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Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program

A Report to the U. S. Nuclear Regulatory Commission

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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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). This report focuses on that portion of the NRC research program dealing with the safety of existing nuclear reactors and advanced light water reactor designs, AP1000 and ESBWR, submitted for certification. In its review of the NRC research activities, the ACRS considered the programmatic justification for the research as well as the technical approach and progress of the work. This review attempts to identify research crucial to the NRC mission. It also attempts to identify 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. The review also attempts to identify areas where greater international cooperation in research useful to the NRC could leverage resources of partners in the research and yield superior technical products. The report does not address research on the vulnerability of existing nuclear power plants to acts of sabotage and terrorism.

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ACNW	Advisory Committee on Nuclear Waste
ACR-700	Advanced CANDU Reactor-700
AEOD	Office for Analysis and Evaluation of Operational Data
ANS	American Nuclear Society
ATHEANA	A Technique for Human Event Analysis
ATWS	Anticipated Transients Without Scram
BEIR	Biological Effects of Ionizing Radiation
BMI	Bare Metal Inspection
BWR	Boiling Water Reactor
CAMP	Code Applications and Maintenance Program
CFD	Computational Fluid Dynamics
CFR	Code of Federal Regulations
CSARP	Cooperative Severe Accident Research Program
CSNI	Committee on the Safety of Nuclear Installations
DOE	Department of Energy
EPA	Environmental Protection Agency
EPIX	Equipment Performance and Information Exchange System
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
FY	Fiscal Year
GALL	Generic Aging Lessons Learned (Report)
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
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronics Engineers
INPO	Institute of Nuclear Power Operations
IPEEE	Individual Plant Examination of External Events
IRIS	International Reactor Innovative and Secure
ISI	In-Service Inspection
ISO	International Standard Organizations
LER	Licensee Event Report
LERF	Large Early Release Frequency
LBLOCA	Large-Break Loss-Of-Coolant Accident
LOCA	Loss-of-Coolant Accident
LWR	Light Water Reactor
MACCS	MELCOR Accident Consequence Code System
MOV	Motor-Operated Valve
MOX	Mixed Oxide
NCRP	National Council on Radiation Protection

ABBREVIATIONS (Cont'd)

NDE	Non-Destructive Examination
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NMSS	Office of Nuclear Material Safety and Safeguards
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OECD	Organization for Economic Cooperation and Development
PARCS	Purdue Advanced Reactor Core Simulator
PFM	Probabilistic Fracture Mechanics
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PSHA	Probabilistic Seismic Hazard Analysis
PTS	Pressurized Thermal Shock
PUMA	Purdue University Multidimensional Integral Test Assembly
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
ROP	Reactor Oversight Process
RPV	Reactor Pressure Vessel
SCI	Secondary Containment Isolation
SDP	Significance Determination Process
SNAP	Symbolic Nuclear Analysis Package
SPAR	Simplified Plant Analysis Risk Model
SSHAC	Senior Seismic Hazard Analysis Committee
TRACE	TRAC-RELAP Advanced Computational Engine
UCLA	University of California, Los Angles
U.S.	United States
USGS	United States Geological Survey
VHP	Vessel Head Penetration

1 INTRODUCTION

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

- Ensure its regulatory framework has a sound technical basis
- Prepare for the anticipated changes in the nuclear industry that could have safety implications
- Develop improved methods for its regulatory process
- Maintain an infrastructure of expertise, facilities, analytical tools, and data to support regulatory decisionmaking

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

- Nuclear power plants age and encounter challenges of material degradation not anticipated when the plants were designed
- The NRC considers applications for extending licenses and increasing the operating power levels of plants
- Reactor fuels are used to higher levels of burnup, new claddings for the fuels are introduced, and mixed oxide (MOX) fuels are 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 greater use of 'best-estimate' analyses
- New technologies such as software-based digital instrumentation and control (I&C) systems are backfit into the existing plants
- New light water reactor (LWR) designs making more use of passive systems are submitted for certification

In this report, the Advisory Committee on Reactor Safeguards (ACRS) presents its observations and recommendations concerning that portion of the NRC Safety Research Program focused primarily on the safety of existing nuclear reactors. This report does consider research activities that support the certification of the advanced LWRs, AP1000 and ESBWR. It does not address research to prepare the agency to certify the ACR-700 design or other advanced reactor designs such as the GEN IV design being considered by the Department of Energy (DOE) or the IRIS design. The ACRS recently reported on the agency's plans for research on these advanced reactor designs [Ref. 1] and the progress in advanced reactor research has not been sufficient to warrant reexamination of these research activities at this time. The present report does not address research on nuclear waste and the licensing of a permanent geological repository for spent reactor fuel. The Advisory Committee on Nuclear Waste (ACNW) monitors this research and will report its conclusions and recommendations separately. This report does not address research on the vulnerability of existing nuclear power plants to acts of sabotage and terrorism. The ACRS will report separately on the technical aspects of this research.

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 issues
- Provides the technical basis for resolution of foreseeable safety issues
- Develops the capability of the agency to independently review risk-significant proposals and submittals by the licensees
- Supports initiatives of the agency, including the Reactor Oversight Process (ROP) and the move to risk-informed and performance-based regulation
- Improves the efficiency and effectiveness of the regulatory process
- Maintains technical expertise within the agency and facilities in disciplines crucial to the agency mission and not readily available from other sources

This review of the NRC Safety Research Program, in particular, attempts to identify 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 attempts to identify areas where greater international cooperation in research useful to the NRC could leverage resources of partners in the research and yield superior technical products.

General observations and recommendations concerning the NRC research activities are presented in Chapter 2 of the report.

Other observations and recommendations concerning the research activities in the following technical disciplines are discussed in Chapters 3 through 15 of this report:

- Analysis and Evaluation of Operational Data
- Containment Systems
- Digital Instrumentation and Control Systems
- Fire Safety Research
- Reactor Fuel Research
- Neutronics and Criticality Safety
- Human Factors and Human Reliability Research
- Materials and Metallurgy
- Probabilistic Risk Assessment
- Radiation Protection
- Seismic Research
- Severe Accident Research
- Thermal-Hydraulics Research

2 GENERAL OBSERVATIONS AND RECOMMENDATIONS

Overall, the NRC has a well-focused, well-planned Safety Research Program dealing with existing reactors and advanced LWRs submitted for certification. The research effort may well be near the minimum needed to support regulatory activities and agency initiatives while maintaining technical competencies crucial to the agency mission. A very large fraction of the research is focused on immediate user needs. Resources available for exploratory research to investigate other avenues of improved safety and more efficient regulation are minimal and may limit the agency's ability to anticipate future needs.

Some of the research activities are especially noteworthy. This includes the research being done in probabilistic risk assessment (PRA) to provide computational tools to support the Reactor Oversight Process. Confirmatory research resolving safety issues of reactivity initiated accidents with high-burnup fuel is also noteworthy. Human factors research at NRC has been re-energized and holds the promise of significant future contributions to the regulatory process.

There are research challenges. The important effort to provide the agency with a consolidated, state-of-the-art, best-estimate thermal-hydraulics computer code is such a challenge. Though this effort has progressed well, the time is approaching when the need for this code will be critical. It may be necessary to increase the effort and focus the thermal-hydraulics research on producing this code and integrating it into the regulatory process. Once this is accomplished, research resources will be freed to improve the predictive capabilities in thermal-hydraulics. Further research in the area of thermal-hydraulics especially in areas other than large-break loss-of-coolant accidents (LB-LOCAs) might be facilitated by more extensive international cooperation akin to that being done in the area of severe accident research.

Some research efforts have produced valuable technologies, which can be integrated into the regulatory process. Further research in these areas should be reduced substantially. An example of such an effort is the research on the realistic structural capacity of reactor containments. This research has produced, benchmarked, and validated computer codes that can be used to assess containment designs and estimate the effects of both materials degradation and construction errors.

Research on seismic engineering has reached the point that the most significant agency needs have been met and greater reliance on engineering consulting firms, as needed, may be adequate for the regulatory process in the future. There is advantage in maintaining modest research efforts to accrue benefits from cooperative research on seismology and seismic engineering.

Other observations and recommendations concerning the research efforts are summarized below and also discussed in individual Chapters.

Analysis and Evaluation of Operational Data

The ACRS supports the research activities now under way in the general area of analysis and evaluation of operational data. Indeed, these data collection and organizational activities, including superior computerized search capabilities, are essential to the agency mission. However, the ACRS is concerned about the vitality and planning of continued efforts to use the database and especially the opportunities being made to explore the database independently of current research needs. Such independent examinations of the database in the search for unexpected interactions among regulatory activities can always be deferred at little cost. Continued deferral will deprive the agency of the opportunity to utilize operational data in many ways that may lead to more effective and realistic regulatory practices.

Containment Systems

Adequate computer models to assess the structural capabilities of containments of existing reactors are now available. They have been benchmarked and validated by comparison to well-scaled test data. The effects of degradation as well as the effects of construction errors can be evaluated adequately by analysis without the need for further experiments.

Debris accumulation in pressurized water reactor (PWR) sumps is an important issue that must be resolved. The planned work to further explore the potential for chemical interactions that generate suspensions, which can lead to large head losses even with low fiber loadings, is extremely important. Research in this area needs to be expedited and promptly reach the point licensees can implement solutions to this issue.

Digital Instrumentation and Control Systems

The current reliance on controlling the design and development process of software, rather than focusing on the software itself, has the potential of straining agency resources in the regulatory process. Although a controlled development process is assumed to lead to a highly reliable product, this reliability remains unquantified. Complicating the regulatory review of these I&C systems is the pace of innovation in the digital electronics industry. Innovations of revolutionary natures are being made at rates far greater than those the regulatory system can respond to. Furthermore, threats to digital systems are greatly expanded with the emergence of the so-called "cyber security" threat posed by the malevolent or just the prankster.

In general, the ACRS supports the research activities now under way in the Digital I&C Systems. These activities, if successful, will provide tools to make the review process more efficient and provide a basis for including software reliability into PRAs, thus reducing the need for relying on controlling the design and development process, and enhancing the ability of the agency to risk-inform its regulations.

Fire Safety Research

The current NRC research efforts in Fire Safety seem incongruent with the estimated risk significance of fire. The limited fire research effort is understandable since resources for such research have been diverted to respond to the events of September 11, 2001. Now that these responses are being completed, the agency should revitalize its Fire Safety research efforts and move the technical capabilities of the agency to be more in line with risks being ascribed to fire.

Reactor Fuel Research

The NRC is completing a confirmatory research effort on reactivity insertion with high-burnup fuel. A sustained NRC research effort in reactor fuels behavior under accident and off-normal conditions is needed especially as competitive pressures force fuel vendors to curtail their research efforts. Continuing expertise and even additional research may be needed if, as now expected, licensees make requests to extend fuel burnups up to and beyond 75 GWd/t. It is quite likely that such industry proposals will be substantiated by minimal experimental research. It is important for the NRC to have a technically justified position on the experimental research that is necessary to support such proposals especially in light of the "surprise" that accompanied extending fuel burnups into the range of 50 to 60 GWd/t.

Neutronics and Criticality Safety

The Office of Nuclear Regulatory Research (RES) is doing a commendable, cost-effective job in maintaining the capabilities of its neutronics and core physics codes and the associated databases. Upgrades to these codes to address both extended fuel burnup and MOX fuels are needed and these needs are addressed in the current research projects. The ACRS anticipates that these extended capabilities will be tested significantly in the certification of the advanced reactor designs.

Human Factors and Human Reliability Research

The attention NRC is paying to Human Factors research is yielding dividends. The Standard Review Plan Chapter 18 has recently been updated to reflect developments especially in the area of digital I&C systems. A risk-informed method to screen licensee submittals for human factors review has the potential of better focusing agency resources on risk- significant human factors issues.

The issues surrounding human performance in teams (such as the team of control room operators) and human performance in organizations, including the question of organizational reliability, are important and need to be considered in the ongoing Human Performance Research planning.

The NRC is experimenting with an ambitious methodology for estimating human reliability that is to assess errors of commission as well as errors of omission. This methodology called ATHEANA focuses on the causes of human performance ("performance shaping factors"). The ACRS looks forward to reviewing the results of the ongoing ATHEANA application efforts and learning whether the method has become easier to apply, whether it yields insights significantly better than those that could be derived from simpler methods, and whether it includes a quantification process for human reliability. Of particular interest to the ACRS is the use of expert opinion elicitation in these applications to quantify the ATHEANA results.

