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Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program

A Report to the U.S. Nuclear Regulatory Commission

Advisory Committee on Reactor Safeguards

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Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program

A Report to the U.S. Nuclear Regulatory Commission

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Advisory Committee on Reactor Safeguards

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

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ABBREVIATIONS

ABWR	Advanced Boiling Water Reactor
ACRS	Advisory Committee on Reactor Safeguards
ACWI	Advisory Committee on Water Information
AECL	Atomic Energy of Canada Limited
AFCI	Advanced Fuel Cycle Initiative
AIChE	American Institute of Chemical Engineers
ALARA	As Low as Reasonably Achievable
AMPs	Aging Management Programs
ANL	Argonne National Laboratory
ANS	American Nuclear Society
AOOs	Anticipated Operational Occurrences
APWR	Advanced Pressurized Water Reactor
ASME	American Society of Mechanical Engineers
ASP	Accident Sequence Precursor
ASTM	American Society for Testing and Materials
ATHEANA	A Technique for Human Event Analysis
ATWS	Anticipated Transient without Scram
BIP	Behavior of Iodine Project
BWR	Boiling Water Reactor
CAROLFIRE	Cable Response to Live Fire
CCI	Core Concrete Interaction
CDF	Core Damage Frequency
CFD	Computational Fluid Dynamics
CFR	*Code of Federal Regulations*
CHRISTI-FIRE	Cable Heat Release, Ignition, and Spread in Tray Installations during FIRE
COL	Combined License
CRDM	Control Rod Drive Mechanism
CSARP	Cooperative Severe Accident Research Program
DI&C	Digital Instrumentation and Control
DOE	Department of Energy
EAC	Environmentally Assisted Cracking
EPAct	Energy Policy Act
EPIX	Equipment Performance and Information Exchange System
EPR	Evolutionary Power Reactor
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
FDP	Fitness for Duty Programs
FSME	Office of Federal and State Materials and Environmental Management Programs
GALL	Generic Aging Lessons Learned
GDC	General Design Criteria
GEH	General Electric-Hitachi
GNEP	Global Nuclear Energy Partnership
GSI	Generic Safety Issue
GWd/t	Giga Watt day per metric ton
HAMMLAB	Halden Man-Machine Laboratory
HDPE	High Density Polyethylene

ABBREVIATIONS (Cont'd)

HERA	Human Event Repository and Analyses
HF	Human Factor
HFE	Human Factors Engineering
HRA	Human Reliability Analysis
HRP	Halden Reactor Project
HSI	Human System Interface
HTGR	High Temperature Gas-cooled Reactor
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
IGSCC	Intergranular Stress Corrosion Cracking
INPO	Institute of Nuclear Power Operations
ISG	Interim Staff Guidance
ISI	In-Service Inspection
JAEA	Japan Atomic Energy Agency
JNES	Japan Nuclear Energy Safety Organization
LANL	Los Alamos National Laboratory
LBB	Leak before Break
LBLOCA	Large-Break Loss-of-Coolant Accident
LER	Licensee Event Report
LERF	Large Early Release Frequency
LSTF	Large Scale Test Facility
LOCA	Loss-of-Coolant Accident
LWR	Light Water Reactor
MACCS	MELCOR Accident Consequence Code System
MASLWR	Multi-Application Small Light Water Reactor
MCCI	Molten Core Concrete Interaction
MCNP	Monte Carlo N-Particle Transport Code System
MDM	Materials Degradation Matrix
MeV	Million Electron Volts
MOU	Memorandum of Understanding
MOX	Mixed Oxide
NAS	National Academy of Sciences
NASA	National Aeronautics and Space Administration
NCRP	National Council on Radiation Protection and Measurements
NDE	Non-Destructive Examination
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NGNP	Next Generation Nuclear Plant
NIST	National Institute of Standards and Technology
NMSS	Office of Nuclear Material Safety and Safeguards
NOAA	National Oceanic and Atmospheric Administration
NPPs	Nuclear Power Plants
NRC	Nuclear Regulatory Commission
NRO	Office of New Reactors

ABBREVIATIONS (Cont'd)

NRR	Office of Nuclear Reactor Regulation
NSIR	Office of Nuclear Security and Incident Response
NUPEC	Nuclear Power Engineering Corporation
OECD	Organization for Economic Cooperation and Development
OSU	Oregon State University
PARCS	Purdue Advanced Reactor Core Simulator
PCI	Pellet Cladding Interaction
PCT	Peak Clad Temperature
PINC	Program for the Inspection of Nickel Alloy Components
PIRT	Phenomena Identification and Ranking Table
PMMD	Proactive Management of Materials Degradation
PNNL	Pacific Northwest National Laboratory
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSHA	Probabilistic Seismic Hazard Analysis
PTS	Pressurized Thermal Shock
PSI	Paul Scherrer Institute
PUMA	Purdue University Multidimensional Integral Test Assembly
PUREX	Plutonium Uranium Extraction
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
R&D	Research and Development
RBHT	Rod Bundle Heat Transfer
RCS	Reactor Coolant System
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
ROP	Reactor Oversight Process
RPV	Reactor Pressure Vessel
RTF	Radioiodine Test Facility
SCALE	Standardized Computer Analysis for Licensing Evaluation
SDP	Significance Determination Process
SERENA	Steam Explosion Resolution for Nuclear Applications
SGTR	Steam Generator Tube Rupture
SNL	Sandia National Laboratories
SOARCA	State-Of-the-Art Reactor Consequence Analyses
SPAR	Standardized Plant Analysis Risk Model
SRM	Staff Requirements Memorandum
SRP	Standard Review Plan
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
SSHAC	Senior Seismic Hazard Analysis Committee
SSWICS	Small-scale Water Ingression and Crust Strength
THERP	Technique for Human Error Rate Prediction
THIEF	THermally Induced Electrical Failure
TRACE	TRAC-RELAP Advanced Computational Engine
TSUNAMI	Tools for Sensitivity and UNcertainty Analysis Methodology Implementation
UREX	Uranium Extraction
U.S.	United States

ABBREVIATIONS (Cont'd)

USGS	United States Geological Survey
V&V	Verification & Validation
VHTR	Very-High-Temperature Reactor
WRS	Weld Residual Stress
xLPR	Extremely Low Probability of Rupture

I. INTRODUCTION

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

- Ensure its regulations and regulatory processes have sound technical bases and these bases are refined as new knowledge develops.

