceed the current regulatory maximum. NRC needs to make these expectations more explicit, particularly its expectations for the experimental data needed to support the analyses of high-burnup fuel behavior under accident conditions.

Completion of NRC's research on high-burnup fuel raises the question of how NRC will maintain expertise in fuel. Continued evolution in fuel cladding alloys can be anticipated. Interest is developing within the industry in fuels with enrichments exceeding 5% ^{235}U. These higher enrichment fuels may necessitate NRC research. If use of MOX fuel becomes more widespread than the planned disposal of excess weapons-grade plutonium, additional research on MOX fuel with reactor grade plutonium may be needed. Research on both higher enrichment fuel and MOX fuel can be done with substantial collaboration with international partners. Such collaboration will further the ideal of international safety evaluations of nuclear power plants.

Table 4. Reactor Fuel Research Activities

Job Code	Title	Comment
Y6586	***Fuel Code Assessment for MOX***	Improve FRAPCON and FRAPTRAN for calculating the behavior of MOX fuel rods; An important activity for licensing core loads for excess weapons-grade plutonium disposal.
Y6580	***Fuel Code Applications for High Burnup Fuel***	Improve FRAPCON and FRAPTRAN for calculating the behavior of high burnup fuel rods; an important activity as licensees press limits on allowable fuel burnup.
Y6788	***Halden Fuel Experiments Under Transient Conditions***	Data on fuel behavior under operational transient conditions for code development.
N6074	***STUDSVIK Cladding Integrity Project***	Stress corrosion cracking, hydride embrittlement and delayed hydride cracking study of ZIRLO clad. Defueled clad segments provided for NRC research.
Y6849	***ZIRLO Cladding Performance***	Adequacy of criteria for ZIRLO cladding performance in a LOCA; an important study of cladding used for high burnup fuel and the synergism between hydriding and oxidation on clad ductility.
Y6850	***M5 Cladding Performance***	Adequacy of criteria for M5 cladding performance in a LOCA; an important study of cladding used for high burnup fuel and the synergism between hydriding and oxidation on clad ductility.
G6923	***Failure of Hydrided Zircaloy under Severe Loading Conditions***	Develop theoretical model of mechanical failure of hydrided Zircaloy cladding.
W6832	***CABRI Water Loop***	NRC support for the CABRI water loop for RIA testing of high burnup fuel; confirmatory testing of high burnup clad and fuel vulnerability to reactivity transient events.
Y6367	***High Burnup Cladding Performance***	LOCA testing of high burnup cladding behavior; important study of cladding used for high burnup fuel and the synergism between hydriding and oxidation on clad ductility.
Y6723	***International Agreement on Fuel Behavior and Materials Science Research***	Data report on BIGR pulse reactor tests.
Y6847	***Clad Performance in ATWS***	Determine the adequacy of criteria and analysis of clad performance in BWR power oscillations; NRC needs to see if this problem can be solved by analysis with minimal experimental confirmation.
Y6195	***Dry Cask Storage License for High Burnup Fuel***	Develop criteria for dry-cask storage and transportation of spent high burnup fuel.

7 NEUTRONICS AND CRITICALITY SAFETY

Neutronics and criticality safety are areas in which NRC must maintain exceptional capabilities through its research program. The neutronics and criticality safety research program is small but appears adequate to ensure that the NRC has capabilities to meet immediately foreseen regulatory needs. The current NRC research activities in neutronics analysis, core physics, and criticality safety are listed in Table 5. Maintenance of the SCALE suite of codes is essential for the analysis of reactor core physics. These codes are complemented by the PARCS code which is part of the TRACE code and is discussed in more programmatic detail in the Chapter 14 of this report dealing with Thermal Hydraulics Research. The availability of the NEWT lattice code is important to licensees since it will be essential for the use of more advanced computer models in future regulatory processes. Currently, this lattice code is being used for the analysis of reactor cores fueled in part with MOX fuel for the disposition of excess weapons-grade plutonium. Several other activities are under way to support the licensing of MOX fuel core at the Catawba reactor for this plutonium disposition activity. These are appropriate programs at the current time. It is noted that NRC is taking advantage, to the extent feasible, of the considerable European experience with MOX fuel made with reactor-grade plutonium.

In the future, more innovative core designs for advanced reactors may be submitted to the NRC. Confirmatory analyses of reactor core physics will be an essential part of the regulatory process for these advanced reactors. The capabilities now available to the NRC in the area of core physics may well be stretched. It will be useful to the agency to understand these future needs. If long-term development activities are identified, such as those that might be needed for analysis of the PBMR, additional research may be needed in this area.

Table 5. Research Activities in Neutronics Analysis, Core Physics, and Criticality Safety

Job Code	Title	Comment
Y6846	***SCALE Code Development for Reactor Physics***	Essential code for neutronics analysis to audit licensee submittals and other regulatory needs.
Y6320	***NEWT Lattice Code***	Generate lattice cross-sections for safety analysis of MOX cores to support licensing of cores for Pu disposal.
N6162	***MOX Benchmark***	Confirmation of uncertainties in PARCS code predictions of MOX core neutronics; Also supports the licensing of Pu disposal activities.
Y6403	***Reactor Core Analysis***	Analysis to predict details of reactivity transient in MOX core. Again, this research supports regulatory activities associated with the DOE program to dispose of excess weapons-grade plutonium.
Y6685	***Experimental Data for High Burnup Spent Fuel Validation***	This project provides NRC with foreign and domestic data on high burnup fuel and MOX fuel for assessment of analytical tools used to predict fuel inventories, decay heating, and radiation shielding.

8. HUMAN FACTORS AND HUMAN RELIABILITY RESEARCH

Human performance plays a critical role in the safe operation of nuclear plower plants. Human performance issues have been main contributors to accidents and unsafe conditions experienced by the current fleet of operating reactors. They can be expected to continue to have a major impact on nuclear power plant safety. As licensees increasingly rely on risk-informed licensing applications that require the quantification of human reliability under accident conditions, the staff needs to be able to evaluate the treatment of operator actions in such applications. As new reactor designs, likely dependent on a higher degree of automation than the current fleet, are introduced, the need for revised guidance and tools for the NRC staff in human factors and human reliability analysis will increase. Therefore, it is very important that the NRC maintain research programs in these areas.

The current NRC research activities in the areas of human factors and human reliability analysis are:

- Human Factors
  (B7488, N6207, Y6843, N6137, Y6529)

- Human Reliability Analysis
  (Y6497, Y6496, N6248)

Current research in the human factors area includes a continuing international collaborative research program at the Halden project (B7488). The ACRS is supportive of this collaborative program and recommends continued NRC participation.

The project "Development of a Regulatory Guide and Analytical technique for Assessing NPP Staffing" (N6207) supports the development of guidance for staffing exemption requests to 10 CFR 50.54(m). This project is almost complete. Guidance is now provided in the recently issued NUREG-1791, "Guidance for Assessing Exemption Requests from the NPP Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)." Publication of this guidance is a significant accomplishment that provides a more flexible approach to staffing of current and future reactors.

Human performance issues, including organizational issues are of great importance to nuclear reactor safety. Inspectors at nuclear power plants currently have limited guidance or means with the Reactor Oversight Process (ROP) to characterize problems associated with human performance. This issue has been highlighted in a recent report from the Inspector General. In response to a Commission request, the project Y6843, "Develop Human Performance Indicators,"has been initiated to study the feasibility of establishing the technical bases for indicators of human performance that would be used to supplement indicators currently used in the ROP. This research is appropriate and very important. It may lead to significant improvements in the NRC inspection program and the ROP.

There is evidence of degrading performance of operations personnel in the nuclear and other industries due to operator overload. The research project N6137, "Impact of Operator Workload on Human Performance," is a five-year effort to assess the impact of operator overload on performance. The plan is to develop licensing requirements as well as inspection guidance and techniques for reviewing the impact of workload on operator performance and plant safety. This is an important new project that deserves support both for the current fleet of operating reactors and for advanced reactor designs.

Advanced reactor designs are likely to introduce much greater automation than exists in current reactors. Certainly, advanced digital control and instrumentation methods as well as new human-system interfaces can be anticipated. These new features of plants are likely to have some effects on human performance. The NRC staff needs to prepare itself to review new concepts and designs proposed by licensees. The project “Human Factors of Advanced Reactors” (Y6529) has been initiated to address this issue and to develop regulatory guidance and analytical techniques to review human factors for advanced nuclear power plants. The ACRS views this five-year project essential for preparing the staff in reviewing advanced reactor designs.