Materials and Metallurgy

Research in this area addresses primarily radiation-induced embrittlement of ferritic pressure vessel steels and austenitic core structure materials, and the characterization and monitoring of environmentally-assisted cracking. The quantification of these phenomena is central to maintaining the integrity in LWR structures.

Projects dealing with Reactor Pressure Vessel (RPV) Embrittlement should be considered with respect to safety issues of RPV integrity and with respect to maintaining the core competency in irradiation embrittlement. The staff needs to evaluate what research should be retained in order to meet these two needs.

The characterization of cracking kinetics should be continued in the areas of irradiation-assisted cracking of irradiated stainless steels used in boiling water reactor (BWR) core structures and of stress-corrosion cracking of nickel base alloys used in PWR steam generators and in large components in all LWR designs. This information, in addition to the evaluation of various inspection techniques, is required for the NRC to maintain an independent capability to examine integrity issues inherent to licensee plans concerning residual life, inspection, and repair. Many of these projects are being conducted within international research programs, thereby leveraging NRC resources.

The ACRS supports the "Proactive Materials Degradation Assessment" Project recently initiated by the NRC in order to move from the economically inefficient reactive mode of regulating materials degradation issues. The plan to lead a phenomena identification and ranking table (PIRT) effort with the industry over the next two years, with expert elicitation from international experts, should identify potential materials degradations that may occur in the future. This knowledge will provide a better basis for future regulatory decisions on timely inspection and repair criteria.

Probabilistic Risk Assessment

The NRC has an impressive PRA research program. The ACRS generally supports the idea that the NRC needs to maintain its PRA capabilities at or near the state-of-the-art level. As the NRC moves toward a risk-informed regulatory system, it will need ever greater amounts of risk information derived from quantitative risk assessments. It will want to maintain its own independent capabilities for conducting PRAs at state-of-the-art levels. A continued vital research program in PRA is, then, crucial to the agency mission.

Radiation Protection

The NRC research activities in the area of Radiation Protection constitute a well-leveraged program that allows the NRC to use quality data and information as the foundation of its basic radiation protection regulation, 10 CFR Part 20, "Standards for Protection Against Radiation."

Seismic Research

The NRC has invested heavily in the understanding of seismic issues and development of its rules to ensure that seismic risks are acceptably low. Licensee submittals concerning risks associated with external accident initiators such as seismic events suggest these efforts have been largely successful.

Much of the NRC research into seismic issues involves cooperative efforts to address essential seismic data. The ACRS supports these efforts which are highly leveraged by contributions from other partners.

The ACRS understands that there are needs for additional work on the engineering of components and structures to avoid seismic damage. Unlike many other issues that are dealt with by NRC research, there is in this Country a very substantial engineering and scientific community addressing this issue. This community is independent of the NRC licensees. Thus, the NRC-sponsored work is not needed to maintain a core competency in this area. If there are open issues associated with licensee submittals such as the seismic behavior of spent fuel casks or buried structures, the NRC can rely on engineering consulting firms on an as-needed basis to deal with such issues.

Severe Accident Research

The ACRS is very supportive of the NRC strategy to maintain and update its capabilities for accident analyses. The ACRS encourages the NRC to give serious consideration to joining international cooperative research agreements now being proposed to conduct:

- Prototypic, in-pile tests of radionuclide releases from, and degradation of, reactor fuel exposed to air for both reactor accident analyses and spent fuel storage pool accidents

- Prototypic, in-pile tests of radionuclide releases from, and degradation of, high-burnup fuels

- Prototypic, in-pile tests of radionuclide releases from, and degradation of, MOX fuel

The ACRS also supports programs to maintain the MACCS code and looks forward to reviewing results of the comparison of the Gaussian plume model to state-of-the-art dispersion models in the context of risk assessments.

Thermal-Hydraulics Research

The decision to consolidate NRC's thermal-hydraulic modeling into a single code was enthusiastically supported by the ACRS. The ACRS understood this to be a challenging undertaking. Indeed, the consolidation has proved to be every bit as challenging as anticipated. A version of the consolidated code, TRACE, is now being tested. Efforts should be focused on integrating TRACE into the regulatory process even at the expense of research to further improve the technical capability of TRACE.

The ACRS encourages anticipatory research to couple computational fluid dynamics (CFD) methods with system codes such as TRACE. This has the potential to enhance the accuracy of thermal-hydraulic analyses.

3 ANALYSIS AND EVALUATION OF OPERATIONAL DATA

The NRC is moving toward a risk-informed regulatory system that will require more realistic estimates of vulnerabilities of nuclear power plants. The ACRS has voiced its belief that the operational database for nuclear power plants is a rich source of information for improving the quality of quantitative risk assessments and identifying omissions in these assessments [Refs. 2 and 3]. For several years following the 1979 accident at Three Mile Island, then the Office for Analysis and Evaluation of Operational Data (AEOD) conducted independently motivated, detailed evaluations of the operational database [Refs. 4 and 5]. The products of these evaluations were of direct use to the quantitative risk analysis community.

In 1999, the function of the AEOD was transferred to RES. It is of considerable interest to the ACRS to see that the intensely inquisitive, independent evaluation of the operational database is sustained in this organizational structure. Furthermore, the ACRS is anxious to see that results of such independent investigations are used by other NRC Offices.

Research Projects in the area of Analysis and Evaluation of Operational Data[1] are listed in Table 1. These Projects are primarily devoted to the sustained acquisition and organization of operational data from licensee event reports (LERs), maintenance data and the like. Currently, the motivation for the data collection efforts appears to be primarily anticipation of operational data needs of other research initiatives by the agency to address

A Success Story
Electrical Grid Stability

The Country was shocked in 2003 by a propagating loss of the electrical grid in Central and Eastern United States that left millions without electrical power. Grid vulnerability became a topic of numerous discussions in the news media. The vulnerability had been anticipated by the NRC study, "Operating Experience Assessment - Effects of Grid Events on Nuclear Power Plant Performance" [Ref. 6]. The study examined grid events that affected nuclear power plants from 1994 to 2001. The events were chosen for examination with attention to the structural changes that have occurred in the electrical energy market as economic deregulation has progressed. The study showed that the number of grid events may be decreasing with time, but the durations of events were increasing - a finding quoted by major networks. Though most events pose no exceptional challenges to nuclear power plant safety, issues are identified in the study that do have the potential to increase risk and challenge the effectiveness of the current regulations. The report documenting the study provides a baseline of grid performance which can be used to gauge the impact of deregulation on, and changes in, grid operations.

The report shows the public that the NRC is attentive to emerging issues that can affect power plant safety and how operational data can be used to anticipate and quantify emerging issues.

[1] Note that occupational exposure data and reports are discussed in Chapter 12, "Radiation Protection," of this report.

regulatory issues. Campaigns for the immediate future and the longer term to utilize the archives of operational data are no longer readily apparent from the research program. An undertaking that is apparent and could be of particular regulatory significance is the search for safety-significant interactions among the ongoing regulatory actions to:

- Extend the license period of nuclear power plants
- Increase the operating power levels of the plants
- Use fuel in the plants to higher levels of burnup
- Evolve plant operations to do more online maintenance

The ACRS is concerned that this search for interactions among the various operational changes has stagnated for lack of management support.

In summary, the ACRS does not critique the research Projects now under way in the general area of Analysis and Evaluation of Operational Data. Indeed, these data collection and organizational activities including superior computerized search capabilities are essential to the agency mission. It is evident that useful products are still being generated in this effort (See sidebar success story on Electrical Grid Stability). The ACRS is concerned about the vitality and planning of continued efforts to use the database and especially the opportunities being made to explore the database independently of current research needs. Such independent examinations of the database in the search for unexpected interactions among regulatory activities can always be deferred at little cost. Continued deferral will deprive the agency of the opportunity to utilize operational data in many ways that may lead to more effective and realistic regulatory practices.

Table 1. **Research Activities in Analysis and Evaluation of Operational Data**[2]

Project	Title	Comment
A9134	***Sequence Coding Search System*** Database from LERs used for NRC studies	This is an important Project for analyzing industry trends and establishing a data source for measuring industry performance.
G6810	***Management of Precursors Study***	This is a short-term grant for a two-day workshop for detecting, analyzing, and benefitting from knowledge of accident precursors.
J8258	***International Common-Cause Exchange Project*** Maintaining awareness of developments in common-cause failures within the international reactor safety community	This is an useful collaborative effort to maintain cognizance of developments in the realistic quantification of common-cause failures in PRA.
Y6215	***Operational Events for Accident Sequence Precursor Program***	This important Project seeks to identify and rank those operating events that were most significant in terms of the potential for inadequate core cooling and the potential for core damage.
Y6546 Y6636	***Industry Trends Program*** ***Support for Industry Trends***	These important Projects produce plant-specific and industry-wide estimates, summary tables, graphs, and charts from operational data to support the Industry Trends Program and various data analysis activities of the NRC.
Y6626	***Access to INPO's EPIX System***	Database on failures of key components; database for SPAR models used in the ROP.
Y6214	***Integrated Data Coding and Analysis Methods***	Improved database management; only a small effort remaining.
Y6468	***Reactor Operating Experience Data for Risk Application***	Collects, codes, and maintains operational data for reactor systems and components, initiating events, common-cause failures, and fire events for use in regulatory activities. Utilizes methods and procedures developed under Y6214.
Y6406	***Assess and Improve Regulatory Effectiveness*** (See also Y6522 in Probabilistic Risk Assessment, Table 9)	Potential for identification of unanticipated, safety-significant interactions among regulatory activities.

[2]See also Y6123, Retrospective Risk Analysis of Human Reliability in Operating Events, Table 7.

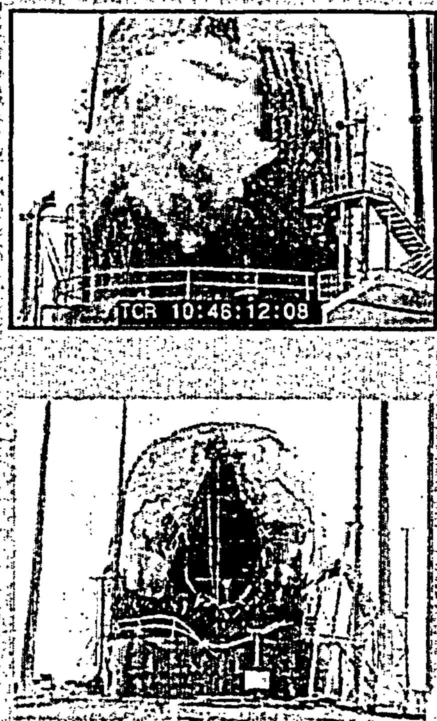
4 CONTAINMENT SYSTEMS

The defense-in-depth safety philosophy has led to encasement of power reactors in robust containment buildings. Containment is the final physical barrier to the uncontrolled release of radioactive materials to the environment in the event of an accident. Comparisons of the environmental releases of radioactivity during the accidents at Windscale, Three Mile Island, and Chernobyl demonstrate the value of robust containments.

The structural capabilities of containments for nuclear power plants in this Country are far greater than might be inferred from their design bases. To realistically assess these capabilities and demonstrate their capacity to provide protection in the event of severe accidents, the NRC, along with other countries and agencies, has established a technically sophisticated program. This program involved testing of quarter-scale containment models to provide databases for validating analytical structural models. "Round robin" comparisons of the predictions of these models to the data obtained in the tests were conducted.

This effort to develop validated technology for the quantitative characterization of containment capabilities has been both essential and quite successful. The technology for predicting containment responses to loads can now be transferred to the capable hands of users of the information such as line regulatory organizations, codes and standards organizations, and risk analysts.

Research studies to date have been on "as-designed" structures. Many containments have suffered some degradation to either the concrete shell or the steel liner and it is important to understand how degradation



Rupture of containment model during hydrostatic test

This test done in cooperation with Japan is one of a series of tests that have been done to benchmark and validate structural capacity computer codes. Typically, the scaled containment models are found to fail at discontinuities or flaws at pressures more than twice the design pressure. Computer codes validated by the testing are used to make realistic predictions of the failure probabilities of reactor containments under accident conditions.

affects the structural margins attributed to containments in risk analyses. Because the computer codes for structural analyses of containments have been benchmarked and validated, the effects of degradation, as well as the effects of construction errors, can be evaluated adequately by analysis without the need for further experiments. Projects Y6164 and W6684 (See Table 2) address such questions and these efforts should be continued. However, the value of general study performed under Project W6684 is questionable.

The aging management of containment structures is being addressed in the current activities associated with license extension. If further work is needed in this area, it should be done by the licensees in support of their applications.