- Prepare for anticipated changes in the nuclear industry that could have safety implications.

- Develop improved methods to carry out its regulatory responsibilities.

- Maintain an infrastructure of expertise, facilities, analytical capabilities, and data to support regulatory decisions.

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

- Advanced Non-LWR Designs

- Digital Instrumentation and Control Systems

- Fire Safety

- Reactor Fuel

- Human Factors and Human Reliability

- Materials and Metallurgy

- Neutronics and Criticality Safety

- Operational Experience

- Probabilistic Risk Assessment

- Radiation Protection

- Nuclear Materials and Waste

- Seismic and Structural Engineering

- Severe Accidents and Source Term

- Thermal Hydraulics

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

Chapter 2 of this report provides a synoptic account of research activities in each of the technical disciplines and highlights some of the accomplishments of the work. Additional details on each of the research areas are included in Chapters 4 through 17.

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

- Identifies and resolves current safety and regulatory issues.

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

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

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

- Improves the efficiency and effectiveness of the regulatory process.

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

The research to support reactor license renewal beyond 60 years is discussed in Chapter 3. In that Chapter, the ACRS notes that much of the current research for “life beyond 60” is focused on known degradation and analysis methods to assure processes currently in place to manage aging beyond 40 years remain valid beyond 60 years. ACRS identifies several additional research areas the nuclear industry needs to pursue. ACRS notes that NRC will have to follow those industry-led research efforts and integrate confirmatory research with these efforts.

2. GENERAL OBSERVATIONS AND RECOMMENDATIONS

General Observations Concerning the Organization of the NRC Safety Research Program

Activities within the NRC Safety Research Program are very closely tied to needs of the line organizations (NRR, NRO, NMSS, FSME, and NSIR) as identified through “User Need” documents. This strategy for the identification of research undertakings together with the associated process for the prioritization of research needs is working well. Research activities are yielding products that are useful to the line organizations and they are doing so on a timely basis.

A risk that could arise within such a strategy for the organization of research is that long-term, more speculative research would not be initiated. The Office of Nuclear Regulatory Research does mitigate this risk. Limited duration studies of the feasibility and potential of more speculative research are funded. Pursuit of the research, once feasibility has been established, does require the endorsement and support by the affected line organization.

A consistent feature of the current research program is enthusiastic support of the activities by the affected line organization. Line organizations appear to have close, continuing involvement in the conduct and the direction of the research. It has become common for presentations of research results to the ACRS to include statements of support from line organizations and even presentations on the uses to be made by line organizations of research results.

Collaborations in the Conduct of Research

Major research activities at NRC often include collaboration with other Federal agencies, industrial institutions, and international partners. NRC is taking good advantage of opportunities to leverage its resources and expertise on issues of common interest to others. The collaboration is providing high quality peer input to the organization and conduct of research.

In the past, in-depth collaborations with international partners were characteristic of research in only a few technical areas. Such collaborations have become much more common. ACRS endorses such collaborations on both the conduct of research and the analysis of results. International collaborations will become ever more important as the nuclear industry becomes more international in nature. Collaborations in research will help in keeping safety regulation of nuclear power cohesive throughout the developed world.

Research Planning

There is much more emphasis on the detailed, documented planning of research now than in the past. In many areas, research plans are being subjected to extensive peer review by the broader nuclear community both by design and as a consequence of extended collaborations with other elements of this larger community. This detailed planning of research has greatly helped in the conduct and quality of the research.

Documentation of Research Results

RES is subjecting its research results to more peer review. Documentation of the final results is variable. In some cases, documentation is outstanding and the results are readily retrieved by interested parties outside the agency. In other cases, documentation is limited to reports readily identified and retrieved only within NRC.

Areas Deserving Attention in the Future

There is a growing emphasis on the use of numerical simulation to resolve issues of reactor safety. Some particular aspects of this trend are noted in our comments below on individual technical areas of research. The Department of Energy (DOE) is undertaking a major computational initiative in support of the existing nuclear reactors. DOE seeks to establish what it calls high fidelity simulation of the thermal hydraulics, neutronics, and mechanics of nuclear power plants during normal operations and anticipated operational occurrences (AOOs). The effort is centered at the DOE national laboratories but both licensees and nuclear steam suppliers are participating in this initiative. It is, then, likely that products of the initiative may well be utilized in submissions made to the NRC for regulatory purposes. NRC needs to be in a position to evaluate products of massively parallel computations.

The growth in the use of computations to resolve reactor safety issues has not been accompanied by a similar growth in the development of experimental facilities capable of yielding data suitable for the validation of computational predictions. Indeed, experimental facilities within the U.S. available to the NRC for obtaining validation data continue to dwindle. There is some confidence that adequate experimental facilities remain in the rest of the world and that these facilities are available to NRC through formal or informal collaborations. NRC is taking advantage of these foreign capabilities in areas of research such as severe accidents, materials, and seismic research. The process is not, however, seamless. Greater availability of test reactors could speed the cost-effective resolution of safety issues. Lack of hot cells within this country and the challenges of transportation do handicap the post test examination of irradiated specimens.

NRC is moving toward greater use of probabilistic assessments of risk in its regulatory processes. NRC relies on methods of probabilistic risk assessment developed decades in the past. There is an interest within the larger risk analysis community of developing modernized tools with superior capabilities for risk analysis. This modernization is in addition to the expansion of scope of risk analysis to include greater breadth by including seismic risk, fire risk, and risk during shutdown operations as well as greater depth to Levels II and III. Of particular interest is the development in modernized risk assessment tools of superior risk metrics that can be used to focus and guide NRC functions of monitoring and inspection. NRC needs to be involved in research to advance the development of tools for the conduct of risk analyses.