The quantification of human reliability continues to be a challenge in risk assessments. Many approaches to the quantification of human reliability have been proposed. However, the benchmark exercise conducted by the Ispra Laboratory of the European Union demonstrated that the choice of model has a significant impact on the results obtained. Not much progress to improve this situation has been made since that exercise was performed. The NRC staff has recently completed an assessment of the strengths and weaknesses of the various methodologies now available for assessing human reliability. The ACRS has been quite impressed with this assessment and hope the work leads to the identification of best methods for the quantification of human reliability in PRA.

Human reliability modeling introduces large uncertainties in PRAs. The NRC staff needs guidance in its review of the human reliability models used by the industry in licensing applications. The project Y6497, “HRA Application and ATHEANA Maintenance,” is intended to improve NRC’s ability to independently model human reliability and to provide guidance concerning risk-informed regulatory applications. Progress has been made with the publication of NUREG-1792, “Good Practices for Implementing Human Reliability Analysis (HRA).” Still, further guidance is needed for reviewers of licensing applications. The NRC has applied ATHEANA model to the human performance issues associated with its recent pressurized thermal shock study. The NRC is also planning to apply the ATHEANA model to a number of ongoing risk assessments, including those for fire and steam generator tube rupture to develop lessons learned on human reliability analysis and to develop guidance for the staff. If needed, modifications to the Standard Review Plan for licensee’s applications will be devised. The ACRS believes that this effort is needed. ATHEANA is a state-of-the-art model of human performance and is complicated to use. Application of the tool will show whether benefits derived from the analyses are commensurate with the enhanced complexity. Application may also show how the complexity of ATHEANA can be reduced. Application of ATHEANA is, however, very much behind schedule. Resources and management attention are needed to either accelerate the efforts or to revise the scope of the application efforts.

Both ATHEANA and SPAR-H (the HRA model used in SPAR) quantify the probability that a human unsafe act will be committed. This probability depends on a number of performance shaping factors (PSFs) that determine the context within the crew operates. The available time for action is one of the PSFs estimated from thermal-hydraulic considerations. The evaluated failure probability ia understood to be the probability that the required action will not be completed within the available time.

An alternative approach to HRA is to recognize the importance of time taken by the crew to complete a task and to develop a probability distribution for this time. The failure probability, then, is calculated from this distribution as the probability that this time will exceed the available time.

Recent experiments performed at Halden, Norway, have shown that there may be significant variability in the time that crews take to perform a given task. Such evidence is very difficult to account for in ATHEANA and SPAR-H. The alternative approach could accommodate such evidence. In addition, the staff is currently supporting research at Idaho National Laboratory (INL) that develops "time lines" for past accidents. This evidence can also be accommodated in the alternative approach.

The staffshould evaluate the merits of an HRA model that focuses on the time required for action.

The project Y6496 is a continuing effort to develop an event database called Human Event Repository and Analyses. This database and analysis capability should significantly improve the treatment of human reliability in nuclear reactors and provide a realistic, performance-based database to assess licensee's quantification of human performance. This effort should be sustained and made an ongoing part of the research program.

The project N6248, "Advanced Reactor HRA Development," is the first year of a proposed five-year effort to develop HRA methods and tools to support an independent staff review of human reliability analyses submitted as part of new reactor licensing applications. Given the importance of human factors to reactor safety and the likelihood that new reactor designs may significantly alter the role of operators and the human-system interface, this project is valuable and should be continued to completion.

Table 6. Human Factors and Human Reliability Research Activities

Job Code	Title	Comment
Y6497	***HRA Application and ATHEANA Maintenance***	Apply ATHEANA to Fire Risk Requantification; upgrade and improve ATHEANA. ATHEANA is NRC's tool for analysis of human reliability. Application of this tool will allow assessment of its worth.
Y6496	***Human Event Repository and Analysis***	Develop a human event repository and analysis tools. This program develops a useful data-base for comparison to model predictions of human events.
B7488	***Halden Reactor Project***	International collaborative research project that addresses man-machine interaction and verification and validation of software, surveillance and support systems, advanced control rooms and fuels and materials. This international effort helps keep staff aware of international developments in human factors and human reliability.
N6207	***Develop Reg. Guide and Analytical Technique for assessing NPP staffing***	Support development of guidance for staffing exemption requests to 10 CFR 50.54 (m). This is an important program as licensees look at manpower costs associated with nuclear power plant operations.
Y6843	***Develop Human Performance Indicators***	Determine availability and viability of human performance indicators for assessing performance at nuclear power plants; This program was undertaken in response to a Commission SRM.
N6137	***Impact of Operator Workload on Human Performance***	An important new effort to assess the impact of operator overload on operator performance and plant safety.
N6248	***Advanced Reactor HRA Development***	The first year of a proposed five -year effort for addressing human performance issues for new reactors. This is a valuable project and should be continued to completion.
Y6529	***Human Factors of Advanced Reactors***	Develop regulatory guidance and analytical techniques to review human factors for advanced reactors. Essential work to prepare the staff in its review of advanced reactor designs.

9 MATERIALS AND METALLURGY

Research in the area of materials and metallurgy is an important focus of the NRC Safety Research Program. Current research activities are concentrated in five areas:

- Environmentally Assisted Cracking in Light Water Reactors
(Projects K6266, K6202, Y6270, Y6388, N6007)

- Steam Generator Tube Integrity
(Projects Y6536, Y6588)

- Non-destructive Examinations
(Projects Y6534, Y6604, Y6649, Y6869, Y6867, Y6541, N6019)

- Proactive Materials Degradation Assessment
(Project Y6868)

- Reactor Pressure Vessel Integrity
(Projects W6953, Y6533, Y6378, Y6638, Y6951, N6204, Y6870, N6223, Y6485, Y6656)

These projects represent a significant investment by the NRC to better understand the issues of materials degradation in the currently operating fleet of nuclear power plants. Such investment is justified in view of significant agency regulatory activities that aging degradation research supports. As plants age, known degradation mechanisms will continue to affect components and new degradation mechanisms may develop. The current program is well focused on improving the agency's ability to independently evaluate licensees' efforts to prevent, detect, and mitigate environmentally assisted stress corrosion cracking. The Proactive Materials Degradation Assessment project is an effort to identify potential material degradation problems before they manifest in operating nuclear power plants.

Unfortunately, the planning of NRC's research in materials and metallurgy is not well documented in the way planning for research on digital instrumentation and control systems has been documented. It is, then, difficult to explain the role and priority of each task within each of the five project areas. In aggregate, the activities in the first four project areas (Environmentally Assisted Cracking, Steam Generator Tube Integrity, Non-destructive Examinations, and Proactive Materials Degradation Assessment) seem to be appropriate. These are the very areas that most challenge the industry and its ability to detect component degradation. The agency must develop the capabilities to assess the acceptability of the industry's initiatives to deal with these degradation challenges. The five project areas are further discussed below.

Environmentally Assisted Cracking

Environmentally assisted cracking is a complicated technical issue that continues to afflict the industry as components age and irradiation effect increases. In recent years, the industry has experienced irradiation assisted stress corrosion cracking (IASCC) of components internal to the vessels of boiling water reactors (BWRs) and stress corrosion cracking of reactor vessel head penetration assemblies in pressurized water reactors (PWRs). Although the industry has responded to these events with initiatives to prevent and mitigate these types of degradation, the event at Davis-Besse makes it readily apparent that the NRC staff must be capable of independently evaluating the adequacy of licensees' initiatives. The research projects now under way seem well designed to ensure that the NRC has the needed technical understanding of the stress corrosion cracking issues.

The project Y6388, "Environmentally Assisted Cracking of LWRs," evaluates environmental effects on fatigue of steels used in light water reactors and provides the NRC with technical data and analytical methods to assess licensees' plans concerning mitigation. The large effort includes tests of neutron-irradiated specimens to improve the understanding of IASCC initiation and stress relaxation. It also provides data on the performance of probes and monitoring techniques in radiation environments. This work is essential and should be continued. A new project, "Investigation of Stress Corrosion Cracking in Selected Materials" (N6007), will develop a better understanding of stress corrosion cracking in PWRs. Such cracking occurs typically in the reactor coolant system boundary. Understanding of such cracking in this boundary is essential for maintaining the defense-in-depth.

Environmentally assisted corrosion of reactor materials is an international concern. The CIR-II Cooperative Agreement (K6202) is a collaboration with the international community for studying the susceptibility of stainless steel to IASCC. Certainly, this collaboration should be continued.