Project W6593, "Effect of Aging and Emerging Issues on MOV performance," should be concluded. Active components such as motor operated valves (MOVs) are included in ongoing maintenance and testing programs which should be adequate to detect any potential aging issues. Further studies could improve reliability, but this should be the responsibility of the licensees or the vendors.

Debris accumulation in PWR sumps is an important issue that must be resolved. The ACRS has pointed out that the technology base is still not complete [Ref. 7]. The planned work to further explore the potential for chemical interactions that generate suspensions, which can lead to large head losses even with low fiber loadings, is extremely important. The Project Y6041 may deserve more management attention. Research in this area needs to be expedited and focused to support the regulatory process and provide a basis for licensees to implement solutions to this issue.

Assessment of containment structures for entombment, Project Y6331, can be subsumed within decommissioning projects. This does not appear to be a particularly pressing issue.

Terminating or transferring some of the Containment Systems Projects (See Table 2) should free resources available to the agency to address containment/confinement issues and security vulnerabilities of reactor containments for advanced reactors. The ACRS believes it would be prudent for the NRC to examine the safety issues associated with underground siting of future reactors and other new nuclear facilities.

Table 2 Research Activities in Containment Systems

Project	Title	Comment
J6043	*Inspection of Aged/Degraded Containment*	This work should be concluded. If additional information is needed, it should be provided by licensees who have the responsibility to demonstrate adequacy.
W6851	*Review Guidance for Lightning*	This effort should be concluded.
Y6757	*Containment Capacity Studies*	Development of technology to estimate capacities of containments to sustain pressure loads has been successful and should now be brought to closure.
W6593	*Effects of Aging and Emerging Issues on MOV Performance*	This work should be concluded. These are active components and ongoing programs should be adequate to detect any potential aging issues.
W6684	*Assessment of Aged and Degraded Structures and Components*	Margins associated with real, degraded structures must be understood for realistic analyses. Such issues are best addressed in terms of more specific conditions such as in Y6164. The value of a general study is questionable.
Y6041	*Assess Debris Accumulation on PWR Sump Performance*	This is an important issue but one that has been slow to resolve; research to support regulatory initiatives should be expedited and focused to support the regulatory process and provide a basis for licensees to implement solutions to this issue.
Y6164	*Structural Risk-informed Assessment Containment Degradation*	This work should be continued. It provides quantification of loss of margin as a result of degradation.
Y6331	*Long Term Containment Stabilization Concrete Structures*	Studies for entombment of nuclear facilities as permitted by regulations. This does not appear to be a particularly pressing issue.

5 DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

The move to software-based, digital I&C systems is taking place for advanced nuclear power plants and more slowly for existing nuclear power plants. Analog, electro-mechanical systems in existing plants are difficult to maintain as replacement parts of suitable quality become less readily available. Furthermore, there is a widespread belief that software-based digital electronic systems offer superior levels of control, information density, and reliability.

Certainly, digital systems have inherent reliability that is potentially much greater than electro-mechanical systems. There have been, however, notable instances of catastrophic failures of digital systems. Usually, these failures have been traced to omissions in the definitions of the requirements and specifications for the software that controls the system. The challenge in the regulation of digital I&C systems is the difficulty in comprehensive testing of these systems due to their diverse capabilities. At this time, in order to ensure adequate quality and reliability of software-based digital systems, the NRC relies primarily on the manpower-intensive monitoring of the design and development processes rather than focusing on the product. It is assumed that a controlled production process will lead to a highly reliable product. This reliability, however, remains unquantified.

The reliance on process monitoring has the potential of straining agency resources in the regulatory process if highly capable tools are not available to the agency. Complicating the design and development of such regulatory tools is the pace of innovation in the digital electronics industry. Innovations of revolutionary natures are taking place at rates far greater than those to which the regulatory framework can respond. Furthermore, threats to digital systems are greatly expanded with the emergence of the so-called "cyber security" threat posed by the malevolent or just the prankster.

There is a vast literature on software reliability. This literature surrounds two schools of thought. One school holds that software reliability can be quantified and methods can be developed for doing so. The other school of thought views software reliability as meaningless by itself. Embedded software failures are only one contribution to the unreliability of the overall system. The staff has judged that these models have not reached a level of maturity that allows them to be used in the regulatory process. The ACRS agrees.

The NRC research activities in Digital I&C Systems are listed in Table 3. Project K6907 is intended to provide a much needed assessment of the merits and demerits of the modeling methods of the two schools of thought. Within this Project, the staff plans to develop and test methods and models for integrated assessment of digital system reliability. Project Y6591 is developing and testing methods for evaluating software reliability. Project Y6332 is an investigation of development of PRA models for digital systems and Project Y6472 is to investigate the integration of digital systems into the PRAs for current generation nuclear power plants. These Projects, if successful, will provide the basis for including software reliability in PRAs, thus reducing the need for relying on controlling the design and development process and enhancing the ability of the agency to risk-inform its regulations.

Studies of digital system performance and reliability appear to involve a larger issue than any one agency can undertake in the face of an innovative digital electronics industry. The staff is working in several areas to try to tap into this wider research, including serving on the steering committee of the DOE I&C and Human Machine Interface advisory committee. The staff is also in the process of developing an international collaborative program and a potential research effort in this area is digital system reliability modeling. The RES staff also chairs the OECD/NEA COMPSYS group, which is charged with developing nuclear power plant specific digital systems failure data. The ACRS encourages these activities. However, it is not evident that the Project at Halden (Y6349) directly addresses the NRC needs.

Two Projects (Y6371 and Y6410) address the issue of aging of cable systems in existing nuclear power plants. Demonstration that cables in a power plant have not aged to the point that plant safety is threatened is the responsibility of the licensee. The NRC needs to understand the issue well enough to ensure that adequately realistic conditions for aging rather than unnecessarily conservative conditions are taken into consideration and that proposed technical responses to the issue are adequate. Consequently, these studies can be brought quickly to a resolution sufficient for regulatory needs.

Similarly, the study of electromagnetic conditions at nuclear power plants (Y6272) has been under way for sometime. An adequate understanding should be soon available to review licensees' arguments that electromagnetic conditions do not threaten plant safety. Therefore, this Project should be concluded in FY 2004.

Several Projects address the issue of cyber security of digital systems (Y6729, Y6712, Y6733, and Y6734). Again, the issue of cyber security seems bigger than what any one government agency can confront, least of all an agency as small as the NRC. RES should explore the possibilities of leveraging the resources it has for cyber security by joining with other government agencies in the study of this issue.

Table 3. Research Activities in Digital Instrumentation and Control Systems

Project	Title	Comment
K6907	***Digital System Performance and Reliability*** A survey of the various modeling methods for digital systems	Program is intended to provide an unbiased assessment of the merits and demerits of modeling methods in the two main schools of thought concerning software reliability. The staff plans to develop within this activity a tool to assess the reliability of an integrated digital system based on fault-tolerant system methods.
Y6475	***Wireless*** Anticipatory research on the potential challenges and regulatory issues of using wireless communication technologies in nuclear power plants	Wireless communication may resolve some fire issues but is very susceptible to cyber attack. There is interest within the nuclear industry in using this technology and NRC research is being done in anticipation of an industry proposal.
Y6590	***Review IEEE Standards for Endorsement*** IEEE Standards are the bases for most Regulatory Guides dealing with issues of digital instrumentation and control	An essential activity since these standards will be the bases for regulatory review of the performance and reliability of digital systems.
Y6332	***Digital Systems Risk*** Investigation of digital I&C system analysis methods	Essential to developing the technology to include digital systems reliability in PRAs.
Y6371	***Risk Associated With Cable Aging***	A simple effort that ought to be concluded in FY 2004.
Y6410	***Collaborative Research on Wire System Aging*** Collaborative research on current conditions, residual lifetime and fire risk of cables	Progress should be monitored to ensure research does not stray beyond needs of the regulatory process and does not enforce overly conservative constraints on licensees.
Y6272	***Characterize Electromagnetic Conditions at Nuclear Power Plants***	Second year of a two-year effort. Should be concluded in FY 2004.
Y6349	***Halden Environmentally Assisted Cracking***	Despite the title, this is I&C; it is not evident that the current Halden program in this area directly addresses NRC needs.
Y6472	***Risk-Importance of Digital Systems***	To investigate integration of digital systems in PRAs.
Y6591	***Software Reliability Code Measurements***	Development of measures to quantify the reliability of safety system software.

Table 3. Research Activities in Digital Instrumentation and Control Systems[3] (Continued)

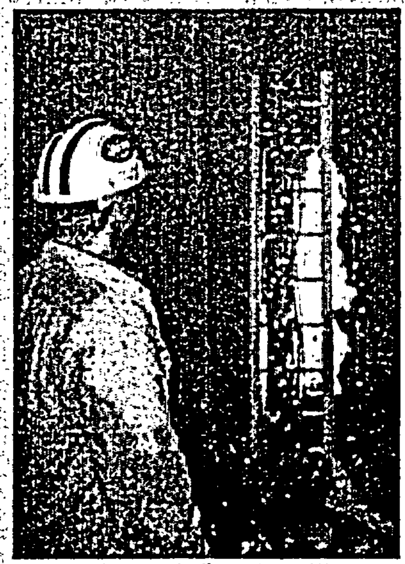
Project	Title	Comment
Y6729	*Classification of Digital System Vulnerabilities*	Vulnerability to cyber attack
Y6712	*Safety System Isolation Study*	Generically qualified systems vulnerability to cyber attack
Y6733	*Protocol Robustness Analyses*	Anticipatory research on the vulnerability of the industry to cyber attack
Y6734	*Security Tool Vulnerability Case Study*	Adequacy of commercial off-the-shelf tools for preventing cyber attack

[3]See also Y6651, Effects of switchgear aging on energetic faults, Table 4

6 FIRE SAFETY RESEARCH

As the NRC moves from a deterministic regulatory process to one based on an ever greater use of information from quantitative risk assessment, it is inevitable that some safety issues of the past will wane in importance and other issues will be found to be of greater importance than previously anticipated. Several industry-sponsored full-scope risk assessments performed about 20 years ago, as well as the insights gained recently from the Individual Plant Examination of External Events (IPEEE) Program [Ref. 8], have shown that fire is a more important accident initiator than might have been concluded in the past. The risk from fire-initiated accidents is significant at some nuclear power plants in both an absolute sense and in comparison to accident initiators internal to the plant equipment and operations.

The importance of fire-initiated accidents was made profoundly evident by the 1975 fire at the Browns Ferry plant. Requirements for fire protection beyond common industrial fire protection practices were imposed on licensees as backfits and compliance with these requirements has been achieved at considerable expense. The agency invested heavily in experimental and analytical tools for the analysis of fire consequences at nuclear power plants. Since these early efforts, progress has not been commensurate with the major strides made in the quantification of the probabilities of accidents caused by internal initiators or the prediction of the progression of such accidents. Fire risks today are typically assessed using what may be considered at best semiquantitative bounding methods. Fire progression models used in safety assessments are decades old and do not compare favorably with the state-of-the-art methods available in other technical



NEI-EPRI-NRC Tests of Circuit Faults In Fires

NEI, EPRI, and NRC conducted fire tests to investigate the probabilities of failures, shorts to the ground and "hot" shorts in instrumentation, control, and power circuits of nuclear power plants during fires. The NRC also completed a review of cable fire testing and operational experience with serious nuclear plant fires including the Browns Ferry fire in 1975 and several fires in foreign nuclear power plants (NUREG/CR-6834 [Ref. 9]). Since then, a method for incorporating circuit analyses into risk assessments is being used by NRC and EPRI to "re-quantify" risk at two representative US nuclear power plants.

areas. Fire-induced faults in I&C systems (hallmarks of the Browns Ferry fire) are still assessed manually and incompletely despite the long-held belief that such analyses could be effectively done by computer. Currently, there is no capability to include fire as a consequence of accidents initiated by other means in PRAs.

The current NRC research efforts in fire safety (See Table 4) seem incongruent with the estimated risk significance of fire. Fire modeling benchmark and validation studies (Y6688) are intended to show where agency models stand relative to the state-of-the-art models rather than advancing the modeling capabilities. The fire risk 'requantification' studies being done cooperatively with the Electric Power Research Institute (EPRI) will hone approximate risk assessment methods rather than making these methods commensurate with risk assessment methods available for accidents initiated by internal events. Lack of adequate risk information for fire comparable to that provided by the NUREG-1150 study of internal initiators [Ref. 10] will continue to hamper the significance determination process for deficiencies in the licensees' fire protection programs especially if these programs are guided by the National Fire Protection Association (NFPA) standard 805 [Ref. 11] rather than Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."