Major Observations on Individual Areas of Research

Major observations, conclusions, and recommendations concerning specific research activities are summarized below. Additional details on the research activities in the various technical disciplines are provided in Chapters 4 through 17.

Advanced Non-LWR Designs

The first research priority in the area of advanced non-LWR designs should be to support the advanced thermal-spectrum, high-temperature, gas-cooled reactor (HTGR) consistent with the 2005 Energy Policy Act (EPAct 2005). Plans now being developed by the NRC staff for this research seem to be well founded. A low priority should be given to monitor the DOE activities with respect to the Advanced Fuel Cycle Initiative (AFCI).

Digital Instrumentation and Control Systems

The staff has issued a new plan for its research on safety issues associated with the transition to software-based digital instrumentation and control for safety systems. The Steering Panel formed for this work should assist in keeping research focused on products that will be of use to the line organizations.

Fire Safety

The NRC's fire safety research program is at the forefront of a rapidly evolving understanding of fire events and the assessment of risks from fire damage in nuclear power plants. Structured collaboration with industry has provided cost-effective solutions and technical insights that surpass independent efforts. The NRC is a recognized leader in national and international fire safety research.

New reactor designs rely heavily on integrated digital instrumentation, protection, and control systems. Extensions of operating licenses and obsolescence of analog equipment will also likely lead to replacement of many currently installed instrumentation and control (I&C) systems with digital platforms. Limited information is available regarding the effects from fire, heat, and smoke on digital equipment. Research should address such issues as the effects of fire and heat on fiber optic cables, the effects of heat on digital equipment, and smoke damage to digital signal processing and computation modules.

Reactor Fuel

There is ample capability to assess risks of failure of current fuels due to centerline melting or departure from nucleate boiling events. The NRC does not have empirical or analytical capability to quantitatively assess the risks of Pellet Cladding Interaction (PCI) fuel failures during AOOs on current or future fuel designs.

Advanced fuel designs are in development and it is imperative that NRC maintain a core capability to analyze the safety of new fuel designs. Of particular concern is the source of experimental data on advanced fuel and cladding behavior under conditions of design basis and beyond design basis accidents.

Human Factors and Human Reliability

The ACRS finds the NRC's Human Factors (HF)/Human Reliability Analysis (HRA) research program and activities to be based on sound rationale and to be focused on both near- and long-term needs of the agency. The research program is yielding products of use to the regulatory process.

Materials and Metallurgy

The ACRS enthusiastically supported the initial vision of the Proactive Materials Degradation Assessment Program and has recommended its continuation. However, the program seems to have lost its momentum and focus. Current research appears to concentrate on conventional reactive management instead of anticipating and preventing materials degradation surprises. The work, published in Expert Panel Report on Proactive Materials Degradation Assessment (NUREG/CR-6923), identified target items of low knowledge and high-to-medium susceptibility to degradation. These target items were essentially predictions of possible materials degradation problems which had yet to occur. They represented areas of greatest uncertainty and concern to the experts involved in the identification process. The next step in the proactive process should be the validation of one or more of these predictions by experimental means or by focused inspections at operating plants. In the absence of a quantitative confirmation of the predictions made in NUREG/CR-6923, it is doubtful that the program can attain its primary goal. If such confirmations are not possible, continuation of the program should be reconsidered.

The effectiveness of the various materials, fabrication, and water chemistry changes introduced to mitigate environmentally assisted cracking in Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs) should be confirmed by long-term and aggressive confirmatory testing. Some mitigation methods may lose effectiveness over time, while others (individually or in combination) may demonstrate the ability to protect critical components from stress corrosion cracking for the 60 year life of the uprated plants.

With the closure of the Steam Generator Action Plan, the vast amount of knowledge gained during this long term research program should be thoroughly documented.

Neutronics and Criticality Safety

NRC needs to establish a long term strategy to upgrade its neutronics analysis capabilities to take advantage of modern computer architecture in a way that facilitates the analysis of low symmetry reactor core designs including quantitative analysis of parametric uncertainty.

A move to reactors based on technologies other than light water is not imminent. There are, however, DOE initiatives in both gas-cooled reactors and sodium-cooled reactors that will challenge NRC neutronics analyses capabilities. Some of the challenges are associated with data bases. NRC can encourage the DOE and foreign counterparts to cooperate in the development of data bases adequate for safety analyses of advanced reactor concepts and reactors using higher fuel enrichments and new fuel compositions such as mixed oxide (MOX) fuels enriched in other actinides.

Operational Experience

The Operational Experience Research program is being managed and executed in a manner consistent with the needs of the NRC. It provides data and tools necessary for regulatory decision making and for the assessment of regulatory effectiveness.

Probabilistic Risk Assessment

It has been nearly twenty years since NRC sponsored the NUREG-1150 Study, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." It is time to begin planning for a new NRC Level 3 probabilistic risk assessment (PRA) to demonstrate advances in PRA methods and suitable approaches for new reactors.

There is a need for the NRC to develop a common approach for the elicitation of expert opinions in areas of regulatory interest where knowledge is incomplete. Common approaches for the treatment of model and parametric uncertainty are also needed.

PRAs contain a wealth of information regarding the ways undesirable plant states, such as core damage and large release of radioactivity, can occur. This information is not utilized in formulating security requirements to evaluate their benefit and their impact on safety. The ACRS recommends that RES establish a research project to explore the possibility of risk-informing security requirements and building on PRAs to create a unified framework for the evaluation of both safety and security.

Radiation Protection

The staff has developed an appropriate and robust research program in the areas of radiation protection. This program includes radiation protection of workers and radiological assessments related to decommissioning and waste management.

Nuclear Materials and Waste

The NRC has been successful in leveraging its nuclear materials and waste research efforts with those of other federal agencies and other organizations. These collaborations are productive and provide significant synergy with NRC research activities.