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 at some PWRs. Through the years, many modes of corrosion of steam generator tubes have been experienced. Regulations on the corrosion were developed when erosion was the dominant concern. Careful water chemistry control by licensees has largely eliminated erosion as a safety concern. But, now, stress corrosion cracking has emerged as the dominant threat to the integrity of steam generator tubes. Incipient stress corrosion cracking is much more difficult to detect. NRC has two research projects to deal with the degradation mechanisms in steam generator tubes, "Steam Generator Tube Integrity Program" (Y6588) and "PWR Primary System Components Severe Accidents" (Y6536). The first project, Y6588, deals with potential tube degradation modes, their resulting leak rates, and the effectiveness of in-service inspections. The second project, Y6536, seeks to improve methods and models used to predict the behavior of degraded steam generators and other PWR components under severe accident loads. Both of these research efforts are important and should be continued.

Non-destructive Examinations

Non-destructive examinations are relied upon to monitor the integrity of the reactor coolant system. The reliability and effectiveness of existing non-destructive examination techniques remain open to question. Certainly, a steam generator tube cracking incident at the Indian Point reactor emphasizes this point. Four projects are under way to improve non-destructive examination techniques (Y6534, Y6604, Y6649, and Y6869) and this work should continue. Two of these projects deal with the effectiveness and reliability of non-destructive examination of reactor vessel penetration assemblies. As the ACRS noted in NUREG-1635, Vol. 6, this is an area that needs increased attention. A third project will provide destructive examination data that should be of tremendous value for the validation of non-destructive examination methods. The project, N6019, will examine non-destructive methods and leak monitoring techniques and the requirements for light water reactor components that have experienced degradation or have been identified as being susceptible to future degradation. The project "Evaluate Reliability and Effectiveness of Advanced NDE," Y6541, will support continued investigation of innovative methods to detect incipient amounts of wastage of ferritic steel. All of 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 problems. As a result, they have responded to such problems in a reactive mode which has proven to be inefficient. Reactive response does not enhance public confidence in the safe operations of nuclear power plants. The project "Proactive Material Degradation Assessment" (Y6868) is an NRC initiative to identify materials and locations in light water reactors where degradation can reasonably be expected in the future. The goal of this project is to develop the technical bases needed to implement regulatory actions to proactively address materials degradation problems. Current inspection and monitoring programs at plants can be reviewed and modified as needed to provide earlier identification of incipient degradation before it affects plant safety. The ACRS admires the vision of this undertaking and supports its continuation. The ACRS looks forward to reviewing the initial results of this ongoing effort soon and learning whether the admirable goal of this project is, in fact, feasible.

Reactor Pressure Vessel Integrity

The integrity of the reactor pressure vessels has been studied for decades. Maintaining the structural integrity of the reactor pressure vessel in a nuclear power plant during both routine operations and during postulated upset conditions, including pressurized thermal shock situations, is a longstanding obligation of licensees. This obligation is 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. Technical bases for these requirements were largely established in the 1980s. NRC is continuing to devote substantial resources to the study of pressure vessel embrittlement though there does not seem to be a comparable interest within the industry who will have most of the research benefits. Indeed, the number of projects in this area seems to have grown since the ACRS last reviewed the NRC research program and questioned the need for research in the area of reactor pressure vessel integrity.

Some of the activities in this programmatic area deal with the finalization of the NRC's work on pressurized thermal shock which is nearing completion. These activities will contribute to the potential revisions of Regulatory Guide 1.99 on radiation embrittlement of reactor pressure vessel materials and Appendices G and H to 10 CFR Part 50 on fracture toughness requirements and reactor surveillance needed to ensure low probability of reactor vessel failure.

The project "International Pressure Vessel Technical Cooperative Program" (Y6378) will ensure NRC participation in the International Atomic Energy Agency (IAEA) deliberation on reactor pressure vessel integrity.

The NRC's comprehensive program on reactor pressure vessel integrity has produced significant results by providing better understanding of the available margin in reactor pressure vessel components. Revisions to 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 to licensees by relaxing current requirements and allowing longer life of reactor pressure vessels. These activities should be completed soon.

RES needs to reevaluate the need for continued research into heavy section steel components. This research may be justified if there is a clear need for NRC to develop its capabilities in the area of probabilistic fracture mechanics so that it can evaluate licensees' applications. If this is the case, the research needs to be clearly focused on this objective and not the research that the industry should perform to meet its responsibilities to ensure reactor pressure vessel integrity. It appears now, however, that it is NRC that is advancing the state-of-the-art and making available information that allows licensees to reduce conservatism in their analyses.

Table 7. Research Activities in Materials and Metallurgy

Job Code	Title	Comment
Environmentally Assisted Cracking in LWRs		
K6266	***CIR-II Cooperative Agreement***	NRC contribution to international research on irradiation assisted stress corrosion cracking.
K6202	***Extension of CIR-II Cooperative Agreement***	Assess the susceptibility of stainless steels to Irradiation Assisted Stress Corrosion Cracking. This program allows NRC to stay abreast of international developments.
Y6270	***Environmentally Assisted Cracking***	Provide neutron irradiated specimens for NRC research programs.
Y6388	***Environmentally Assisted Cracking of LWRs***	Develop data on irradiation assisted stress corrosion cracking in PWRs and BWRs. This program provides NRC staff with the data and analytical methods to review licensees' activities and plans to limit corrosion.
N6007	***Investigation of Stress Corrosion Cracking in Selected Materials***	User need for a better understanding of stress corrosion cracking in PWRs. This program supports the regulatory process.
Steam Generator Tube Integrity		
Y6536	***PWR Primary System Components Severe Accidents***	Methods and models to predict PWR reactor coolant system component behaviors under severe accident loads; This is an essential research program.
Y6588	***Steam Generator Tube Integrity Program***	Wide-ranging program in support of the steam generator integrity action plan. ACRS supports this action plan and regularly monitors its progress.

Table 7. Research Activities in Materials and Metallurgy (Continued)

Job Code	Title	Comment
Non-destructive Examinations		
Y6534	***Piping NDE Reliability***	Program addresses Inconel cracking in weld metal and base metal. This is an essential program to ensure licensees adequately monitor nickel alloys in plants.
Y6604	***Evaluate Reliability of NDE Techniques***	Addressing the inspection of cast stainless steel components and dissimilar metal welds; evaluation of reliability and accuracy of in-service inspection. This is an essential program to facilitate NRC monitoring of licensee activities.
Y6649	***Phase II - Alloy 600 Cracking***	Independent assessment of industry analyses of CRDM nozzle cracking. This is a classic NRC program of confirmatory research.
Y6869	***Barrier Integrity Research Program***	Evaluate RCS leakage experience and leak detection capabilities. This is an essential program to facilitate NRC monitoring of licensee activities.
Y6867	***Cooperative Activities Reactor Coolant System Pressure Boundary Components***	Complete non-destructive examinations of nozzles from vessel heads. Plan destructive tests. This is an important program to validate analyses NRC uses in its regulation of licensee activities.
Y6541	***Evaluate Reliability and Effectiveness of Advanced NDE***	Identify innovative NDE techniques in coordination with industry and international community. This program allows NRC staff to stay abreast of international developments in NDE.
N6019	***NDE & Leak Monitoring Requirements***	Assess adequacy of current inspection and monitoring requirements. Assemble data on probabilities of failure of passive components. This is an essential program to facilitate NRC monitoring of licensee activities.

Table 7. Research Activities in Materials and Metallurgy (Continued)

Job Code	Title	Comment
Proactive Materials Degradation Assessment		
Y6868	***Proactive Materials Degradation Assessment***	Identify materials and locations in LWRs where degradation can reasonably be expected. This program is intended to better equip NRC to anticipate materials degradation problems at nuclear power plants. This program should be continued. The ACRS looks forward to reviewing the initial results.
Reactor Pressure Vessel Integrity		
N6204	***Review and Revisions of Pressurized Thermal Shock Reports NUREGs 1806 and 1809***	Support documentation of thermal hydraulics analyses for pressurized thermal shock, and document Calvert Cliffs RELAP5 calculations to support FAVOR calculations. This program should be completed.
Y6485	***Technical Support - Pressurized Thermal Shock Rulemaking***	Support for the pressurized thermal shock rulemaking effort. This is essential support for the regulatory process.
W6953	***Heavy-Section Steel Irradiation Program***	Evaluation of Master Curve methodology for reactor pressure vessels. The ACRS questions the need for the large investment in heavy section steel research.
Y6870	***Cooperative Program on Irradiation***	Development of a cooperative program with DOE to study reactor pressure vessel materials.
Y6378	***International Pressure Vessel Technical Cooperative Program***	International cooperative effort to understand embrittlement of reactor pressure vessels and other components. This program will keep staff aware of international developments in reactor pressure vessel integrity.
Y6533	***HSST-3 (Heavy Section Steel Technology)***	Development of fracture mechanics methodologies; The ACRS questions the need for the large investment in heavy section steel research.