The limited fire research effort in comparison to the estimates of risk posed by fire is understandable since resources for such research have been diverted to respond to the events of September 11, 2001. Now that these responses are being completed, the agency should revitalize its fire safety research efforts and move the technical capabilities of the agency to be more in line with the risks being ascribed to fire. The revitalized efforts should include:

- Automated methods for associated circuits analysis
- Improved models of fire progression and damage
- Better integration of fire risk, including induced fire, in plant PRAs
- Development of methods to estimate the reliability of human performance under fire conditions

Table 4. Fire Safety Research Activities[4]

		Project
Y6037	***Fire Risk Assessment*** • Tools for circuit failure mode and likelihood analysis • Tools for fire detection and suppression analysis • IEEE-383 rated cable self-ignited fire frequency analysis: feasibility study • Fire modeling toolbox: input data and assessment • Experience from major fires • Cable failure mode likelihood studies • Fire risk 'requantification' studies • Fire significance determination process • Fire risk assessment tools precursor analysis • Support for ANS fire PRA standard	A Program with a large number of important elements for fire safety. Activities support the ROP and improved risk analysis of nuclear power plants.
Y6688	***Fire Model Benchmarking and Validation***	Assesses the current status; should be used to define what improvements agency needs in its modeling capabilities.
Y6651	***Effects of Switchgear Aging on Energetic Faults***	Deals with a significant fire protection issue.

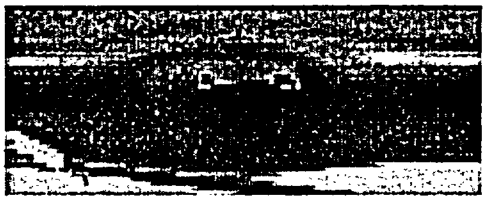
[4] See also Collaborative Research on Wire System Aging (Y6410) in Table 3 and Low-Power and Shutdown Risk Study - Level 1 (W6904) in Table 9.

7 REACTOR FUEL RESEARCH

Reactor Fuel Research at the NRC is completing a quintessential example of risk-informed, confirmatory research. By now, the basis of the work is well known though it still provides an object lesson in the need to sustain technical expertise in selected areas of research crucial to the agency's mission.

In the past, the NRC research provided data and models of fuel behavior under off-normal and accident conditions while fuel vendors and licensees focused their attention on fuel behavior during normal operations. When typical fuel burnups at the end of life were less than about 30 GWd/t, a point was reached in the NRC work that understanding of fuel behavior was thought to be sufficiently sophisticated and there was a confidence that licensee proposals to use fuel at higher burnups could be evaluated by extrapolation with existing fuel models. As a result, the NRC-sponsored research in fuel performance was greatly curtailed in favor of other priorities. End-of-life fuel burnups crept upward even though it was known in the reactor fuel community that microstructural changes in the fuel occurred at average burnups exceeding about 52 GWd/t.

Testing of high-burnup fuels in France and later in Japan and Russia [Ref. 12] showed that high-burnup fuels were susceptible to damage and there was the potential for loss-of-fuel coolability during reactivity-initiated accidents. In response to these experimental findings, the NRC made the regulatory decision that adequate safety would be preserved by limiting fuel burnups to 62 GWd/t unless requests for higher burnups were supported by persuasive evidence of adequate fuel capability. A confirmatory research program was initiated to substantiate this regulatory decision.



Ballooned and ruptured high-burnup fuel rod following a LOCA test and subsequent testing.

NRC is using these tests of high-burnup fuel, along with extensive metallurgical and materials testing, to confirm regulatory decisions on the safe use of reactor fuels at burnups of up to 62 GWd/t. These tests are part of a larger international effort to examine the effects of high fuel burnup and new fuel claddings on reactor behavior under design basis and severe accident conditions. Testing could include integral tests of degradation and fission product release.

Though the technical issues presented by high-burnup fuel dealt with design basis accidents, the scope of the confirmatory research was designed by using risk information. This risk information pointed to PWR reactivity accidents initiated by control rod withdrawal, LOCAs and anticipated transients without scram (ATWS) in BWRs to be of risk concern with respect to high-burnup fuel. On the other hand, BWR rod drop reactivity accidents were found not sufficiently probable to merit explicit experimental investigation.

This risk-significant research has also been well organized. Technical issues and considerations were identified using a prestigious panel of international experts in comprehensive phenomena identification and ranking efforts. Both the vulnerabilities of the fuel and the probabilities of threats have been considered in these efforts. The research itself was organized in several cooperative agreements with other Nations, which greatly leveraged the resources the NRC was able to devote to this effort.

Research on high-burnup fuel behavior during reactivity-initiated accidents is now reaching a conclusion with the definition of usable criteria for fuel performance. The fuel performance computer codes, FRAPCON and FRAPTRAN, have been upgraded to treat fuel taken to elevated levels of burnup.

High-Burnup Fuel Research efforts are now focusing on fuel behavior under LOCA and ATWS conditions. In connection with these research efforts (See Table 5), the ACRS makes the following recommendations:

- The NRC should withdraw from the CABRI water loop Project (W6832) if tests planned in this area cannot be changed to involve more appropriate pulse shapes and energies for identifying cladding failure thresholds.

- Single rod tests of high-burnup fuel behavior under LOCA conditions should continue and should include the "quenching" phase of the hypothesized scenario. Researchers should consider whether bounding peak clad temperature conditions should be imposed on the single rods or whether more realistic but limiting single rod (or assembly) temperature scenarios are more meaningful.

- If single rod tests continue to show behavior that is not extraordinarily different than anticipated based on low-burnup fuel behavior, considerations should evolve from issues of fuel embrittlement to issues of fuel coolability. The NRC should consider joining the international efforts being planned for in-pile tests of multi-rod fuel bundles under LOCA conditions.

- Research on mechanical properties and oxidation kinetics of high-burnup cladding, including Zirlo and M5 cladding (Y6367) should continue. This research will allow amending 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," so that alternative fuel claddings that are important both to the nuclear industry and to the NRC can be used. Research on E110 cladding (Y6723) should not be an emphasis for the NRC since there are no indications that the licensees will use this cladding. Limited efforts to understand why this alloy performs poorly in comparison to the compositionally similar alloy M5 might be of use, but should not require continued use of E110 in tests to characterize alloy response to accident conditions.

- Knowledge and understanding that has been gained in the resolution of the reactivity-initiated accident issues as well as in the resolution of the LOCA issues of high-burnup fuel may make it possible to resolve by analysis the ATWS issues of high-burnup fuel without explicit testing of ATWS scenarios. This possibility should be explored by RES.

- Plans to upgrade fuel performance codes (Y6580) to address MOX fuel should be pursued. The MATPRO database documentation should be updated to reflect modern data actually being used in the fuel performance codes.

Understanding of reactor fuel is crucial to the NRC mission. It is widely anticipated that the nuclear industry will be proposing further changes in fuel cladding and fuel burnup. Consequently, the agency must maintain expertise in the reactor fuel area. Expertise outside the agency and its contractors are probably not sufficiently independent of fuel vendors and licensees to be used on an as needed basis to review fuel proposals by licensees. A sustained NRC research effort in reactor fuel behavior under accident and off-normal conditions is especially needed as competitive pressures force fuel vendors to curtail their research efforts. Continuing expertise and even additional research may be needed if, as now expected, licensees make requests for extending fuel burnups up to and beyond 75 GWd/t. It is quite likely that such industry proposals will be substantiated by minimal experimental research. It is important for the NRC to have a technically justified position on the experimental research that is necessary to support such proposals especially in light of the "surprise" that accompanied extending fuel burnups into the range of 50 to 60 GWd/t.

Table 5 Research Activities in Reactor Fuel

Project	Title	Comment
W6832	***CABRI Water Loop*** International program for in-pile testing of high-burnup fuel behavior during reactivity-initiated accidents	This Project's focus has changed at the behest of partners and no longer meets regulatory needs in reactivity accidents. NRC should withdraw from this activity if the tests planned for this facility cannot be changed to involve more appropriate pulse shapes and energies for identifying cladding failure thresholds.
Y6195	***Dry Cask Storage License for High-Burnup Fuel***	Behavior of high-burnup fuel in dry cask storage.
Y6367	***High Burnup Cladding Performance*** Testing of high-burnup fuel clad properties	An essential Project to verify efficacy of the existing fuel damage criteria for high-burup fuel operation. Results provide access to data from foreign programs.
Y6580	***Fuel Code Applications for High Burnup Fuel*** Upgrade the fuel behavior codes FRAPCON and FRAPTRAN for high burnup fuel	FRAPCON and FRAPTRAN are essential regulatory tools and need to be kept up to date as licensees propose ever more aggressive use of fuel.
Y6586	***Fuel Code Assessment for MOX Fuel*** Modification of FRAPCON and FRAPTRAN for MOX fuel to dispose of excess weapons grade plutonium	FRAPCON and FRAPTRAN will be used extensively to assess proposals for using MOX fuel in commercial PWRs.
Y6723	***International Agreement on Fuel Behavior and Materials Science Research***	Research on boron dilution accidents and on E110 cladding. Since there are no indications that the licensees will use the E110 cladding, research on this cladding should not be an emphasis for the NRC. Limited efforts to understand the poor performance of this alloy compared to the similar alloy M5 might be of use.

8 NEUTRONICS AND CRITICALITY SAFETY

Neutron transport modeling and criticality safety are closely related to fuel safety but are distinguished here. These technical areas are also crucial to the agency mission. It is essential for public confidence, if nothing else, that NRC has the capability to independently ensure neutronic safety of reactor cores and fissile materials. The tools that the NRC uses for these independent evaluations must be maintained at the state-of-the-art level.

The NRC research Projects in Neutronic Analysis, Core Physics, and Criticality Safety are listed in Table 6. The PARCS code, an analytical tool for neutronics analysis, is also included in Chapter 15, Thermal-Hydraulics Research, of this report.

The ACRS assessment is that RES is doing a commendable, cost-effective job in maintaining the capabilities of its neutronics and core physics codes and the associated database. Upgrades to these codes to address both extended fuel burnup and MOX fuels are needed and these needs are addressed in the research Projects. The ACRS anticipates that these extended capabilities will be tested significantly in the certification of the advanced reactor designs. The effort to provide guidance for high-enrichment fuels (Y6510) reflects a belief that licensees will indeed propose eventually to extend fuel burnups beyond 75 GWd/t as discussed in Chapter 7, Reactor Fuel Research, of this report. Indeed, DOE is sponsoring research to examine fuels enriched to 7 percent.

Table 6 Research Activities in Neutronics Analysis, Core Physics, and Criticality Safety

Project	Title	Comment
Y6320	*NEWT Lattice Code*	Lattice physics cross-sections for MOX and high-burnup fuels.
Y6403	*Reactor Core Analysis*	To compare the PARCS code predictions for MOX to French data.
Y6587	*Reactor Analysis for High-Burnup Fuel*	Analyses with PARCS code of high-burnup fuel for transients and accidents involving rapid reactivity changes.
Y6685	*Upgrade Neutronic Code - Nuclear Fuel Comp/Safety Assessment*	To upgrade ORIGEN source model for MOX and conventional fuels.
Y6771	*MOX Neutronics*	To upgrade PARCS code for MOX fuel
Y6510	*Extend Fuel Enrichment*	Anticipatory research for code and guidance for fuel enrichments of 5-10%.
Y6517	*High-Burnup Source Term for Storage*	To develop chemical and radioassay data for high-burnup fuel to revise guidance on decay heat and shielding for transport and storage casks.

9 HUMAN FACTORS AND HUMAN RELIABILITY RESEARCH

The ACRS continues to believe that human performance issues will be important for the continued safe operation of nuclear power plants. The quantification of human reliability under accident conditions will continue to be among the most challenging aspects of quantitative risk assessment. Consequently, it is important for the NRC to maintain an active research program both in the areas of Human Factors and Human Reliability Analysis (HRA). The ACRS believes that issues surrounding human performance in teams (such as the team of control room operators) and human performance in organizations, including the question of organizational reliability, are important and need to be considered in ongoing Human Performance Research planning. These are both important to "safety management", a concept that Chairman Diaz discussed recently [Ref. 13].