Seismic and Structural Engineering

Efforts by the NRC staff to revitalize the seismic safety research in support of regulatory activities have been exemplary. There is a well developed research plan that has been broadly reviewed for both technical quality and programmatic impact. Credentials and expertise of those pursuing the research are quite impressive. The research is having impact on the regulatory processes and is expected to have even greater impact in the future. The active research involving collaborations with the larger seismic community is assuring the agency has at hand expertise to help resolve novel issues that arise as new plant designs are submitted for certification and new license applications are submitted for staff review.

Severe Accidents and Source Term

The ACRS supports the overall strategy that the NRC staff has developed to support regulatory decisions for severe accidents through computer code and associated model development such as the MELCOR code and targeted experimental data analysis and evaluation. This approach can successfully maintain and update NRC modeling capabilities for severe accident safety analyses for light water reactors and can set the stage for advanced reactor accident analysis. The NRC approach to leverage resources by international experimental collaborations is notable and should be continued. The planned program extensions and experimental collaborations are well worth the investment.

Thermal Hydraulics

The RES staff is to be commended for the progress that has been made in developing and moving forward with incorporation of TRACE into the regulatory process. Much work remains to be done to enable its reliable use for the analysis of the new LWR designs, an urgent matter which should be conducted with high priority. The priorities for further development of TRACE require careful evaluation.

The international collaborative efforts are also to be commended, as they take advantage of facilities that are of a scale and capability that do not currently exist in the U.S. Furthermore, they draw on the expertise of international partners, who have continued to maintain a very high level of capability in the thermal-hydraulics field. However, complementary development of national facilities to address safety related thermal hydraulics issues should be seriously considered. Such facilities would enable the retention of U. S. expertise and provide the capability to conduct experiments for supporting confirmatory thermal-hydraulic analyses of new reactor designs.

NRC currently has only a modest effort in the area of computational fluid dynamics (CFD) and it is limited to some use of commercial CFD codes. Although commercial CFD codes are used in the process industry for qualitative indications of phenomena, they are validated to a much less rigorous standard than codes for nuclear use, and the source codes are not available. They appear to include a number of *ad hoc* fixes to improve stability and robustness which may affect their predictive capability for situations that cannot be studied experimentally.

It is inevitable that the licensees will increasingly capitalize on the extraordinary advances in computing power and computational science to resolve problems which the current generations of thermal-hydraulic codes such as TRACE are unable to do. A recent LWR Sustainability Research and Development (R&D) Program, prepared by DOE in close collaboration with industry R&D programs, considers features such as a next-generation system analysis code as essential components to improve understanding and utilization of safety margins. NRC thermal-hydraulic research has to position the agency to address the research coming from the LWR Sustainability Program.

3. LIFE BEYOND 60

The NRC position on license renewal is based on:

(1) a policy decision that if plants are safe enough to operate currently, maintaining that licensing basis is sufficient for future operation and

(2) a technical position that current regulations (especially 10CFR 50.65, the maintenance rule) and operating procedures are sufficient to manage degradation in active systems and components and that license renewal should focus on the aging management of passive, long-lived, safety-related components.

There are no indications at present that the NRC licensing position will or should change. From this perspective, the critical research required for license renewal beyond 60 years relates to aging management of passive components and structures, namely the reactor coolant system, the reactor internals, concrete, cables, and support system components such as buried piping.

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," those plants that enter into a period of extended operation are required to have in place aging management programs (AMPs) that mitigate aging degradation effects during the period of continued operations. As a result of the engineering margins incorporated into the design of US nuclear reactors and AMPs, the plants' structures, systems, and components (SSCs) inherently afford a measure of protection against the deleterious effects of aging. For those plants that may elect to pursue subsequent license renewal periods beyond the 60 year operating period, the staff is investigating whether additional research into aging effects is needed.

The potential for new modes of degradations and gaps in our understanding of currently recognized modes of degradation has been recognized. In 2004, the Electric Power Research Institute (EPRI), through an expert opinion elicitation process, developed a Materials Degradation Matrix (MDM), which identified these potential mechanisms and provided the status of current research progress and needs. In addition, the NRC has developed the Proactive Management of Materials Degradation (PMMD) program to address the regulatory issues involved with previously unidentified materials aging processes, and the Generic Aging Lessons Learned (GALL) report covering aging issues associated with Part 54 reviews. However, the scope of the assessments performed for the PMMD program were more limited in scope than is required for an assessment in terms of 10 CFR Part 54 and did not include for example degradation of cables, concrete, and buried piping.

In order to obtain a broader range of inputs, the NRC and DOE jointly sponsored a public workshop on February 19–21, 2008, in Bethesda, MD, to discuss potential research and development issues related to ensuring that, if licensees elect to pursue subsequent license renewal periods, continued long-term operation can be conducted safely. Panel and public discussions were held on: systems structures and components, materials degradation issues, diagnostics and prognostic technologies, and the future needs (beyond the scope of 10 CFR Part 54) of the nuclear industry to continue long-term operation. Participants included representatives from the NRC, DOE, industry, national laboratories, academia, the International Atomic Energy Agency, the public, and international organizations.

The workshop participants recognized radiation embrittlement of reactor pressure vessels as a fundamental issue for life extension beyond 60 years. Based on our current understanding of embrittlement mechanisms, recent studies by the NRC on pressurized thermal shock (PTS) indicate that margins are considerably larger than expected and most vessels, being operated as they currently are, should probably be able to operate past 60 years. It also appears that there is sufficient margin on upper shelf toughness to justify extended operation.

To assess the need for additional work it is important not only to consider the level of our scientific understanding of the nature of possible aging mechanisms, but also the adequacy of current engineering approaches to the management of these aging issues. Thus, the critical issue for reactor vessels is whether our current correlations for embrittlement can be extrapolated to 80 years of operation. These correlations are largely empirical and caution must be used in extrapolating them to conditions outside the database on which they are based. For the current renewal period, the reactor pressure vessel materials surveillance program provides a means to detect unexpected degradation. The surveillance program was not planned to extend for the duration of operations now being discussed, and a research program to determine whether there are new mechanisms of embrittlement that could become significant and thus invalidate our current understanding is warranted. The potential need for modifications to the surveillance programs to address extended life was recognized in the ACRS report on PTS, and there appear to be industry efforts under way to address this need. NRC staff needs to be aware of these efforts.