Table 7. Research Activities in Materials and Metallurgy (Continued)

Job Code	Title	Comment
Y6951	***Fracture Mechanics Technology for LWR***	Fracture mechanics of heavy section steel. The ACRS questions the need for the large investment in heavy section steel research.
Y6638	***Statistical Analysis of RPV Steels***	Assist NRC staff in developing a revision to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials." This research directly supports the regulatory process.
N6223	***FAVOR 4.1 Sampling Validation***	Validation of new features of the FAVOR computer code for fracture analysis of vessels. FAVOR is NRC's computer code for fracture mechanics analysis and is used extensively.
Y6656	***Risk Inform Appendices G & H***	Develop a risk-informed revision to 10 CFR 50, Appendix G on Fracture Toughness Requirements and Appendix H on Reactor Vessel Material Surveillance Program.
N6227	***SMIRT-18 Conference Registration***	Costs associated with presentation of papers on NRC research projects at the Structural Mechanics in Reactor Technology meeting.
N6097	***SMIRT 18***	Financial support to publish proceedings of the 18th International SMIRT conference.

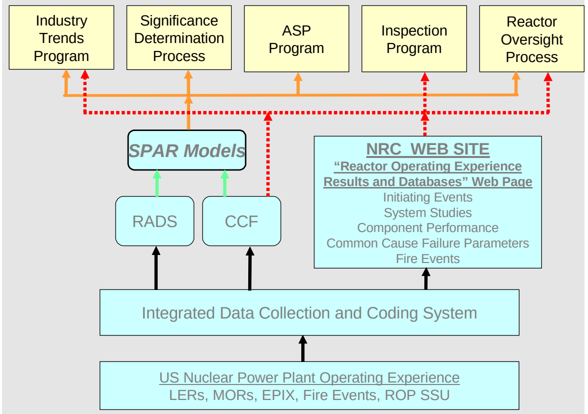
10 OPERATIONAL EXPERIENCE

The analysis of operating data is a cornerstone in the NRC's increased use of risk information in regulatory processes. Such analysis provides current information on initiating events, component failure data, and the risk profiles of licensees. Comparison of these results to goals in the agency's Strategic Plan provides a measure of regulatory effectiveness and inputs for the agency's annual report to Congress on significant operating events.

The NRC research activities associated with operational experience are listed in Table 8. The Accident Sequence Precursor (ASP) Program, Y6815, and the Industry Trends Program, Y6546, alert the staff and industry to component failures as old or replacement components age or operations change. Data derived from operating experience will validate or refute the assumption that aging management programs are sufficient to ensure the operability of both active and passive components. The operating experience programs provide data that can be the bases for regulatory decisions to improve safety. These programs also support the Reactor Oversight Process, including the determination of the safety significance of inspection findings and the development of industry performance indicators.

Two tasks in the research of operational events, "Method to assess Effect of Design and Operations Margins," N6082, and "Procedure Development for External Events," Y6814, are important efforts to extend the use of quantitative risk assessment into external events, including fire, and low power and shutdown operations.

ACRS is supportive of the research activities in the area of operational experience and recommends that these activities be continued. In light of the limited resources



Uses of Operational Data and Analyses in Regulatory Activities

allocated to these tasks, RES has done a commendable job in producing outputs in well-documented and thorough fashion. Tasks that are currently in the 2005 Research Plan related to Operational Experience should remain funded and should be continued for the foreseeable future.

Staff engaged in the collection and analysis of operating experience data might also be able to improve the state-of-the-art in PRA modeling. Specifically, they might be able to use operating experience data to derive higher resolution models of system and component operability. Currently, PRAs use success criteria models. A system or component that meets the success criteria is deemed operable This "go/no go" model is not entirely realistic. There is no assessment of margins, equipment aging, changing plant conditions, etc. Success criteria models may not provide adequate answers for some applications such as power uprates, containment overpressure credit, license

renewal, sump screen clogging, or any set of plant conditions that are in some way off-normal or even outside the design specifications of the equipment. There have been several events that were surprises because the phenomena that caused or contributed to the failure mode had not been realistically modeled. Certainly, the recent Davis-Besse event involving corrosion of the reactor pressure vessel head penetrations comes to mind. Staff granted a small extension to ordered shutdown date for reactor pressure vessel penetration inspections. They did so, in part, because the calculated risk was small. Unfortunately, the phenomenological modeling of the head penetrations and their corrosion was incorrectly used in the risk assessment.

Development of improved models of system and component operability models will require that choices be made concerning areas where improved modeling will yield useful improvements in the risk predictions. The issues of interest may themselves dictate where choices for improved modeling should be made. Some modeling improvements are being made now on an *ad hoc* basis. There is no need to continue to do so if a more structured approach could result in better models with wider applications.

Table 8. Research Activities in Operational Experience

Job Code	Title	Comment
N6082	***Method to Assess Effect of Design and Operations Margins***	Provides a methodology to assess the effects of changes to design and operation on plant safety margins. This program provides direct support for the regulatory process.
Y6468	***Reactor Operating Experience Data for Risk Applications***	Collect operational data for reactor systems, components, initiating events, common-cause failures and fire events. Data collected in this program is of use for validation of PRA models.
Y6546	***Industry Trends Program***	Includes grid concerns. This is an essential program for NRC.
Y6864	***Operating Event Technical Support***	Support for technical expertise in operating events.
Y6816	***SDP/ASP Standardization***	Develop analysis guidelines for operating events during low power/shutdown conditions. This program will extend the ASP program to include events during shutdown operations.
Y6815	***Accident Sequence Precursor Analysis***	Systematically screen, review and evaluate operating events. This is a flagship program at NRC.
Y6987	***Expert Elicitation Process - Accident Sequence Precursor Program***	Develop guidelines for obtaining and using expert opinion in ASP analyses. The useful elicitation of expert opinion is of growing importance in the risk-informed regulatory system.
Y6814	***Procedure Development for External Events***	Expand the scope of ASP analyses to include the calculation of risk from external events and from low power and shutdown modes of operation. This program will help extend the scope of the ASP program.

11 PROBABILISTIC RISK ASSESSMENT

Probabilistic risk assessment has become an essential technology for NRC as it evolves the regulatory system to make greater use of risk information. The NRC research activities in probabilistic risk assessment are shown in Table 9. Probabilistic risk assessment has become pervasive within the research program. Other activities nominally part of the development of PRA are addressed in other Chapters of this report. See especially the discussions of Digital Instrumentation and Control Systems (Chapter 4), Fire Safety Research (Chapter 5), Human Factors and Human Reliability Research (Chapter 8), and Operational Experience (Chapter 10). The staff involved in PRA research has been extraordinarily productive since the last ACRS report the NRC research program. A major focus of the current PRA research is to support the ROP, which uses risk information for monitoring the operations of nuclear power plants and acting on inspection findings and deviation of performance indicators from established thresholds.

The ROP makes heavy use of the SAPHIRE computer code and the SPAR models of specific plants. The SPAR model development program has become an essential element of the ROP. The ability to develop a SPAR model for each nuclear power plant has only been feasible because of the existence of Level I, internal events, PRAs for each plant**.** Each SPAR model begins with a basic model of a plant system for a generic category of plants (e.g., a BWR4 reactor with a Mark I containment). The SPAR model is then made plant specific through upgrades based on discussions with the licensee. NRC has found it essential to develop its own risk-assessment model for each plant as a matter of practicality. It would be difficult for the NRC staff to take a variety of plant PRAs, which use different platforms and approaches, make them operational at NRC, and have knowledgeable staff available to execute and update each plant model. NRC development of SPAR models for individual plants has also enhanced the plants' risk assessments.

A major issue that confronts the use of risk information in nuclear power plant regulation is the question of incompleteness of individual plant risk assessments. The Individual Plant Examination (IPE) program and subsequent evolutions at the nuclear power plants led to development of Level I, internal events, PRA models of all of the operating. These PRAs meet (or with modest effort can meet) the requirements of industry standards for internal events PRAs. The same is not true for the assessment of risk from fires, floods, seismic events and for plant modes of operation that differ from full power operations. Furthermore, the capabilities to assess risk at Level II, radionuclide release and source terms , lag far behind the Level I capabilities.