The NRC research activities in the areas of Human Factors and Human Reliability Analysis are listed in Table 7. Human Factors is a broad discipline that extends well beyond the nuclear arena. The NRC research strategy with respect to human factors is to remain aware of pertinent developments in the field and based on this awareness, provide tools and guidance to line regulatory organization to facilitate the regulatory process. The NRC remains aware of developments in the international nuclear community through its participation in the international cooperative Halden Project (B7488) and the ACRS continues to support participation in this Project. The NRC's capabilities in the area of human factors are likely to be tested in the near future as plant staffing requirements are challenged certainly by advanced reactor designs proposed for certification and, perhaps, even by existing reactors searching for competitive advantage in a deregulated energy market. The NRC must have defensible bases for its plant staffing requirements for nuclear power plant operations and security. Research is beginning to address these staffing issues (Y6630). The attention NRC is paying to Human Factors Research is yielding dividends. The Standard Review Plan Chapter 18, "Human Factors Engineering," has recently been updated to reflect developments especially in the area of digital systems. A risk-informed method to screen licensee submittals for human factors review has the potential of better focusing agency resources on risk-significant human factors issues.

The Human Reliability Research at the NRC is undertaking a potentially significant effort to establish the basis for crediting operator actions at nuclear power plants (Y6022). Risk assessments now credit only proceduralized actions for which operators are trained. It is, however, well known that operator actions outside of normal procedures can provide significant safety benefit. This research holds, then, the promise of adding greater realism to quantitative risk assessments.

Quantification of human reliability remains a troublesome feature of risk assessments. Numerous plausible approaches have been devised that often yield disparate results. The database for the validation of these approaches remains distressingly limited. Uncertainties from both parameters and the models are seldom evaluated. The results of the benchmark exercise conducted by the Ispra Laboratory of the European Union [Ref. 14], although admittedly fairly old by now, are troubling. These results show that the choice of HRA model has a significant impact on the results. This is one of the few areas in Level 1 PRA in which model uncertainty is significant. The ACRS has not seen a critical review by the NRC of the merits and disadvantages of the existing models. Often in the regulatory process simple, transparent methods such as THERP that deal only with human errors of omission or industry-developed models that have not received extensive review are used. The regulatory process seems to accept these simple methods in the absence of guidance on the necessary levels of rigor and accuracy needed for quantification of human reliability. Acceptance of such simple models without substantiation by a robust database undermines any proposal to develop credible human reliability models.

The ACRS notes that RES is preparing a document on its assessment of "good practices" that will support all aspects of HRA, including the quantification process (W6994). These descriptions of good practices are needed to supplement information provided in the ASME PRA standard. The ACRS understands that the second phase of this project (planned for 2005) will be a review and evaluation of existing HRA approaches for their capability to meet the "good practices" when employed to address different regulatory applications. The ACRS anticipates that this review will address the important issue of model uncertainty.

The NRC is developing an events database called Human Event Repository and Analyses (HERA) (Y6123). This database may be a key step toward much improved analyses of human reliability in nuclear reactors. It should provide an useful basis for evaluating the model uncertainties mentioned above.

The NRC is experimenting with an ambitious methodology for estimating human reliability to assess errors of commission as well as errors of omission. This methodology called ATHEANA (W6994) focuses on the context that drives human performance ("performance shaping factors"). The ACRS has been critical of this approach in the past because of both its cumbersome, manpower-intensive character and its lack of quantification. The NRC is now applying this methodology to two important issues:

- Human performance during fires
- Human performance issues associated with pressurized thermal shock

The ACRS looks-forward to reviewing the results of these application efforts and learning whether the method has become easier to apply and whether it yields insights significantly better than those that could be derived from simpler methods. Of particular interest to the ACRS is the use of expert opinion elicitation in these applications to quantify the ATHEANA results.

The staff needs to show how the Davis-Besse incident has affected the set of performance shaping factors and what guidance is being provided to the ROP.

Table 7. Research Activities in Human Factors and Human Reliability

Project	Title	Comment
Y6123	*Retrospective Risk Analysis of Human Reliability in Operating Events*	Valuable Project to develop HERA database on human reliability from past operational experience.
Y6832	*Human Reliability Analysis for Byproduct Material*	Identifies human reliability issues for NMSS - tools of use for quantification of risk.
B7488	*HALDEN Reactor Project*	This is a long-term Project in Human Performance that is contributing to NRC work in human factors.
Y6022	*Credit for Operator Action*	This is a small effort to provide guidance for NRC reviewers. It has the potential of adding greater realism to PRA.
Y6630	*Development of Regulatory Guide and Analytical Techniques for Assessing Nuclear Power Plant Staffing*	An important proactive measure in anticipation of trends in current reactors and themes in the development of more advanced reactors.
W6994	*Review, Application and Refinement of ATHEANA*	This is a very ambitious effort to characterize human performance. At issue is whether ATHEANA can be applied practically to the issue of reactor safety. This effort will also yield guidance for performing and reviewing human reliability analyses and supplementing information provided in the ASME PRA standard.

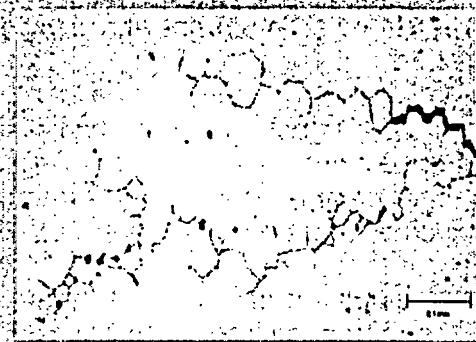
10 MATERIALS AND METALLURGY

Research Projects in the area of Materials and Metallurgy are listed in Table 8. These Projects can be grouped into the following broad categories:

- Reactor Pressure Vessel Embrittlement (W6953, Y6533, Y6378, W6986, Y6396, Y6638, Y6737)
- Neutron Fluence Determination (Y6391, Y6742)
- Environmentally Assisted Cracking (Y6536,Y6588, K6266, Y6270,Y6388, Y6722)
- Monitoring and Inspection (Y6534, Y6604, Y6649, Y6869, Y6882, Y6909)
- LOCA Frequencies and International Standards (Y6296, Y6538, Y6744)
- Proactive Materials Degradation Assessment (Y6919)

These Projects represent a well-balanced program that address current regulatory needs, including the need for the Office of Nuclear Reactor Regulation (NRR) to maintain an independent analytical capability in various areas of materials and metallurgy.

Capabilities are transitioned to NRR as they are developed, and new projects are introduced as their need becomes apparent. The ACRS recommendations on various Projects in the Materials and Metallurgy area are given below and summarized in Table 8. Many of these Projects are part of collaborative programs, involving about 65 International Organizations, thereby giving good leverage of the NRC resources.



Example of a Predominantly Axial, Intergranular Stress Corrosion Crack

The Projects dealing with reactor pressure vessel embrittlement should be considered both with respect to safety issues of reactor pressure vessel integrity and with respect to maintaining whatever core competency the agency needs in irradiation embrittlement. The staff needs to evaluate what research should be retained in order to meet these two needs. The remaining technical issue involves high-neutron fluence on the low-copper, high-nickel pressure vessel material found in just a few plants (W6953, Y6533). Work in Y6533 will support risk-informing the pressure- temperature limits in 10CFR Part 50, Appendix G, "Fracture Toughness Requirements." Project Y6638 provides support for revising Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

Project Y6378 ensures continued NRC participation in the International Atomic Energy Agency (IAEA) deliberations on pressure vessel integrity and will contribute to the preservation of core competency.

Peer review of the technical basis for the reevaluation of the PTS screening criterion (Y6737) is most important and should be continued.

Work in Project W6986 should be terminated since its objectives have been met and fracture mechanics technology development will be consolidated in Y6533.

The development of analytical capabilities for neutron fluence (Y6391,Y6742), especially in the core regions near the bottom and top of the core, are important since cracking of major welded subassemblies in BWRs has occurred in these regions. The ability of the NRC to independently check the licensees' arguments for disposition and repair of these components with complex geometries is important. Hence, the uncompleted tasks of these Projects should be continued in FY2004.

Corrosion, especially environmentally assisted cracking, is a complicated technical issue involving the interplay of chemistry, corrosion science, metallurgy, and mechanics. The current concerns about irradiation effects on cracking of core components further complicate this situation.

The management of environmentally assisted cracking incidents in LWRs is the prime responsibility of the licensees. However, the NRC must have independent capabilities to assess the licensees' positions regarding the structural integrity of reactors and their proposed mitigation actions. Indeed, recent events at the Indian Point and Davis-Besse nuclear plants have demonstrated that licensees' corrosion control programs merit continued scrutiny and monitoring. Consequently, the NRC should have active programs in the areas of materials degradation data collection (in order to assess the kinetics of degradation), monitoring and inspection (e.g., NDE), and codification.

A particularly vulnerable element of the reactor coolant system in PWRs is the steam generator. Rupture of steam generator tubes can lead to accidents that allow radioactive materials released from the core to bypass the reactor containment and enter into the environment. Risk analyses show that severe accidents involving containment bypass can be risk dominant at some PWRs. Modes of corrosion (e.g. cracking, denting) of the steam generator tubes have changed over the years as changes have been made in both the materials used for the tubes and in the PWR coolant chemistry. The complexity of these phenomena and the risk significance of tube failures make it important for the NRC to monitor these issues. This need does not abate as further changes from alloy 600 tubes to alloy 690 tubes are made by the industry. Although alloy 690 is more resistant to primary water stress corrosion cracking, the complexity of the environments possible on the secondary side of steam generators makes it impossible to preclude the potential for stress corrosion cracking. Therefore, the Projects (Y6588, Y6536) associated with the NRC's Steam Generator Tube Integrity Program should be continued.

In recent years, environmentally assisted cracking in LWRs has raised other concerns such as irradiation-assisted stress corrosion cracking (IASCC) of stainless steel core internals of BWRs and stress corrosion cracking /general corrosion of vessel head penetration (VHP) subassemblies in PWRs. These latter incidents have been the subject of several Bulletins and an Order. Consequently, it is appropriate for the NRC to sponsor research to independently evaluate licensees' plans to mitigate these phenomena. Projects Y6270 and Y6388 should be continued in FY2004 since they provide the essential cracking kinetics data required for in-service inspection (ISI) regulations. Project K6266 should also be continued since it relates to the quantification and understanding of IASCC issues. The value of this research is enhanced by collaboration with several International Organizations. Project Y6722 was instigated by the Davis-Besse incident and several of the tasks have been completed. This Project should be continued to complete the remaining tasks and to ensure adequate independent analysis of the licensees' extensive program spawned by this incident.

Nondestructive examination methods are crucial for monitoring the integrity of the reactor coolant system. These methods for detecting corrosion and cracking have improved greatly in recent years largely due to NRC research activities. However, the reliability of these methods in the field continues to be an issue. Sustained NRC research (Y6534, Y6604, Y6649, Y6869, Y6882, Y6909) on the evaluation of these nondestructive examination methods is crucial to the agency mission. Projects Y6649, and Y6869 merit increased attention; these relate to the inspection methodologies (technique and frequency) that should be used for VHP assemblies. In the latter project, a technical basis for reevaluation of requirements related to reactor coolant leakage will be developed.

A major NRC regulatory initiative is risk-informing 10CFR50.46 and redefining large-break LOCA in terms of LOCA frequency distributions. Part of this work involves the development a new probabilistic fracture mechanics (PFM) code for piping. The current PFM code, PRAISE, has been a valuable tool, but an improved code is needed to better address materials degradation modes. Thus, Project Y6538 should be continued. Project Y6744 maintains NRC access to the CSNI pipe failure database. This is a further input needed for risk-informing 10CFR50.46.

Project Y6296 involves NRC participation in the development of international standards and should be continued.

Over the years, the NRC has been surprised by a number of materials degradation issues. The ACRS supports the "Proactive Materials Degradation Assessment" Project (Y6919) recently initiated by the NRC in order to move from the economically inefficient reactive mode of regulating materials degradation issues. The plan to lead a PIRT effort with the industry over the next two years, with expert elicitation from international experts, should identify potential material degradations that may occur in the future. This knowledge will provide a better basis for future regulatory decisions on timely inspection and repair criteria.