The limited availability of surveillance materials suggests that a more fundamental understanding of radiation damage will be needed to assure integrity of reactor vessels at the higher fluences associated with lives beyond 60 years.

Issues with core internals such as irradiation assisted stress corrosion cracking are not fully resolved even for the case of license extension to 60 years. Licensees have typically committed to participate in the industry programs for investigating and managing aging effects on reactor internals; to evaluate and implement the results of the industry programs; and to submit an inspection plan reactor internals based on these efforts to the NRC for review and approval. Improved materials choices are available that have been demonstrated to at least substantially mitigate many of the materials problems that have been experienced in current LWRs, but the degree of improvement of performance that can be achieved for core internal materials is unclear.

New reactors may be able to design around the problem, e.g., by ensuring that all internals are replaceable. For existing reactors demonstration of acceptable performance for life beyond 60 years may require both improved understanding and improved inspection techniques.

In addition to the research associated with the passive, long-lived, safety-related components associated with 10 CFR Part 54, life beyond 60 efforts will seek to make extended operation economically and operationally justifiable. For example, plants seeking extended operation will very likely need to convert to digital instrumentation and control systems. While the NRC needs to address the safety issues associated with such changes, the research needed to develop such systems is clearly the responsibility of industry and DOE.

Similarly in addressing issues such as the cracking of nickel alloys used in the reactor coolant system and steam generators, NRC effort will focus on the potential for cracking in the materials in use such as Alloy 600TT and Alloy 690. Industry and DOE efforts could consider the development of less expensive and easier to weld alloys.

It was recognized at the workshop that cable aging and buried piping can be issues both in terms of 10 CFR Part 54 and economical operation. Only a small fraction of cable and buried piping are typically involved in evaluations related to 10 CFR Part 54. However, managing the aging of much larger fractions of cable and buried piping will be important to economic operation of plants during an extended period of operation. NRC programs directed towards extended operation will need to consider confirmatory research on degradation mechanisms that can affect safety, but recognize that industry has the primary responsibility to address aging problems, especially those that primarily affect the ability to operate economically.

The staff has had focused discussions with DOE, the domestic industry (e.g., the Nuclear Energy Institute and the Electric Power Research Institute), and potential international partners to begin development of an integrated research plan in order to better leverage resources and prevent unnecessary duplication of efforts. In addition, public outreach continues to ensure appropriate stakeholder participation. In the longer term (fiscal year 2010 and beyond), the NRC, in collaboration with DOE, the industry, and international partners, will begin work on the priority areas identified in the integrated "Life Beyond 60" research plan to ensure that adequate technical information is available for regulatory decisions if licensees pursue subsequent license renewal terms.

The Office of Nuclear Regulatory Research needs to determine how its research in this area will be integrated with that being done by DOE and the industry.

The work to date in this area has focused on the additional characterization of recognized degradation processes and what is necessary to assure that current aging mitigation strategies remain valid for lifetimes greater than 60 years. These current efforts should be augmented by further efforts on the Pro-active Materials Degradation Assessment Program to identify degradation processes not now manifest or anticipated.

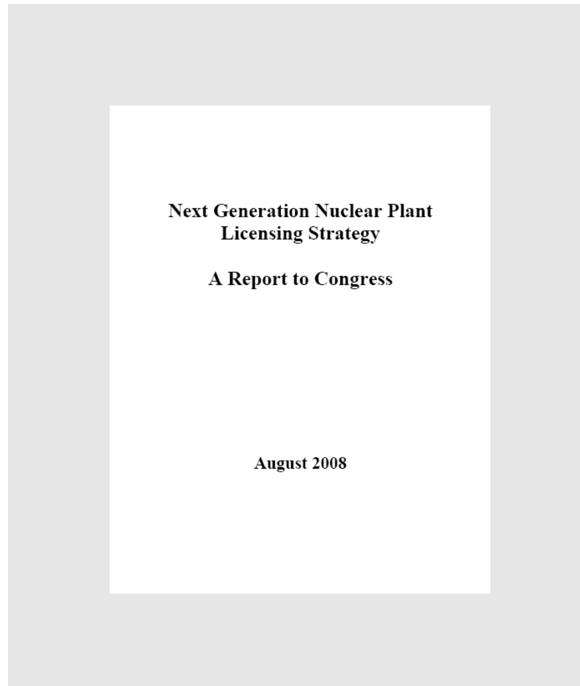
4. ADVANCED NON-LWR DESIGNS

Background

The DOE Generation IV program for advanced reactors was first developed in 2000-2002 with the publication of the Generation IV Technology Roadmap (DOE, 2002). From 2002 to 2005, the primary goal of the Generation IV program in the US was to focus on the use of high-temperature (850ºC to 1000ºC) process heat and innovative approaches to yield energy products, such as hydrogen, that might benefit the transportation and the chemical industries. To that end, the DOE began to interact with reactor designers, potential process heat users, and industrial and international organizations to support the Next Generation Nuclear Plant (NGNP) design and development needs for the thermal-spectrum, gas-cooled, graphite-moderated reactor concept; the so-called Very High Temperature Reactor (VHTR). The DOE effort was reinforced by the passage of the Energy Policy Act of 2005 (EPAct 2005),[1] which authorized appropriation of funds for research and construction activities for the NGNP project. DOE selected the VHTR as the lead design concept for the NGNP. Specifically, the VHTR program is focused on this type of reactor design and the EPAct 2005 authorizes the NRC to collaborate with the DOE in safety research related to licensing issues as the project proceeds through licensing to construction and operation.