The NRC staff has plans to expand the scope of the SPAR models to include treatment of risks from fire-initiated events, seismic events and shutdown modes of operations. These plans are, however, not well developed. There is furthermore the question of availability of resources needed to undertake these efforts. The expansions of the scope of SPAR models will be challenging because all licensees do not have sophisticated risk assessments in these areas for comparison and validation of NRC's SPAR models with expanded scope. The NRC staff could develop generic models accounting for the major features of the plant designs, but the staff would not be able to upgrade the generic models to become plant-specific models as was done for the treatments of risk from internal events. In addition, fire and seismic risk assessments differ qualitatively from internal events risk assessments since the events occur in "areas" of a plant and affect multiple systems rather than just specific components in specific systems. Fire and seismic risk assessments require detailed knowledge of spatial relationships in addition to functional relationships. Spatial relationships, of course, vary substantially even among plants of the same generic type. Despite these challenges, the regulatory oversight value of full-scope SPAR models is very high. Over the next year, the staff should develop its approach and plans for the expansion of the scope of the SPAR models to treat external events, shutdown modes of operation and even to go to Level II analyses that include accident progression and the release of radionuclides to the environment. Even if it is not possible to have plant-specific models in the near term, the generic shells should be available and can be adapted to be plant specific in the future or can be upgraded in particular areas to address specific regulatory issues.

Another barrier to the greater use of risk assessment in the regulatory process is the question of uncertainty in the risk predictions. There are, of course, parametric uncertainties and the agency has active programs to better understand the important parametric uncertainties (See especially Chapter 10, Operational Experience). There are also issues of uncertainty in the models adopted in PRA. Uncertainties in the models of human reliability and passive system reliability are significant examples. It has become common now for the NRC and the licensee to agree upon a model appropriate for particular regulatory activities. This agreement can often be based on familiarity or expedience. The disturbing trend is for the staff to conclude, then, that there are no longer uncertainties associated with the results predicted by the agreed upon models. Staff needs to ensure that it treats uncertainty in risk assessments in a more defensible manner. Research needs to provide the tools and understanding so that this can be done.

The staff has also been revising 10 CFR 50.46 to account better for risk information. This is challenging and important work. Even more challenging is the effort to develop a "technology-neutral" alternative to the current regulatory framework. The ACRS views such a technology-neutral regulatory framework as essential in the future and feels that it needs more attention.

Altogether the scope and the number of PRA research activities are quite impressive. The ACRS cautions, however, that NRC should not allow its work in such a crucial technology as risk assessments become totally devoted to the support of line activities. Methods development is still important. As an example, the ACRS notes that considerable research is being reported in the literature regarding Binary Decision Diagrams as tools for solving large fault tress without resort to cutoff frequencies as is now done. Some researchers report that the unavailability of highly redundant systems could be underestimated significantly when cutoff frequencies are used for the analysis. Although no definitive evidence has yet been produced to show that methods used in the NRC's SAPHIRE code are inadequate, the staff needs to review the literature concerning Binary Decision Diagrams and evaluate the need to adopt this technology. The growing importance of the SAPHIRE code and the SPAR models in the regulatory process warrants such an investigation.

Table 9. Probabilistic Risk Assessment Research Activities

Job Code	Title	Comment
N6027	***PRA for Dry Cask Storage Follow Up***	A variety of tasks including uncertainty analysis and extension to multiple casks. This program supports licensing and inspection oversight of cask vendors.
N6105	***Guidelines for the Communication of Risk Information***	Complete the technical basis for the internal risk communication guidelines. This task completes the technical basis for internal risk communication guidelines. The ACRS remains concerned that publically available information on risk analyses may not be sufficient to ensure public confidence in a risk-informed regulatory process.
Y6842	***Guidance for the Development of Latent Errors***	Quantitatively assess the importance of latent errors and the treatment of latent errors in PRAs. This project has been deferred until FY2007. The ACRS cautions that operating experience shows that latent errors may be four times more common than active errors in important reactor events. The work should not be deferred further.
J8263 **Y6370**	***Reactor Oversight Process Support*** ***Development of Risk-based Performance Indicators***	Development of performance indicators to be incorporated into the ROP. Support for the Mitigating Systems Performance Index. These programs support the ROP.
Y6626	***Access to INPO's EPIX System***	Data-base on equipment performance and reliability.
J8258	***International Common Cause Exchange Project***	Sharing of data on common-cause failures with the international reactor safety community. This program keeps staff abreast of international findings concerning common-cause failures.
N6008	***Passive Components Conditional Core Damage Probability***	This program should prioritize passive components for consideration in the proactive materials degradation assessment (Project Y6868, Materials and Metallurgy, Chapter 9).

Table 9. Probabilistic Risk Assessment Research Activities
(Continued)

Job Code	Title	Comment
Y6153	***SPAR Model Development: Level2/LERF***	Develop SPAR models for evaluation of large early release frequencies.
N6090	***SPAR Model Development: Shutdown Models***	Develop logic models for analyzing low power and shutdown internal events.
W6355	***SPAR Model Development: Low Power Shutdown***	Identify methods to characterize risk during low power or shutdown operations.
W6467	***SPAR Model Development: Level 1 Rev. 3 Models***	Revision of Level 1 SPAR models to better reflect as built and operated plants.
Y6595	***SPAR Model Development: External Events Analysis***	Development models of external events for the SPAR codes
N6075	***SPAR Model Development: Enhanced Level 1, Revision 3 Models***	These are important programs to support the expanded scope of the SPAR models.
Y6394	***Maintain and Support SAPHIRE Code and Library of PRA***	Testing to ensure that SAPHIRE is a state-of-the-art PRA code.
N6172	***Participate in the MERIT Program (Maximizing Enhancements in Risk Informed Technology)***	Base program supports risk informing 10 CFR 50.46 and includes development of a probabilistic LOCA code, non-piping component degradation, and pressurized water stress corrosion cracking. This international program supports one of the important NRC initiatives.

Table 9. Probabilistic Risk Assessment Research Activities
(Continued)

Job Code	Title	Comment
N6111 **Y6538**	***Technical Support for 10 CFR 50.46 Task Order 3*** ***Technical Development of LOCA Frequency Distributions***	Quantification of the effect of break size reduction and alternative break locations on margin to existing alternate acceptance criteria Provide LOCA frequency estimates for use in revision of 10 CFR 50.46. These programs are needed to support risk informed revisions to 10 CFR 50.46.
K6081	***PRA Techniques in Risk-informed and Performance-based Regulation***	Develop methods for uncertainty analysis for risk-informed purposes. This is a cooperative agreement with a broad scope. In addition to potential methodological contributions it has an educational value.
N6107	***10 CFR 50.48c related Technical Activities***	In collaboration with EPRI, develop a comprehensive set of risk methods, tools and data to understand and evaluate risks from fires.
W6224 **Y6492**	***Risk-informing Part 50*** ***Assess Possible Part 50 Risk-informed Changes***	Develop recommendation on changes to 10 CFR Part 50 to make it risk-informed. Develop recommendations to specific requirements in 10 CFR Part 50 to make them risk-informed. These program support the initiative to risk inform 10 CFR Part 50.
W6970 **W6971**	***Support to Develop Consensus PRA Standards*** ***Support in Development of Consensus PRA Standards***	Provide guidance on the use of industry standards for PRA. Revise Regulatory Guide 1.200 based on industry pilots and Revision 1 to ASME PRA standard. These program support the Commission's phased approach to PRA quality.

Table 9. Probabilistic Risk Assessment Research Activities
(Continued)

Job Code	Title	Comment
Y6103	***Low Power and Shutdown Risk Study - Level 2***	Program to extend the scope of SPAR models to include accident progression for accidents initiated during shutdown operations. Premature at this point.
N6133	***Development of Consensus on PRA***	Support for staff in development of ANS Low Power and Shutdown operations PRA Standard.
N6134	***Low Power/Shutdown Level 1 and Fire Risk Standard***	Project provides support for staff involvement in the development of ANS standards on PRA for low power/shutdown operations and fire-initiated events.
Y6371	***Risk Associated with Cable Aging***	Addresses the inclusion of aging effects into PRA.

12 SEISMIC RESEARCH

As the design of nuclear power plants improves, the seismic hazard and seismic response of the plants can make an increasingly important contribution to risk. Seismic hazard analysis and structural response are not areas where NRC must maintain state-of-the-art expertise. Such expertise is available to the NRC on a contractual basis. As noted in our previous report, seismic research activities at NRC can be confined to support needed updates to regulatory guides and collaborative work with the international community to stay abreast of developments in other Countries. The current research program is, indeed, largely focused on needs of the regulatory process and a few important international collaborations.