Table 8. Research Activities in Materials and Metallurgy

Project	Title	Comment
Reactor Pressure Vessel Embrittlement		
W6953 Y6533	*Heavy-Section Steel Irradiation Program* *HSST-3*	The remaining technical issue involves high-neutron fluence on the low-copper, high-nickel pressure vessel material found in just a few plants. Work in Y6533 will support risk-informing the pressure-temperature limits in 10CFR Part 50, Appendix G.
Y6378	*International Pressure Vessel Technology Cooperative Program*	Representation of NRC at IAEA Division of Nuclear Power on RPV structural integrity issues.
W6986	*Fracture Mechanics Technology for LWR Materials*	Should be terminated since its objectives have been met and fracture mechanics technology development will be consolidated in Y6533.
Y6396	*Radiation Embrittlement Damage Analysis & Prediction*	Longer term confirmatory research with current emphasis on embrittlement at end-of-life fluences.
Y6638	*Statistical Analysis of RPV Steels*	Longer term support for rev 3 of RG 1.99
Y6737	*Peer Review of PTS Technical Basis*	A critical Project at the end of a very successful program on reevaluation of the PTS screening criterion.
Neutron Fluence Determination		
Y6391	*Boiling Water Reactor Fluence*	Support for development of independent NRC analysis of Industry code for neutron fluence. Uncompleted tasks of this Project should be continued in FY2004.
Y6742	*BWR Reactor Vessel Samples*	Measurement of helium and boron concentrations in BWR core samples and correlation to fluence. Critical to weld repair decisions. Uncompleted tasks should be continued in FY2004.

Table 8. Research Activities in Materials and Metallurgy (Continued)

Project	Title	Comment
Environmentally Assisted Cracking		
Y6536	***PWR Primary System Components'* Behavior Under Severe Accident Loads**	Addresses the potential for steam generator bypass in severe accidents and should be continued.
Y6588	***Steam Generator Integrity Program -3***	Addresses potential degradation modes for new, degraded or repaired tubes and an assessment of their consequences (e.g. leak rates) and control (e.g., ISI) techniques. Should be continued.
K6266	***CIR-II Cooperative Agreement***	Cooperative program, organized by EPRI to predict the irradiation assisted cracking characteristics of new and existing LWR core materials. Good leverage of NRC funds. Should be continued.
Y6270	***Environmentally Assisted Cracking***	Evaluation of IASCC and fracture behavior of stainless steels used BWR core structures. Relevant to license renewal. Should be continued.
Y6388	***Environmentally Assisted Cracking in LWRs***	Quantification of a variety of cracking phenomena (corrosion fatigue, IASCC) to develop independent life prediction capability for low alloy steel, stainless steel and nickel base alloys in LWRs. Should be continued.
Y6722	***Degradation of RPV in Boric Acid***	Project instigated by Davis-Besse incident. Several tasks have been completed. Should be continued to complete the remaining tasks.

Table 8. Research Activities in Materials and Metallurgy (Continued)

Project	Title	Comment
Monitoring and Inspection		
Y6534	*Piping - Non-Destructive Examination (NDE) Reliability*	Cooperative international program to develop NDE methods. Should be continued.
Y6604	*Evaluation of the Reliability of NDE Techniques*	Evaluation of effectiveness, reliability and adequacy of advanced NDE methods necessary for e.g. cast stainless steels. Should be continued.
Y6649	Alloy 600 Cracking - Phase II	Merits increased attention. Relates to inspection methodologies to be used for VHP assemblies.
Y6869	*Barrier Integrity Research Program*	Merits increased attention. Provides technical basis for reevaluating the requirements of reactor coolant leakage.
Y6882	*Technical Assessment of Bare Metal Inspection Techniques*	Evaluation of the inspection techniques to be used for PWR bottom head penetrations. Should be continued.
Y6909	*Examination of North Anna Unit 2 Head*	Establish, via inspection of nozzles from North Anna Unit 2 and Davis-Besse, the development of the leak path as cracking/degradation occurred. Relates to adequacy of BMI.

Table 8. Research Activities in Materials and Metallurgy (Continued)

Project	Title	Comment
LOCA Frequencies and International Standards		
Y6296	*ISO Participation*	Ensure NRC participation in the development of international standards. Should be continued.
Y6538	*Technical Development of LOCA Frequency Distributions*	Support for risk-informing 10CFR 50.46 and redefining LBLOCA. Should be continued.
Y6744	*CSNI Piping Data Base*	Cooperative program to maintain NRC access to CSNI pipe failure database. Should be continued.
Proactive Materials Degradation Assessment		
Y6919	*Materials Degradation PIRT*	Project recently initiated by the NRC. The plan to lead a PIRT effort with the industry over the next two years, with expert elicitation from international experts, should identify potential material degradations that may occur in the future. This knowledge will provide a better basis for future regulatory decisions on timely inspection and repair criteria.

11 PROBABILISTIC RISK ASSESSMENT

As the NRC moves toward a risk-informed regulatory system, it will need ever greater amounts of risk information derived from quantitative risk assessments. It will want to maintain its own independent capabilities for conducting PRAs at state-of-the-art levels. A continued vital research program in PRA is, then, crucial to the agency mission.

The NRC research activities in the area of PRA are listed in Table 9. A major thrust is the support for the ROP which is a major NRC initiative in the use of risk information for monitoring the operations of nuclear power plants. Support for the development of the SAPHIRE code suite and the SPAR models is important and continuing (Y6394, W6355, W6467, Y6153, Y6553). Also under way in support of the ROP is the development of objective, risk-based, performance indicators (J8263, Y6370). A candidate metric called the Mitigating System Performance Index is being tested. There is also continued support for the significance determination process (Y6553) which is an important step in the assessment of findings from the ROP that continues to challenge the agency.

Another major element of the current PRA Research is the expansion of the method into the analysis of dry cask storage of spent fuel(Y6423, Y6502, Y6612). Such storage of spent fuel will become ever more necessary as the development of a permanent geologic repository for spent fuel continues to be delayed.

Research activities of importance are also under way to support risk-informed revisions to 10 CFR 50.46 (W6224, Y6538).

In addition to the many demands on resources for the application of PRA methods, resources are still made available for developing agency capabilities in PRA (K6007). The ACRS generally supports the idea that the NRC needs to maintain its PRA capabilities at or near the state-of-the-art level. A detailed examination of the tasks funded within this development effort shows that many overlap with distinct efforts in other fields. Some items with substantial overlap include the study of turbulent mixing in complex geometries and software reliability. Greater coordination among research within the NRC may eliminate the need for such overlapping research efforts.

The ACRS would prefer to see work to expand the scope of the NRC PRA capabilities. Work to develop capabilities to analyze risk during low-power and shutdown conditions at nuclear power plants (W6904 and Y6103) is very important. The ACRS strongly supports these efforts that will extend PRA capabilities toward the eventual goal of handling all modes of plant operation. Work to better integrate fire-risk assessment, including fires induced by other accident initiators, is a need not now being met despite evidence of risk significance coming from the IPEEE Program.

Altogether the NRC has an impressive PRA Research Program. This Program is all the more impressive since many of the agency resources in risk analysis are being applied to security issues that are not addressed in this report.

Table 9. Research Activities in Probabilistic Risk Assessment[5]

Project	Title	Comment
Y6423 Y6502	*Dry Cask PRA* *PRA for Dry Cask Storage - Peer Review*	Expanded application of PRA into an issue that will be of growing importance over the next few years.
Y6612	*Reliability of a Secondary Containment Isolation (SCI)*	Radionuclide release paths for accident involving drop of a dry spent fuel storage cask.
J8263 Y6394 W6355 W6467 Y6153 Y6553	*Reactor Oversight Process Support* *Maintain and support SAPHIRE Code and Library for PRA* *SPAR Model Development: Low Power/Shutdown* *SPAR Model Development: Level 1 Rev 3* *SPAR Model Development: Level 2/LERF* *Significance Determination Process Front-end Interface for SPAR Models*	Essential support for the Reactor Oversight Process.
Y6370	*Development of Risk-based Performance Indicators*	Exploratory activity that has yielded a candidate metric that is now being tested.
Y6522	*Synergistic Effects of Power Uprates*	Important effort to identify issues not revealed in the examinations of individual changes to the licensing bases of nuclear power plants.
K6007	*Probabilistic Assessment and Applications* • Model and parameter uncertainty • Software reliability • Thermal-hydraulic uncertainty • Turbulent mixing within complex geometries • AP1000 LBLOCA uncertainty analysis • Importance analysis • Causal modeling	Program of several activities to improve PRA capabilities. Greater coordination among research within NRC may eliminate the need for overlapping research efforts in this Project.

[5] See also Y6332, Digital Systems Risk, Table 3

Table 9. Research Activities in Probabilistic Risk Assessment (Continued)

Project	Title	Comment
W6144	*Technical Support in Risk Assessment*	Staff augmentation
W6224 Y6538	*Risk-informing 10 CFR Part 50* *Technical Development of LOCA Frequency Distributions*	Support for the effort to risk-inform 10 CFR Part 50
W6904 Y6103	*Low-Power and Shutdown Risk Study - Level 1* Low-Power and Shutdown Risk Study-Level 2	Essential study of one of the important elements of risk posed by nuclear power plants during low-power and shutdown operation that has been inadequately studied in the past
W6970	Support to Develop Consensus PRA Standards	Necessary NRC participation in standards setting activities
Y6335	Risk-Informed Initiatives for Nuclear Materials	Safety goals and guidance for nuclear materials and waste
Y6430	*PTS Risk Assessment*	Activities in support of the agency's reevaluation of the PTS screening criteria
Y6486	*Severe Accident Initiated Steam Generator Tube Rupture Sequences*	Essential element of the agency strategy to resolve issues of steam generators

12 RADIATION PROTECTION

The NRC research activities in the area of Radiation Protection are listed in Table 10. These activities constitute a well-leveraged program that allows the NRC to use quality data and information as the foundation of its basic radiation protection regulation, 10 CFR Part 20. A major thrust of these activities is the collection of occupational exposure data (Y6133 and Y6698). Another important thrust is the support for commissions (ICRP and NCRP) that establish radiation protection standards (G6071, G6251, G6589). There are also activities that support dose assessment by the NRC (Y6433, Y6460, Y6470). These are all essential activities for the agency mission and should be sustained.

A potentially important activity (Y6330) would harmonize requirements from multiple agencies for complex site assessments. This is a laudable joint effort with the Environmental Protection Agency (EPA) to develop common databases for use in environmental dose assessment calculations. Another cooperative effort (Y6470) is yielding a modeling platform that gives access to a wide range of system and process models as well as databases used by other government agencies. This platform will eventually yield a powerful environmental modeling capability for use in the evaluation of complex, contaminated sites.

Table 10. Research Activities in Radiation Protection

Project	Title	Comments
G6071	***BEIR - VII*** The Phase II study deals with low doses and low dose rates. The BEIR-VII effort is to conduct a comprehensive reassessment of health risks resulting from exposure to ionizing radiation since the 1990 BEIR-V [Ref. 15] report	NRC contribution to the development of consensus standards in radiation protection.
G6251	***Radiation Protection Standard Development*** ICRP task groups will provide information on radiation-associated risks of cancer and severe heredity effects, application of recommendations in terms of secondary limits, and the principles of optimization and management of radiological protection	ICRP support; ICRP is considering the revision of ICRP-60 [Ref. 16] by 2005 that simplifies the current system of radiological protection.
G6589	***NCRP Study on Radiation Protection Issues***	NCRP support.
Y6133	***Collection and Analysis of Occupational Radiation Exposure Data***	Analysis of required licensee submittals on occupational exposures.
Y6298	***Technical Bases Information Clearance***	Information to support Commission decisions on rulemaking.
Y6330	***Support for Interagency Cooperative Research***	Interagency harmonization of complex site evaluations.
Y6407	***NUREG-1640 [Ref. 17] Finalization - Clearance***	Development of conversion from dose to concentration for materials
Y6433	***Extremity Code*** This Project is to correct errors and limitations in the VARSKIN Mode 2 computer code identified by licensees	Update the VARSKIN code to confirm licensee skin dose assessments.
Y6460	***Update Codes for Assessing Radiation Doses***	Update dose coefficients
Y6470	***FRAMES Software Development***	Well-leveraged effort to provide agency a dose assessment tool for complex situations. Good example of interagency cooperation.
Y6698	***Support NEA Information System on Occupational Exposures*** Data are used to compare U.S. and foreign nuclear power plant initiatives and identify best practices during maintenance and operational procedures	Essential data for agency mission.

13 SEISMIC RESEARCH

Seismic events produce a baseline risk for nuclear power plants. The NRC has invested heavily in the understanding of seismic issues and development of its rules to ensure that seismic risks are acceptably low. Licensee submittals concerning risks associated with external accident initiators such as seismic events suggest these efforts have been largely successful. Issues concerning the estimation of seismic hazards of power plant sites in the eastern United States based on methodologies developed independently by EPRI and NRC have been resolved. The NRC research activities in the seismic area are listed in Table 11.