As part of this joint effort on safety research for NGNP licensing, the NRC has completed initial activities to provide the background information for the VHTR reactor. The activities included:

- development of information sources for VHTR technology; and



Next Generation Nuclear Plant
Licensing Strategy

A Report to Congress

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- conduct of a Phenomena-Identification and Ranking Table (PIRT) process for the VHTR in collaboration with the DOE and its contractors to identify the safety phenomena that require additional research and development. These important phenomena are associated with tools, standards, data, etc. required for the design and licensing review of such reactor technology.

Current Research Activities

The staff is focused on the development of appropriate evaluation models/methods/ guidance using information from past prismatic, gas-cooled, reactor designs and pebble-bed designs as well as the current conceptual VHTR designs from the industrial teams working with the DOE and its national laboratory contractors.

[1] See Subtitle C: Next Generation Nuclear Plant Project.

The staff has developed a plan for advanced reactor research. The plan is well-written and comprehensive. Because the final design for the VHTR has yet to be determined, the plan is focused on research applicable to the range of possible designs.

In fiscal year 2006, DOE shifted the emphasis of the Advanced Fuel Cycle Initiative program (AFCI) and the Generation IV fast-reactor development program. At that time, spent fuel management efforts began to focus on using a closed fuel cycle with reprocessing and recycling to a fast reactor as a main objective of the DOE spent fuel management and nuclear energy programs. The Global Nuclear Energy Partnership (GNEP) program was announced in early 2006. This new priority was short-lived and since the start of fiscal year 2009 the DOE has established the AFCI research and development program to address fuel cycle challenges, such as advanced fuel development.

Assessment and Recommendations

Despite rather extensive changes made by DOE in its programs and plans for development of advanced reactors, the NRC staff has been able to develop a focused plan on research needed to support the licensing of a very high temperature gas-cooled reactor. This plan addresses:

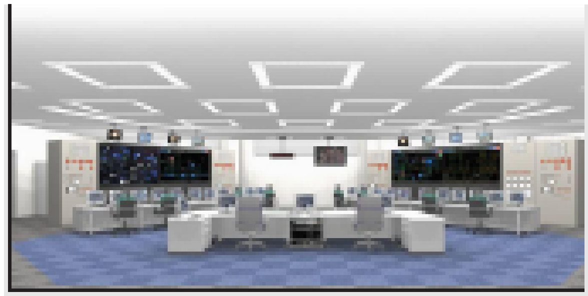
- the reactor fuel,
- nuclear graphite and structural materials performance,
- human factors and instrumentation,
- Thermal-fluid analysis and containment analysis,
- structural and seismic analysis, and
- accident and source term analysis

5. DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

The challenges that arise in the transition from the use of analog to digital, software-based, technology for instrumentation and control systems in nuclear power plants are widely appreciated. These challenges are being met as software-based digital technology is applied in upgrading nuclear plant control systems. There have been, however, no applications of such digital systems to reactor safety systems which are the systems of primary interest to the NRC. In part, this has been due to the lack of standards and guidance from the regulators on the critical issues involved in the application of digital technology to reactor safety systems. Among the challenges that have been identified are:

- testing methods for software based systems,
- software quality and associated development and test processes,
- the representation of software based systems in PRAs, and
- the maintenance of safety division independence in a software-based communications environment.

Safety systems involve simple actuations and do not involve complicated feedback and control functions. But, software-based digital systems offer the designer much greater flexibility and functionality than analog systems. Taking advantage of the enhanced flexibility and functionality carries with it increased circuit complexity, a new mode of common cause failure in software programming, and the potential for loss of safety division independence through inter-division data communication.



Candidate Generation III Nuclear Power Plant Control Room

Recognizing the difficulties faced by industry and the NRC, the staff, in 2007, established a new research plan. The staff also established a Steering Committee to identify technical and regulatory areas involving digital instrumentation and control requiring guidance and developed and oversaw a plan of action in concert with industry. Seven key issues were identified:

- Cyber security
- Diversity and Defense In Depth
- Risk-informing Digital Instrumentation and Control
- Highly Integrated Control Room - Communications
- Highly Integrated Control Room - Human Factors
- Licensing Process Issues
- Digital instrumentation and control systems in nuclear fuel facilities.

Task Working Groups were formed to address each issue. Results to date include the issuance of five Interim Staff Guidance (ISG) documents with the ISGs for Licensing Process Issues (ISG-6) and Digital I&C Systems in Safety Applications at Fuel Cycle Facilities (ISG-7) completing development. A pilot program for ISG-6 is under consideration. ACRS considers that these

ISGs have provided much needed and valuable guidance for industry and for staff reviews.

Another noteworthy research project is the study on Diversity Strategies for Nuclear Power Plant Instrumentation and Control Systems. The focus of the study is to develop a metric for system diversity and to address the issue of how much diversity is sufficient for safety systems.

6. FIRE SAFETY

Background

The current focus of the fire safety research program is to support the NRC's regulatory needs during licensee transitions to risk-informed, performance-based fire protection programs that meet the requirements of 10 CFR 50.48(c) and the referenced 2001 edition of the National Fire Protection Association (NFPA) Standard NFPA-805. Continuing research activities also develop improved information to support deterministic fire protection programs for licensees that do not adopt a risk-informed approach. In addition to direct support for currently operating reactors, the fire safety research programs also provide input to new reactor licensing reviews and assessment of the risk from fires in new reactor designs.

Current Research Activities

The NRC research activities in fire safety are depicted in Figure 1. Research activities primarily support programs in the Office of Nuclear Reactor Regulation (NRR) and, to a lesser extent, the Office of Nuclear Material Safety and Safeguards (NMSS). These activities are grouped into four technical areas:

- fire modeling,
- fire testing,
- fire & electrical systems analysis, and
- fire risk assessment.

There are also efforts underway on fire research knowledge management. The following sections briefly discuss the major projects in each area.

Fire Modeling

Fire models provide a phenomenological basis for the evaluation of fire growth, detection, and suppression, and the analysis of potentially risk-significant fire scenarios.