Table 10. Seismic Research Activities

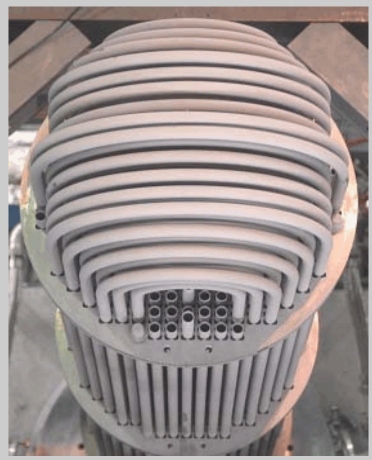
Job Code	Title	Comment
N6020	***Seismic-induced Passive Component LOCA Frequencies***	Review of work by national laboratories and industry on piping degradation and failure under earthquake loads; Work being done to upgrade Regulatory Guides.
Y6481	***SSHAC Method***	10-year update of the Probabilistic Seismic Hazard Assessment used in evaluation of early site permits; work to support update required by regulations.
Y6718	***Soil-structure Interaction for Buried Structures***	Review adequacy of current NRC guidelines concerning soil-structure interactions; work to update Regulatory Guides.
N6112	***Evaluation of Seismic Siting***	Review of ASCE Standard 43-05, "Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities."
N6076	***Japanese Collaboration on Seismic Issues***	Collaboration with Japan on seismic tests and analyses; Collaborative work give NRC access to extensive work underway in Japan.
W6081	***Japanese Collaboration on Seismic Issues***	Supports work in U.S. in connection with collaboration.
N6102	***Reg. Guide 1.165 Update Technical Basis***	Review of technical advances in the development of seismic response spectra; prepare draft revision to Regulatory Guide 1.165.
N6103	***Enhancement of the CARES Code (Computer Analyses for Rapid Evaluation of Structures)***	The CARES computer code is used to predict the free field and structural response to seismic input.
N6219	***Resolve Regulatory Guide 1.92 Public Comments***	Regulatory Guide provides up-to-date guidance for using the response spectrum and time history methods for estimating seismic response of power plants.
N6104	***Ground Motion Seismic Hazard Studies***	Collection and review of new data on the propagation of earthquake motion in the Central and Eastern U.S.; work to support required update in regulations.
Y6796	***IAEA Coordinated RES Project on Seismic Ground Motion***	NRC contribution to international effort to understand earthquake effects on nuclear power plants. Collaborative effort keeps NRC staff abreast of any international developments.
Y6757	***Containment Capacity Studies***	Confirmatory analyses of structural response and failure modes of containments under extreme loading including seismic loads.

13 SEVERE ACCIDENT RESEARCH

In the past, NRC invested heavily in the experimental and analytical characterization of severe reactor accidents. A substantial technology has been established to understand the progression of severe reactor accidents and the radiological consequences of such accidents. Once its immediate needs were met to understand severe reactor accidents sufficiently well to estimate risks to the level of confidence needed to provide assurance of adequate protection, the NRC substantially curtailed its investments in severe reactor accident research. The current NRC research activities in the severe accident area are listed in Table 11.

Research on severe accidents has been continuing in other countries. Substantial programs are under way in both Europe and Japan. NRC has developed an effective strategy to maintain the technology for severe accident analysis and to update this technology with research results from international programs. The body of knowledge coming the NRC's past work and the ongoing international work are systematized in the useable form in the MELCOR accident analysis code. At the same time, the NRC is entering into international cooperative research programs to obtain data for validating the MELCOR code and improving its accuracy and realism. NRC provides the Cooperative Severe Accident Research Program (CSARP) as a forum for the exchange of severe reactor accident information among Countries. One outcome of this focus of the NRC's research into severe reactor accidents is that many Countries and institutions have adopted the MELCOR code as the preferred tool for the severe accident analysis.

A new version of the MELCOR code has been released to users. NRC is collaborating with researchers in Russia to modernize MELCOR to use FORTRAN 95 coding. MELCOR is being used for licensing actions. The capabilities developed to perform detailed



Aerosol Trapping in a Steam Generator (ARTIST)

NRC is participating in ARTIST international cooperative research program to conduct an experimental study in Paul Scherrer Institute in Switzerland to measure the aerosol removal on secondary sides of steam generators during severe accidents at PWRs that bypass reactor containments. Such bypass accidents are often risk dominant for PWRs. The high risks associated with such accidents may stem from conservatism in the aerosol decontamination assumed in accident analysis models for steam generators. Test results are expected to provide the basis for more realistic analyses of these accidents.

parametric uncertainty analyses with the code are especially attractive.

RES is also maintaining the MACCS code for the analysis of consequences of accidents at nuclear facilities. This code is widely accepted in the U.S. as a tool for consequence analysis. Its maintenance at near the state-of-the-art is important to the agency and the ACRS is supportive of the current research programs.

Collaborative severe reactor accident research programs that NRC has joined are making good technical progress and there have been notable accomplishments in the last 2 years.

- **PHÉBUS - FP**

The Phébus-FP program consists of large-scale prototypic experiments involving the degradation of irradiated reactor fuel, release of fission products as vapors and aerosols, and transport of these fission products through a model of a reactor coolant system into a model of a reactor containment. These are the most prototypic and most comprehensive severe accident experiments that have ever been performed. The last of these tests was completed recently. The experiments have proved to be invaluable for the validation and improvement of the MELCOR code and the validation of the alternative source term used for a large number of licensing actions. The program has revealed a number of unanticipated phenomena and refined understanding of other phenomena. NRC has joined a second-generation program that will involve about 15 Nations to conduct separate effects tests to further understand the important accident phenomena revealed in the PHÉBUS-FP test program. This follow-on program addresses the containment chemistry of radioactive iodine, fission product chemistry in the reactor coolant system, the effects of boron carbide control rods on core degradation and fission product chemistry, and the release of fission products from high-burnup fuel and MOX fuel.

- **ARTIST**

The ARTIST test program is an international collaborative effort undertaken in Switzerland to ascertain the amount of decontamination that can occur in the secondary side of steam generators in PWR accidents initiated by steam generator tube ruptures or initiated by other means but involving steam generator tube ruptures. Such accidents have been found to be risk dominant for some PWRs. During last year, the scoping test program has been completed. Results of the tests show that decontamination is modestly larger than what had been anticipated in accident analyses. Plans are being formulated now to conduct integral system tests and additional tests to support modeling of secondary side decontamination.

- **MASCA**

The MASCA test program and its predecessor the RASPLAV program were undertaken to understand the technical feasibility of retaining core debris within reactor pressure vessels, especially with water flooding the outside of the vessel. These programs were conducted in Russia and involved the development of technology to produce large scale melts of prototypic core debris involving UO_2, ZrO_2, and Zr. The major tests in the program have now been completed. Efforts are under way to identify and maintain the experimental capabilities that have been developed for the MASCA program since these capabilities may be essential for the investigation of severe accidents in reactors that do not use light water technology.

OECD-MCCI

This is an international collaborative experimental study being conducted at the Argonne National Laboratory to investigate the viability of using an overlying layer of water to cool core debris interacting with structural concrete. This program is nearing completion.

Planned modifications of the MELCOR code to address the ACR-700 have been curtailed since the application for certification of this reactor has not been submitted. There still may a need to upgrade the modeling of iodine chemistry in reactor containments to respond to recent findings concerning the effects of tri-sodium phosphate buffer in reactor sumps on sump pump screen blockage.

The ACRS is very supportive of the strategy NRC has developed to maintain and update its capabilities for severe accident analyses.

The leveraging of resources through international collaborative experimental research is especially important. The planned extensions and continuations of current collaborations are well worth the investment. This type of collaboration in experimental research could be emulated in other NRC research areas such as fire safety research and thermal-hydraulics research.

Table 11. Severe Accident Research Activities

Job Code	Title	Comment
Y6321	***Benchmark, MOX Fuel Release, Source Term Experiments***	International Collaborative follow-on to the PHEBUS-FP experiments.
Y6328	***Assessment and Analysis of PHEBUS-ST***	In-kind support for the follow on to the PHEBUS-FP experiments. This work is providing data on fission product behavior during reactor accidents for use in MELCOR development.
Y6628	***Consequence Models and Uncertainty Assessment***	Uncertainty analysis of the MACCS code for computing reactor accident consequences.
Y6313	***OECD-MCCI Program***	International collaborative research on the interactions of core debris with concrete. This program should be completed next year
Y6690	***Analysis Support for OECD-MCCI Program***	In-kind and financial support for the international collaborative research on ex-vessel core debris interactions with concrete.
Y6312	***MASCA Program***	International collaborative research on the behavior of molten core debris in the lower plenum of a reactor vessel. This program has resolved safety issues with respect to invessel retention of core debris. The program has developed the capability to produce and test large-scale melts of uranium dioxide that may be of use in advanced reactor safety model development and validation.
Y6802	***MELCOR Severe Accident Code Development and Assessment***	Computer model for the analysis of severe reactor accident and repository for severe accident research results. This is the agency tool for Level 2 PRA including source term characterization; MELCOR is the repository for severe accident research results obtained by the agency.
Y6721	***AGT W/IBRAE-RAS on Nuclear Safety Analysis Codes***	Support for Russian investigators in the development of a FORTRAN-95 version of MELCOR. This program is modernizing the coding in MELCOR by cost-effective use of expertise in Russia.