Much of the NRC research into seismic issues involves cooperative efforts to address essential seismic data. This work is highly leveraged by contributions from other partners. There is advantage in maintaining modest research efforts to accrue benefits from cooperative research on seismology and seismic engineering.

Projects W6829 and Y6718 are addressing specific issues that should be done by licensees. It is not clear that the NRC needs an independent assessment of these issues. Unlike many other issues that are dealt with by the NRC research, there is in this Country a very substantial engineering and scientific community addressing issues of seismic hazards and the engineering of structures to avoid seismic damage. By and large, this community is independent of the NRC licensees. Thus, the NRC-sponsored research is not needed to maintain a core competency in this area. If there are open issues associated with licensee submittals such as the seismic behavior of spent fuel casks or buried structures, the NRC could rely on use of engineering consulting firms on an as-needed basis to deal with such issues.

A Success Story
Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts

Early industry-sponsored PRAs, such as those for Zion and Indian Point, showed the significance of seismic contributions to risk from nuclear power plants. Later studies confirmed these findings. A difficulty in quantifying seismic risk, especially for plants east of the Rocky Mountains, was the sparsity of data. This led to extensive use of expert judgment with attendant large uncertainties in the analyses. Important probabilistic seismic hazard analyses sponsored by the NRC and by EPRI in the 1980s using different experts and different methods for eliciting and processing expert judgments produced very different results.

The controversy associated with these divergent results was not conducive to public confidence in the regulatory process. To redress this situation, NRC, the Department of Energy, and EPRI formed the Senior Seismic Hazard Analysis Committee (SSHAC) to provide a procedure for obtaining reproducible results. In a two-volume report, SSHAC identified key issues regarding the quantification of uncertainties and the utilization of expert opinion.

A National Research Council Panel reviewed this work and concluded that "... the SSHAC report's discussions and recommendations on uncertainty and the use of experts are quite independent of PSHA and can be applied to other types of risk analysis. The Panel believes that the SSHAC report makes a solid contribution to the methodology of hazard analysis, especially in the use of expert opinion" [Ref. 18].

Table 11. Research Activities in Seismic Phenomena

Project	Title	Comments
W6081	*Japanese Collaboration on Seismic Issues*	Five-year cooperative agreement with Japan. Highly leveraged and should be sustained
W6829	*Seismic Behavior of Spent Fuel Storage Cask Systems*	Why should NRC do this work rather than licensees?
Y6166	*Cooperative Geoscience Research*	Continuing work on seismic hazard of central and eastern U.S. Adequate understanding now available for regulatory processes. The need for this work is questionable.
Y6233	*Garner Valley Downhole Seismic Array Operation and Analysis of Data*	Ongoing data collection of limited potential impact on regulatory processes.
Y6481	*Senior Seismic Hazards Analysis Committee (SSHAC) Method*	Expert opinion elicitation for 10-year update of a semiquantitative analysis for 10 CFR Parts 72 [Ref. 19] and 100 [Ref. 20].
Y6718	*Soil-Structure Interactions for Buried Structures*	It is not apparent that NRC needs to conduct this research even in anticipation of advanced reactor technology. It is only necessary that NRC have available to it the resources necessary to adequately and independently review licensee submittals concerning this issue.
Y6796	*IAEA Coordinated Research Project on Seismic Ground Motion*	This is primarily an effort to acquaint other Nations through IAEA with NRC data and information on ground motion from nearby earthquakes of moderate intensity.
Y6797	*Evaluation of USGS 2002 Seismic Hazards Assessment*	Support for SSHAC method. Provides seismic hazard information for central, eastern and other regions of U.S.

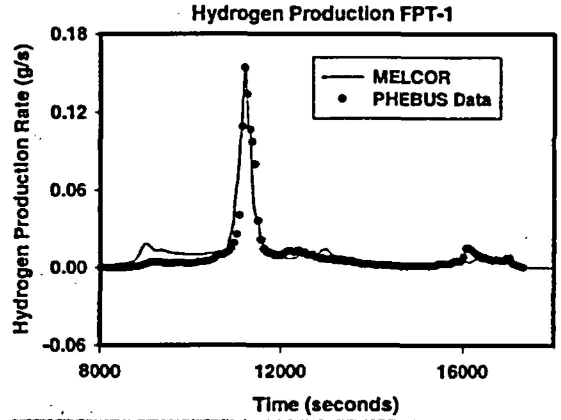
14 SEVERE ACCIDENT RESEARCH

In the past, the NRC invested very heavily in the experimental and analytical characterization of severe reactor accidents that dominate the risk to the public posed by the use of nuclear reactors to produce electrical energy. A very substantial technology has been established to understand the progression of 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 and to provide assurance of adequate protection, the NRC substantially curtailed its investments in severe reactor accident research. The NRC Severe Accident Research activities other than those directly associated with advanced reactors are listed in Tables 12 and 13.

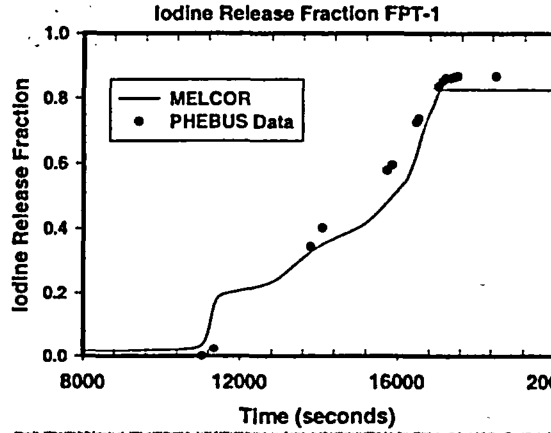
Research on Severe Accidents has been continuing in other Countries because they have more restrictive acceptance criteria than adequate protection. Substantial programs are under way in both Europe and Japan. The NRC has developed quite an effective strategy to maintain the technology for severe accident analyses and to update this technology with research results from international programs. The body of knowledge coming from the NRC's past work and ongoing work are systematized in 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. Current agreements include the following:

Realistic Reactor Accident Analyses

MELCOR Prediction of PHEBUS Data
Hydrogen Production FPT-1



MELCOR Prediction of PHEBUS Data
Iodine Release Fraction FPT-1



RES is succeeding in achieving greater accuracy and realism in its severe reactor accident analyses for plant risk assessments by validating MELCOR model predictions with data from several international cooperative experimental programs. Figures above show comparisons of MELCOR code predictions with data from the PHEBUS FPT-1 in-pile test of irradiated reactor fuel degradation.

- MASCA

  This is an experimental study under way in Russia on the behavior of reactor core debris in the lower plenum of a reactor pressure vessel. Results of these studies will be needed to evaluate safety claims related to the certification of both the AP1000 and ESBWR designs.

- ARTIST

  This is an experimental study under way in Switzerland to measure the aerosol removal on the secondary sides of steam generators during 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.

- PHEBUS-FP

  This Program consists of five large-scale, in-pile integrated tests of fuel degradation, fission product release, radionuclide transport through the reactor coolant system, and aerosol behavior in the containment. These tests have been designed to validate reactor accident models. Exceptionally prototypical results for code validation are being produced by this international program (see sidebar). Additional information is being provided by supporting separate-effects experimental programs, such as the French Program VERCORS, to investigate fission product release from MOX and high-burnup fuels under accident conditions.

- OECD-MCCI

  This is an experimental study of the viability of using an overlying layer of water on reactor core debris that has escaped the reactor coolant system and is interacting with structural concrete of reactor containments. This research will provide data for improved or new models of core debris coolability for systems-level accident analysis codes. Improved models will be used in the risk analysis of existing plants and the safety evaluation of advanced reactor designs. This Program is to be completed in FY2005.

These highly leveraged international programs are providing the bases for validating the NRC's accident analysis code, MELCOR. The MELCOR code has proven its utility to the regulatory process in recent years in connection with resolving issues such as the need for hydrogen igniters in ice condenser and Mark III containments and risk-informing 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors." MELCOR analyses are also important for the certification of the AP1000 and ESBWR advanced LWR designs. MELCOR analyses will likely be instrumental in the certification of other advanced reactor designs.

The ACRS is very supportive of the strategy the NRC has developed to maintain and update its capabilities for accident analyses. The ACRS encourages the NRC to give serious consideration to joining international cooperative research agreements now being proposed to examine:

- Prototypic, in-pile tests of radionuclide releases from, and degradation of, reactor fuel exposed to air for both reactor accident analyses and spent fuel storage pool accidents

- Prototypic, in-pile tests of radionuclide releases from, and degradation of, high-burnup fuels
- Prototypic, in-pile tests of radionuclide releases from, and degradation of, MOX fuel

The ACRS also supports programs to maintain the MACCS code and looks forward to reviewing results of the comparison of the Gaussian plume model to state-of-the-art dispersion models in the context of risk assessments.

The ACRS foresees the following challenges in the area of severe accident analysis:

- Assessment of in-vessel retention with respect to materials interactions
- Spent fuel pool accident progression assessment especially the effects of hydrides, effective ignition temperature, extent of pool involvement, and releases of fission products and actinides, as well as transport energetics
- Evaluation of severe accident uncertainties for PRA applications
- Sound technical basis for evaluating energetic fuel-coolant interactions
- Aerosol behavior in steam generator secondaries and other pathways that bypass reactor containments
- Effects of high burnup on core-melt progression and fission product release

Table 12. Severe Accident Research Activities

Project	Title	Comment
Y6312	*MASCA Program* International cooperative research on core debris interactions with the pressure vessel head	Supports certification of the AP1000 and ESBWR designs. The MASCA experimental efforts will be completed soon.
Y6313 Y6668	*OECD- Melt Coolability and Concrete Interactions Program* *Analytical Support for Y6313*	International cooperative research on core debris coolability during interactions with concrete.
Y6321	*Benchmark MOX Fuel Release Source Term Experiments*	Research to define fission product release from MOX fuel.
Y6377 Y6802	*MELCOR Code Development and Assessment* *MELCOR Severe Accident Code Development and Assessment*	Repository for research results generated in cooperative international programs. Code is being used extensively for revising regulations and certifying advanced LWRs.
Y6504 Y6607	*Steam Generator Fission Product Retention* *Support for ARTIST Tests of Fission Product Retention in Steam Generators*	Effort to add realism to predictions of releases during bypass accidents.
Y6571	*CSARP Meeting*	Useful annual meeting to review cooperative research results.
Y6512	Tests of Severe Accident Phenomena in Oxidizing Medium	Phenomenological tests and analyses for spent fuel pool accidents.
Y6664	*Containment Analysis and Experiments for AP1000*	Support for AP1000 certification.
Y6668	*Containment Analysis Support*	Containment analysis of advanced reactors.
Y6848	*High Burnup Fission Release Data*	Receipt and analysis of data from French VERCORS tests.
Y6696	AP 1000 Severe Accident Phenomena Evaluation	Severe accident phenomena assessment of AP1000.
Y6721	*Agreement with IBRAE-RAS on Nuclear Safety Analysis Codes*	Very cost-effective effort to modernize the MELCOR code.

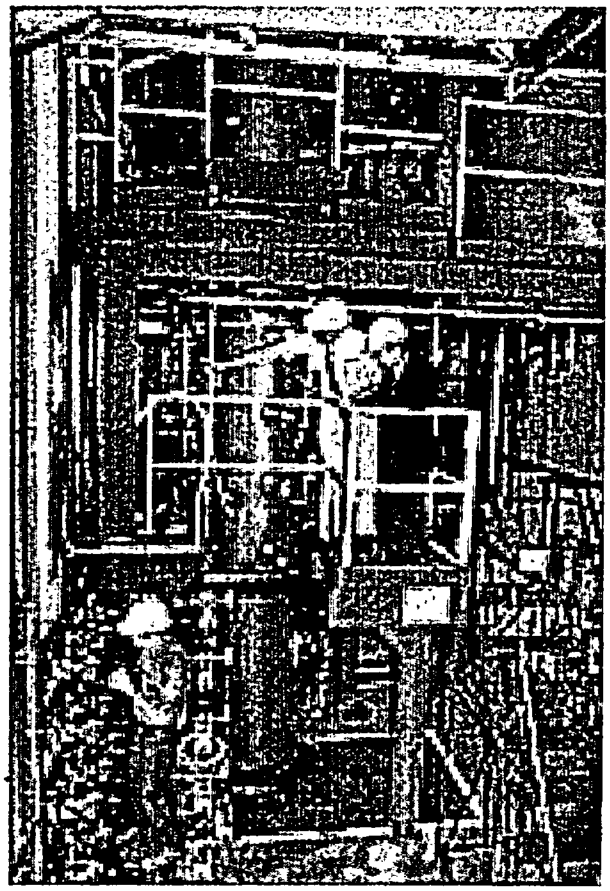
Table 13. Research Activities for Accident Consequence Models

Project	Title	Comment
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.
Y6628	*MACCS Uncertainty Assessment for Consequence Models*	Support for emergency planning.
Y6469	*Evaluation of Radionuclide Pathways and Uptakes*	Upgrade information on uptake pathways.