CHRISTI-FIRE

The objective of the CHRISTI-FIRE (Cable Heat Release, Ignition, and Spread in Tray Installations during FIRE) experimental program is to perform fire tests on grouped electrical cables to better understand the fire hazard characteristics, including heat release rate and flame speed. This type of quantitative information will be used to develop more realistic models of cable fires for use in fire probabilistic risk assessment (PRA) analyses.

In 2007, NRC and EPRI completed a collaborative project for verification and validation (V&V) of five commonly used fire modeling codes (NUREG-1824). That effort identified strengths and weaknesses in each code with respect to specific fire modeling issues. In 2008, a Phenomena Identification and Ranking Table (PIRT) exercise was completed to assess the predictive capabilities of these fire models for a number of postulated fire scenarios (NUREG/CR-6978). A collaborative project between RES and EPRI is using the results of the verification and validation of fire models along with the PIRT evaluations to develop an integrated "Fire Model Users' Guide" for the approved models. The Guide will describe the capabilities of each code to evaluate specific phenomena during realistic fire scenarios, including limitations, precautions, and lessons learned from practical applications. The Users' Guide will benefit risk analysts who use the fire models to evaluate potentially important fire scenarios, and it will support inspection efforts that use the models as input to the significance determination process (SDP).

Future research projects are planned to extend the current fire modeling capabilities to address specific limitations documented in the V&V reports and the PIRT evaluations, and to improve predictions of phenomena observed during fire experiments.

Fire Testing

This element of the research program includes two major projects.

The Cable Heat Release, Ignition, and Spread in Tray Installations during Fire (CHRISTI-FIRE) program is performing full-scale fire tests to measure the heat release rates and flame spread characteristics from fires in bundles of electrical cables. The tests are conducted with realistic cable tray configurations and loadings, and they include cables with thermoplastic and thermoset insulations. The quantitative data collected by this project will be used to develop more realistic cable fire models for PRAs and to enhance the predictive capabilities of fire modeling codes.

Small-scale fire tests will be performed to evaluate the performance of spent nuclear fuel shipping cask seals during beyond-design-basis fires that exceed the manufacturers' rated temperatures. This information will be used by NMSS to further develop risk insights related to the transportation of spent fuel.

Fire & Electrical Systems Analysis

The experience from actual fire events confirms that damage to electrical cables may disable equipment and cause unexpected responses from instrumentation and control (I&C) signals. Realistic evaluation of fire-induced circuit damage, particularly involving spurious signals caused by "hot shorts", is a very significant effort in fire risk assessments and in pilot applications of the methods described in NFPA-805 and NUREG/CR-6850.

The Direct Current Electrical Shorting In Response to Exposure-Fire (DESIREE-FIRE) project will extend the Cable Response to Live Fire (CAROLFIRE) test program (NUREG/CR-6931) to examine fire damage to DC circuits. The tests are being performed in collaboration with EPRI. They will include realistic cable tray configurations and loading, and they will evaluate the effects from fire damage to circuits for DC-operated components and typical I&C applications.

A planned future project will compile the results from industry-sponsored cable tests, CAROLFIRE, and DESIREE-FIRE. A structured expert elicitation process will examine the available data and develop conditional probabilities and uncertainties for various AC and DC circuit failure modes that may be caused by fire-induced cable damage.

Fire Risk Assessment

Integration of fire risk into a full-scope PRA framework is a very important research activity. The requirements of NFPA-805 and the guidance in NUREG/CR-6850 provide cornerstones for the development of risk-informed, performance-based fire protection programs and a comprehensive assessment of fire risk.

A collaborative project with EPRI is evaluating elements of human reliability analysis (HRA) methods that apply to post-fire mitigation actions. The goal of this project is to recommend specific methods and guidance for the modeling and quantification of human errors during fire scenarios.

An NMSS-sponsored quantitative risk assessment is being performed to evaluate the risk from potential fires and explosions of "red oil" at the Mixed-Oxide Fuel Fabrication Facility. This effort included a pioneering attempt to use the methods of probabilistic risk assessment to estimate risks of red oil fires in fuel reprocessing facilities. The feasibility of using probabilistic methods rather than the usual integrated safety assessment methods for nuclear facilities is noteworthy. The results from this probabilistic study will be used to support design and licensing reviews.

Pilot applications of the methods described in NFPA-805 and NUREG/CR-6850 are currently in progress. A revision to NUREG/CR-6850 is planned to incorporate lessons learned from these applications and improvements to fire modeling and fire-induced damage assessments from on-going research activities.

<u>Fire Research Knowledge Management</u>

Fire research continues to be a rapidly-evolving element of the RES mission. Compilation and dissemination of the information gained from this research, including new results and insights, is vital for understanding the current state of knowledge and planned near-term advancements. Three excellent resources prepared by RES include:

- "The Browns Ferry Nuclear Plant Fire of 1975 and the History of NRC Fire Regulations," NUREG/BR-0361, January 2009. This DVD preserves the history of the Browns Ferry fire and documents its influence on the development of enhanced fire protection regulations.

- "Fire Protection and Fire Research Knowledge Management Digest," NUREG/BR-0465, Revision 1, February 2009. This CD contains a compilation of fire-related reference materials that are useful for NRC inspectors, reviewers, licensees, and other stakeholders. It includes 10 CFR Part 50, guidelines for fire protection, fire inspection manuals and procedures, generic letters, bulletins, information notices, regulatory guides, and fire-related NUREG reports.

- "A Short History of Fire Safety Research Sponsored by the U.S. Nuclear Regulatory Commission, 1975 – 2008," NUREG/BR-0364, June 2009. This report provides an historical perspective on NRC-sponsored fire safety research, summaries of current research activities, and planned near-term research programs.

These references are updated annually.

Assessment and Recommendations

The NRC's fire safety research program is at the forefront of a rapidly evolving understanding of fire events and the assessment of risks from fire damage in nuclear power plants. Structured collaboration with industry provides cost-effective solutions and technical insights that surpass independent efforts. The NRC is a leader in national and international fire safety research.