Table 11. Severe Accident Research Activities
(Continued)

Job Code	Title	Comment
Y6848	***High Burnup Fission Product Release Data***	Refine release models in MELCOR for the effects of high fuel burnup; code analyses will be used to create a licensing source term applicable to high-burnup fuel and reflecting improved modeling of severe accidents.
Y6517	***High Burnup Source Term for Storage***	Establish the technical basis for the extension of regulatory guide on spent fuel heat generation in a spent fuel storage facility to include high-burnup fuel
Y6504	***Steam Generator Fission Product Retention***	International collaborative research on the retention of aerosols on the secondary sides of steam generators in containment bypass accidents (ARTIST program). This program provides an experimental resolution of a long-standing issue of source terms from accidents that bypass containments.
Y6607	***Support ARTIST Tests***	In-kind support for the ARTIST program - see Y6504 above.
Y6486	***Severe Accident Initiated Steam Generator Tube Rupture Sequences***	Investigation of the potential for induced steam generator tube failure during severe accidents leading to containment bypass. This is an important part of the Steam Generator Action plan and the analysis of plant behavior under accident conditions.
Research Programs to Maintain the MACCS Code for Consequence Analysis		
Y6785	***Plume Model Adequacy Evaluation***	Test the assumption that simple plume treatments in MACCS code are adequate by comparing with the state-of-the-art dispersion model. This activity is important to show MACCS is adequate for regulatory needs.
Y6628	***MACCS Uncertainty Assessment for Consequence Models***	Support for emergency planning.
Y6469	***Evaluation of Radionuclide Pathways and Uptakes***	Upgrade information on uptake pathways. This project upgrades the code to take advantage of more recent information.

14 THERMAL-HYDRAULICS RESEARCH

Thermal hydraulics, especially the dynamics of two-phase flow, have always been essential elements of the regulatory evaluation of design basis accidents. NRC confirmatory evaluation of licensees' submittals in the area of thermal hydraulics has long been a major element of many licensing actions. Thermal-hydraulic analyses have grown ever more sophisticated. This trend is likely to continue for existing plants as licensees seek power uprates and take advantage of NRC's willingness to allow best-estimate analyses (with scrupulous attention to uncertainties) in the place of deliberately bounding, conservative analyses. To evaluate the adequacy of the licensees' analyses, NRC must have state-of-the-art thermal-hydraulic computational tools and equally sophisticated understanding of both thermal-hydraulic phenomena and the limitations of computer codes. NRC attempts to maintain its competence in the thermal-hydraulic field through its research program.

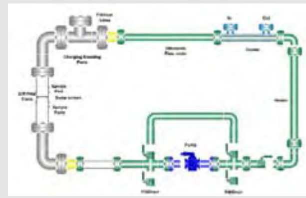
Major elements of the current NRC thermal-hydraulics research program can be grouped into three general areas:

- PWR sump screen blockage issues
- TRACE computer code development
- Experimental studies of thermal-hydraulic phenomena

These major features of the current thermal-hydraulics research program are discussed below.

PWR Sump Screen Blockage

The sump screen blockage issue for PWRs is the analog of a previous issue identified for BWRs. Debris from coatings and insulation can be generated during the high-pressure blowdown of the reactor coolant system




Chemical Effects/Head-Loss Tests in a Simulated PWR Sump Pool Environment

GSI-191 addresses the potential for debris accumulation on PWR sump screens to affect emergency core cooling system (ECCS) pump net positive suction head margin. In response to a concern expressed by the ACRS, RES has initiated a program to investigate the potential for chemical reactions that can occur in the containment pool to produce chemical products that can increase the head losses over those due to the physical debris alone.

NRC and the nuclear utility industry jointly developed an Integrated Chemical Effects Tests (ICET) program to determine if chemical reaction products can form in representative PWR post-LOCA containment sump environment. These tests were conducted by Los Alamos National Laboratory (LANL) at the University of New Mexico (UNM). Chemical products were observed in all five test series.

A head–loss loop was set up at Argonne National Laboratory (ANL) to investigate the potential head loss associated with the chemical products observed in the ICET tests.

These recent research results indicate that a simulated pool environment containing phosphate and dissolved calcium can rapidly produce a calcium phosphate precipitate that , if transported to a fiber bed covered screen, produces significant head loss.

following a major pipe break. This debris can clog the screens protecting the intake pumps for the emergency cooling system and prevent adequate coolant flow. Blockage issues have been exasperated by the discovery of mechanical and chemical effects that magnify the blocking effects of debris trapped on the sump screens. As a result, it is difficult to design screens that are of sufficient size to ensure emergency core cooling. The industry is looking to the NRC for guidance on acceptable methods for sizing screens to protect the sump intakes of the cooling pumps.

The NRC is still in the exploratory phase of research on sump screen blockage. It is still identifying phenomena that affect blockage. It is far from developing tools and methods that can be used with confidence for making predictions. NRC staff is now analyzing the licensees' responses to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors." These responses should reveal the licensees' views of current predictive techniques and their applicability, as well as indicate what methods they expect to use to assess the adequacy of their current and modified screen systems. The NRC staff needs to have sufficient technical knowledge to evaluate these methods. Current NRC research is focused on significant gaps in knowledge, establishing what phenomena play significant roles, and on developing general awareness of what analytical steps are needed to describe the phenomena adequately. The ACRS would expect that many of the details of predictive methods, such as the coefficients in correlations, computational schemes, and methods for developing suitable conservatism to account for uncertainty, could be left to the licensees or to industry-sponsored organizations such as EPRI. This is possible, however, only when the phenomena are well understood and a technical basis has been established for their prediction. When this is not the case, the NRC may need to develop sufficient predictive ability of its own to achieve authoritative competence to evaluate licensees' submittals.

For example, the NRC-sponsored research has revealed the "thin bed effect". This appears to involve a dense agglomeration of fine particles that fill the pores in a layer of debris, such as fiberglass, but the mechanism by which it occurs and how it influences the pressure drop are not understood. Previous NRC acceptance of pertinent Nuclear Energy Institute (NEI) guidance now appears premature in light of confirmatory research that has revealed much larger influence of the bed structure (e.g. up to a factor of about 100 on pressure drop for the same mix of fibers and particles) than had previously been thought to be possible. Research in this area should be continued and expanded as needed in order to reduce the very large uncertainties surrounding these effects and to determine if a predictive capability is feasible.

Other important phenomena, such as chemical and downstream (of the screen) effects are now being investigated by RES. These are essentially exploratory studies that have uncovered some significant effects, but have yet to reveal their scope and magnitude. Predictive capability remains to be demonstrated. The NRC needs to evaluate the results of these studies and determine how much it can rely on the nuclear industry to develop reliable predictive tools and how much independent predictive capability it requires. Development of a predictive capability may require investment of substantial resources and time.

TRACE Computer Code Development

Several years ago, the NRC recognized that it could not sustain the continued maintenance of several thermal-hydraulic codes for each general type of nuclear power plant. It elected to consolidate its existing codes for the confirmatory analysis of licensee submittals on design basis thermal-hydraulic issues into a single code now called TRACE. The consolidation is now largely completed. The TRACE computer code is viewed by the NRC research staff as “as good as anything else that is out there.” The long-term validation and improvement phase of code development is at hand. Current research is devoted to improving features of the TRACE code, making it easier to use and validating it against available data. Some of the data already exist and other data are being generated. In addition, the integration of the TRACE code, coupled with the CONTAIN code to model containment response and the PARKS code for neutronic analyses into the regulatory processes of the agency has begun.

The TRACE code is reputed to now be able to serve as the “workhorse” thermal-hydraulic analysis code for the agency. In the course of its work to consolidate thermal-hydraulics codes into TRACE, the research staff has found many ways to improve the code. Such improvements should be done. Now, however, it is far more important that the integration of TRACE into the regulatory process be completed in an expeditious manner. The research staff working on the development needs to have input from users of the code on needed features and capability of the code. Inevitably, the introduction of a new computational tool will slow and detract the regulatory process for some transient period. There is no way to counter this difficulty associated with the introduction of a new computer code. It must be endured and the sooner this is done, the sooner the challenges associated with the use of a new code in the regulatory process can be overcome. Once TRACE is integrated into the regulatory process, the developers will receive valuable advice on how their efforts to improve the code should be directed to enhance the regulatory process.