15 THERMAL-HYDRAULICS RESEARCH

The NRC regulatory process makes extensive use of thermal-hydraulic analyses for certifying advanced reactor designs, processing changes in the licensing bases of existing reactors, and investigating thermal-hydraulic safety issues. To meet the many needs for such thermal-hydraulic analyses, the NRC developed and maintained several thermal-hydraulic transient analysis codes. Development and assessment of these codes require separate-effects experiments to provide data for developing specialized models as new operating regimes and phenomena are encountered. Further, as new reactor concepts are proposed, it is necessary to obtain data for assessing the applicability and accuracy of agency codes for the different nuclear steam supply systems. The NRC has satisfied these needs through the agency staff and contractors, and by developing and maintaining specialized thermal-hydraulic test facilities at universities and national laboratories. In addition, the NRC has joined international efforts on the limited basis of exchanging data when opportunities have arisen.

The burden to maintain, modernize, and support several thermal-hydraulic transient analysis codes was significant. A prudent decision was made to consolidate the agency analysis capabilities into a single code now called TRACE. Consolidation was envisaged as an ambitious effort to provide capabilities in one code that could reproduce results of the other NRC codes, some of which had unique features. There was also the desire to modernize the consolidated code to some extent and certainly to rationalize the many correlations and approximations devised over



The PUMA Thermal-Hydraulics Test Facility at Purdue University

This Experimental Facility provides data for the development and validation of thermal-hydraulic models of passive safety systems in advanced light water reactor designs.

the years to obtain computational results consistent with data, especially in the complicated area of two-phase flow. Ultimately, the consolidated code can be used for independent evaluation of licensee's thermal- hydraulic analyses as well as for use in analysis of safety issues. To avoid the loss of legacy plant models that have been developed for the existing codes, RELAP5, TRAC-P, and TRAC-B, a Symbolic Nuclear Analysis Package (SNAP) is being developed which will be able to convert existing models, with little or no modification, to TRACE models. This graphical user interface is also intended to enhance productivity by providing a user-convenient environment for future input model development, graphical display of results, and input checking.

The consolidation of NRC's thermal-hydraulic modeling into a single code was a decision that was enthusiastically supported by the ACRS. The ACRS understood this to be a challenging undertaking. Indeed, the consolidation has proved to be every bit as challenging as anticipated. While consolidation is nearing completion, many "loose ends" remain to be resolved through the assessment and improvement phases of the project. The assessment and refinement efforts constitute an important fraction of the Thermal-Hydraulics Research Projects listed in Table 14.

The TRACE code has been developed to the point where it has provided useful validation calculations for the ESBWR and AP1000 designs, as well as for the steam generator hydrodynamic blowdown loads related to Generic Safety Issue (GSI) 188, "Steam Generator Tube Leaks / Ruptures Concurrent with Containment Bypass." However, it is not yet ready for routine regulatory application. Therefore, the NRC will need to maintain the RELAP5 code (Y6392). RELAP5 is being used to carry out intensive thermal-hydraulic analyses in support of the agency's initiative in reevaluating the PTS screening criterion and in supporting the certification of the AP1000 advanced LWR design. The maintenance of this older code should include correction of deficiencies in RELAP5 when necessary for satisfactory completion of these analysis efforts.

In addition to the work described in the list of sponsored projects, there have been significant contributions from the RES staff itself in the development and improvement of the TRACE code. While the system codes provide important thermal-hydraulic analysis capability, it is encouraging to see that computational fluid dynamic (CFD) methods are being used by the RES staff for resolving complex multidimensional thermal-hydraulic issues. Ultimately, the coupling of CFD methods with the system codes will add a new dimension of confidence to thermal-hydraulic analyses.

The development of new TRACE models and the refinement of these models to include the operating regimes of the advanced LWRs have created the need for new empirical data. To this end, the NRC is conducting separate-effects experimental research at universities. Electrically heated rod bundle reflood experiments are being conducted at Penn State University (Y6671). These tests use advanced instrumentation to permit greater insights into the reflood heat transfer process so that peak clad temperatures can be predicted with less uncertainty. Subcooled boiling experiments and associated model development are being conducted at UCLA (W6749) to develop more accurate models and reduce uncertainty associated with void generation expected in the ESBWR. Entrainment at "Tees" and in the upper plenum of the AP1000 is being investigated at Oregon State University (Y6507, Y6795) to reduce uncertainty in prediction of the AP1000 accident response. Experiments on low pressure choked two-phase flow typical of natural circulation operation and experimental and modeling research on interfacial area transport for improving the accuracy of two-phase models are being conducted at Purdue University (W6698, Y6769). The motivations for these efforts are the improvement of break flow models at low pressure and improvement of two-phase flow modeling accuracy by adding transport equations for the interfacial area. This is a generic effort to improve the accuracy of two-phase flow transient models.

To provide data needed for assessing the applicability and accuracy of TRACE simulations for the ESBWR and AP1000 designs, experiments are being conducted in integral system test facilities. The PUMA and APEX facilities are scaled models of the ESBWR and AP1000 designs. The NRC also participates in international programs such as the CAMP Program for exchange of code assessment and user experience and the SETH Program in Germany and the PANDA Program in Switzerland for exchange of experimental data. The SETH Program will contribute experience and data on boron mixing. The PANDA Program will provide natural circulation data for full-height facilities for code assessment and verification of the scaling methods used in the design of the PUMA facility.

The code consolidation effort has not gone as smoothly as was predicted at the time the work was initiated. The effort is most challenging and given the substantial additional workload for thermal-hydraulic research that has distracted from the effort, delay is not a surprise. We note that the code consolidation project involves many separate organizations and is funded from a variety of task orders having different NRC project managers. One NRC project has about 17 separate task orders that address a variety of thermal-hydraulic issues at separate organizations. Surely, the coordination of these efforts must be challenging.

Another observation is that no research is directed at development of probabilistic methods for use with TRACE for establishing margin of safety at a prescribed confidence level. There is a growing need for uncertainty estimates in thermal-hydraulic analyses used in the regulatory process.

The time is approaching when the TRACE effort must reach fruition and a consolidated code be integrated into the regulatory process. There may be ways to improve the effectiveness of this effort, such as:

- Creating a TRACE code peer-review group to provide greater focus on technical issues confronting the development and help achieve consensus on the approach.

- Combining the code consolidation efforts and thermal-hydraulic model development experiments under one NRC project manager to achieve more effective integration.

- Benefitting from an international effort leveraging NRC resources with similar undertakings within the CAMP community.

The NRC bears the substantial burden of developing and maintaining experimental facilities for thermal-hydraulic research. The burden has been lessened by the use of well-scaled facilities that are less than full height such as the PUMA and APEX facilities. Data from these facilities together with results of thermal-hydraulic simulations have proven the correspondence between data from the scaled facilities and the results that would be expected from the full-height systems of AP1000 and ESBWR. This conclusion is strengthened by the data from full-height international facilities such as PANDA and ROSA-IV. The ACRS believes that the PUMA and APEX facilities are of proven, current utility and will help in the certification of the ESBWR and AP1000 advanced reactor designs.

In the longer term, the NRC may want to expand international cooperative research agreements for maintaining thermal-hydraulic research facilities. The successes being had in such arrangements in fuel behavior research and severe accident research may serve as examples of the advantages of such an international approach.

Table 14. Thermal-Hydraulics Research Activities

Project	Title	Comments
W6698 Y6769	*PUMA Integral Test Facility* *PUMA Test Facility*	Experimental facility to be used for ESBWR design certification. Provides data on LOCA natural circulation instabilities and low-pressure choked flow data for assessment of the TRACE code
Y6673 Y6300 Y6525 Y6583 Y6830	*TRAC-M Development Assessment - Small LOCA Processes* *User Support for Consolidated TRAC Code* *TRAC-M Code Maintenance* *Advanced Reactor Development of TRAC-M Code - Advanced Reactor Portion of Y6300* *TRAC-M Development and Assessment - ROSA-IV Tests*	Consolidated NRC thermal-hydraulics code.
Y6666	*Advanced Numerical Methods in TRAC-M*	This work to add a droplet field and advanced numerical methods to TRACE should be accelerated. The work needs greater focus and could benefit from peer review
Y6507 Y6795	*APEX Experimental Validation* *APEX AP1000 Test Program*	Conduct tests for AP1000 review and do confirmatory analysis using TRACE. Conduct beyond-design-basis test for AP1000
Y6667	*SNAP Implementation*	Maintain and modify the graphical user interface for TRACE to provide ability to import TRAC-P and RELAP5 plant models for TRACE.

Table 14. Thermal-Hydraulics Research Activities (Continued)

Project	Title	Comments
W6749	***Thermal-Hydraulics Research*** Task orders to support TRACE consolidation: • Interfacial area transport • Modularization of TRACE • Development assessment numerical techniques • Phase separation in "Tees" • Subcooled boiling at low pressure • SNAP runtime and output visualization development • SNAP nodalization knowledge-based expert system • Improvements to PARCS • BWR model development and assessment • OECD/NRC benchmark for a BWR turbine trip transient • Multiphase CFD enhancements to nuclear reactor safety analysis • MELCOR assessment against SCDAP/RELAP5 • Automated code assessment program • Two-phase CFD enhancements for NPHASE code • Peach Bottom turbine trip analysis	This task is very diverse and might benefit from prioritization and additional technical direction perhaps from a peer-review group. Separate-effects tests within this Project should be closely coordinated with the TRACE code development.
K6987	***Analytical Support for Serena Program***	Program without any specific objectives; should be closed if not yielding products to crucial thermal-hydraulic activities.
Y6392	***Maintain, Apply, Assess, and Develop NRC Computer Codes***	Consolidate RELAP5 capabilities into TRACE. Maintain RELAP5. Analysis of advanced reactor designs. TRACE assessments of PWR steam and feedwater line breaks. This is a necessary activity until TRACE can be used to do the analyses.
Y6428	***Evaluation of Steam Generator Tube Rupture Performance***	This is a major effort using SCDAP/RELAP to analyze steam generator tube rupture accidents.
Y6503 Y6662	***AP1000 analysis*** ***AP1000 Confirmatory Thermal-Hydraulics Analysis***	TRACE analyses of AP1000 large and small break loss of coolant accidents.

Table 14. Thermal-Hydraulics Research Activities (Continued)

Y6772	***AP1000 Confirmatory Research***	Measure entrainment for AP1000 certification Needs better coordination with TRACE development and assessment.
Y6526	***Administer CAMP Meeting***	Support for users of RELAP5; This is a valuable international collaboration.
Y6571	***SETH Program - Test Facilities***	Boron mixing programs in the PKL facility in Germany and the PANDA facility in Switzerland.
Y6589	***Thermal-Hydraulics Research*** Five task orders to support various thermal-hydraulics research: • two-phase CFD enhancement • MOX neutronics • Interfacial area transport • separate effects experiments for model development • MELCOR assessment and application	Applicable parts of this work need to be coordinated and combined with the TRACE code development. The longer-term tasks need peer review to establish relevance.
Y6598	***RELAP5 Analyses for Pressurized Thermal Shock***	Complete PTS analyses for four plants.
Y6671	***Rod Bundle Reflood Experiments - Phase 2***	Rod bundle heat transfer tests in support of TRACE reflood model development. Needs to be more closely coordinated with the TRACE code development.
Y6804 **Y6806**	***ESBWR Containment Support*** ***Review of GE Scaling Report***	Thermal-hydraulics support for ESBWR design certification.

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11. ABSTRACT *(200 words or less)*

This report 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). This report focuses on that portion of the NRC research program dealing with the safety of existing nuclear reactors and advanced light water reactor designs, AP1000, and ESBWR, submitted for certification. In its review of the NRC research activities, the ACRS considered the programmatic justification for the research as well as the technical approach and progress of the work. This review attempts to identify research crucial to the NRC mission. It also attempts to identify 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. The review also attempts to identify areas where greater international cooperation in research useful to the NRC could leverage resources of partners in the research and yield superior technical products. This report does not address research on the vulnerability of existing nuclear power plants to acts of sabotage and terrorism.

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