Highest priority should be given to the integration of TRACE code into the regulatory process. As this integration progresses, the research staff can continue its efforts to improve and further develop TRACE on a “time available” basis. The ACRS is concerned that efforts to improve TRACE lack prioritization and defensible organization. Placing the TRACE code in the hands of users will also identify a host of needed improvements. Prioritization of technical improvements might be aided substantially by commissioning a detailed peer review of TRACE. To do this, the staff will have to have available code documentation of outstanding scope and quality. Such high quality code documentation will also be needed if the code is to become part of the regulatory process. Code documentation, then, is a task that ought to take precedence in the thermal hydraulic research effort.

Experimental Studies of Thermal-Hydraulic Phenomena

Thermal-hydraulic phenomena involving the flow of two-phase mixtures of steam and water are very complicated especially those involving blowdown from high pressure systems. Thermal-hydraulic phenomena that arise in advanced light water reactor designs that emphasize passive response to accidents are driven by subtle forces that require sophisticated understanding to ensure plant safety. As a consequence, NRC has long felt that it cannot rely solely on computer code projections of thermal-hydraulic phenomena to ensure adequate protection of the public health and safety. Experimental confirmation is also required. As the computer models used to analyze thermal-hydraulic phenomena have become more sophisticated, the experiments needed to validate model predictions have become progressively more integral in nature.

Experimental facilities have become larger and more complex. RES has an interest in maintaining these facilities for use in addressing future as well as current regulatory issues. Maintenance of large, complex experimental facilities has become a significant expense in this research area. The major experimental facilities used by NRC in the U.S. are the APEX and PUMA facilities as well as RBHT facility at Penn State University. Abroad, NRC is conducting tests at the PKL facility, the SETH tests and tests at the ROSA facility. Additional experimental needs may arise in connection with the design certification of the ESBWR.

APEX is a medium-size, scaled, integral test facility that proved useful for the certification of the AP600 and AP1000 reactor designs. It has been modified to provide data crucial to the analysis of thermal shock to reactor vessels. It is proposed now that the APEX facility be used for confirmatory analyses for AP1000 and for some “thermal hydraulic integral experiments.” These proposed applications would benefit from review to assess their focus and applicability.

PUMA is a medium size, scaled facility especially suited for evaluating passive emergency core cooling systems. It is being modified to be applicable to testing the emergency core cooling systems for the ESBWR.

The RBHT test program has been under way for a number of years with the purpose of improving core reflood models that are a key part of evaluating the adequacy of pressurized water reactor emergency core cooling systems. The reflood models may become critical if applications are submitted for large power uprates in PWRs. The proposed research program at the RBHT facility needs evaluation to see if the quality, scope and detail of the data are properly matched to the proposed uses of these data.

NRC has wisely not sought to duplicate large test facilities available overseas. Use of these facilities is possible through international programs. The SETH program was useful for resolving Generic Safety Issue (GSI) 185 and assessing the emergency heat removal systems in the ESBWR. Future work under this program at the ROSA and the PKL facilities in support of the TRACE code needs to be more clearly focused.

It is essential for NRC to maintain an ability to assess thermal-hydraulic phenomena that occur both in existing reactors and in future reactors. It is evident that the development of computer codes to predict thermal hydraulic phenomena and the experimental validation of these predictions will grow more burdensome with time. Major development efforts can be anticipated if very innovative designs using coolants other than water are brought forward for certification. It is not likely that the nuclear institutions of any one country will be able to develop adequate codes and conduct sufficient validation of these codes alone. International cooperative development of codes and conduct of experiments appear essential as NRC research moves beyond TRACE with its current capabilities and especially if analyses are needed for coolants other than water. NRC already takes substantial advantage of international experimental capabilities. Extending this international flavor in thermal-hydraulics research to include the development of computer codes will contribute to current ideas of multi-national design approval process. It may slow code development. It also may ensure that sufficient resources for code development are available so that it is feasible to meet the more exacting standards that are likely to be demanded in the future.

Table 12. Thermal-Hydraulics Research Activities

Job Code	Title	Comment
N6106	***Confirmatory Head Loss Testing***	Experiments to measure head loss across sump pump strainers in PWRs.
Y6871	***PWR Sump Screen Penetration and Throttle Valve Testing***	Experiments to determine the type and quantity of debris that can pass through typical PWR sump screens.
N6100	***Head Loss Testing***	Assess the susceptibility of recirculation screens to debris blockage during design basis accidents.
Y6999	***Integrated Chemical Effects Tests***	Five tests to determine representative chemical and material environments in PWRs that can contribute to sump blockage.
N6121	***GSI-191 Chemical Effects Simulations***	Experiments to determine chemical effects that can contribute to sump screen blockage.
N6198	***Transportability of Coatings***	Parametric study to ascertain if coatings can be transported to sumps under accident conditions.
N6083	***BWR ECCS Suction Concerns***	Technical assessment of Generic Issue 193 “BWR Suction Concerns.”
Y6769	***PUMA Test Facility***	Facility for the conduct of thermal hydraulics tests. This facility can produce data for natural circulation systems for use in ESBWR design certification.
Y6852	***PWR Thermal-Hydraulics Integral Experiments***	Tests at the APEX facility at Oregon State University.
N6042	***OECD/ROSA Program***	International collaborative tests of reactor accident thermal hydraulic phenomena.
Y6945	***Rod Bundle Heat Transfer Test Program - Phase 3***	Experiments at Penn State University in support of TRACE code analyses of small and large break loss of coolant accidents. To date, there is little evidence that data from this facility can be of value for TRACE code development. Further work in this facility should be scrutinized carefully to assure that it meets agency needs.

Table 12. Thermal-Hydraulics Research Activities
(Continued)

Job Code	Title	Comment
Y6589	***Thermal-Hydraulic Research***	Perform analytical and small-scale experimental work in support of the TRACE code. Neutronic work in this program in nearly complete. Long-range thermal hydraulic work needs to be shown necessary for agency needs.
N6043	***Thermal-Hydraulic Sub-channel International Standard***	Analysis for international standard problem for a BWR subchannel benchmark.
Y6571	***SETH Program - Test Facilities***	Thermal-hydraulics tests in two international efforts: PKL on boron dilution and PANDA in support of ESBWR certification.
Y6974	***OECD-PKL Program and Test Facility***	International collaborative research on boron dilution accidents including mid-loop operation.
N6213	***TRACE Verification and Validation***	Verification and validation of the TRACE thermal-hydraulics analysis code. This work is viewed as vital to the verification and validation of TRACE.
Y6673	***TRAC-M Development and Assessment - Small LOCA Processes (In the past, the TRACE code was called TRAC-M)***	Simulate separate effects tests with the TRACE code and show acceptable agreement with predecessor codes. Good progress has been made in this important work.
Y6666	***Advanced Numerical Methods in TRAC-M (In the past, the TRACE code was called TRAC-M)***	Advanced numerical methods for the TRACE code. This work is not essential for the current range of efforts to make TRACE useful to the agency.
N6147	***TRACE Development and Assessment Against Specified Tests***	Use TRACE code to evaluate level swell tests done at several facilities. This is a small part of the TRACE validation and verification effort.
N6201	***Gravity Reflood and SBLOCA TRACE Assessment***	Use the TRACE code to assess PUMA facility tests. This work necessary to lend credibility to TRACE for ESBWR analysis.

Table 12. Thermal-Hydraulics Research Activities
(Continued)

Job Code	Title	Comment
Y6525	***TRAC-M Code Maintenance (In the past, the TRACE code was called TRAC-M)***	Maintenance of the TRACE code. This is an essential activity.
N6040	***Data Acquisition***	Recover old input decks for the TRAC-PWR model.
N6072	***Implementation of ACR-700 (Misleading title, Project deals with PUMA input deck)***	This work is no longer necessary.
Y6198	***Continuation of Support for System Code Analysis***	Support for the SCDAP/RELAP5 computer code and the analysis of steam generator tube rupture accidents.
Y6392	***Maintenance, Application, Assessment and Development of NRC Computer Codes***	Consolidation of RELAP5 capabilities into TRACE. This work appears to overlap most of the TRACE development tasks. Incorporation of RELAP capabilities into TRACE has proven difficult because of code philosophy differences.
Y6667	***SNAP Implementation***	Graphical user interface for TRACE and other NRC computer codes. This work is important because of poor direct input methods inherited in TRACE from the underlying TRAC models.
Y6662	***AP1000 Confirmatory Thermalhydraulics Analysis***	Confirmatory thermal hydraulic analyses of a wide range of design basis accidents hypothesized to occur in AP1000. This work is complete.
Y6526	***Administer CAMP Meeting***	Meeting of users of NRC thermal-hydraulics codes. This program will assist in the international acceptance of TRACE.
N6030	***Flow-induced Vibrations and Effects on BWR components***	Analysis of component vibration that can lead to fatigue failure in BWRs.

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