The focus and priorities for current research projects are determined primarily by user-identified needs. This process is responsive to immediate and near-term technical issues, and it should remain an important part of integrated planning. However, research priorities and programs should more actively anticipate emerging applications and intermediate- to long-term requirements that are not fully dictated by current user needs.

Data and insights from the CAROLFIRE tests have considerably advanced the understanding of fire-induced cable damage. It is expected that the results from current projects such as CHRISTI-FIRE and DESIREE-FIRE will also provide valuable knowledge. The NRC should continue to encourage and support additional testing and fire experiments through collaborative efforts with US industry and international organizations.

Other current research projects are also evaluating HRA methodologies. The goal of those activities is to recommend or develop an integrated methodology that captures the best practices and reduces the variability of assessments from the current assortment of HRA methods. The resulting integrated HRA methodology should apply equally to scenarios that are initiated by internal events, internal hazards (e.g., fires and floods), and external events (e.g., earthquakes) during any plant operating mode (e.g., full power, low power, or shutdown). The fire research described above is focused primarily on only one facet of this problem (i.e., human performance after fires that occur during full power operation). This work should be more closely coordinated with other RES activities to avoid potential duplication or divergence of NRC-sponsored HRA research and conclusions.

Guidance is currently being developed for the assessment of risk during low power and shutdown operating modes. Many of the current fire research programs and results are directly applicable to the evaluation of fire damage during these operating conditions. However, additional information is needed to evaluate such issues as fire initiation frequencies, human-caused fires, effectiveness of detection and suppression, and propagation of heat and smoke through compromised fire barriers that are uniquely associated with personnel activities and SSC configurations during plant shutdown.

New reactor designs rely heavily on integrated digital instrumentation, protection, and control systems. Extensions of operating licenses and obsolescence of analog equipment will also likely lead to replacement of many currently installed I&C systems with digital platforms. Limited information is available regarding the effects from fire, heat, and smoke on digital equipment. Research projects should address such issues as the effects from fire damage and heat on fiber optic cables, the effects of heat on digital equipment, and smoke damage to digital signal processing and computation modules.

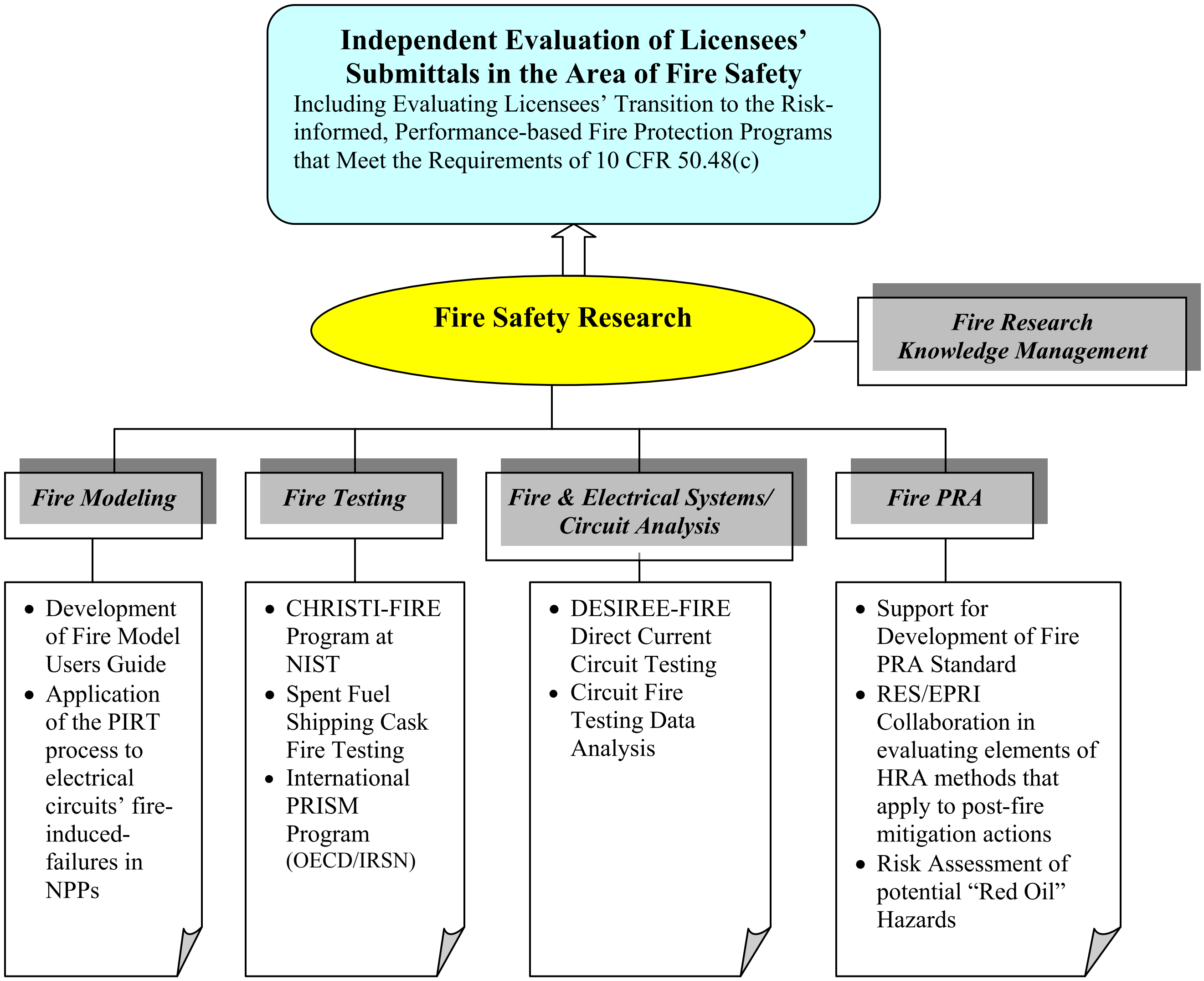


Figure 1. Current NRC Research Activities in Fire Safety

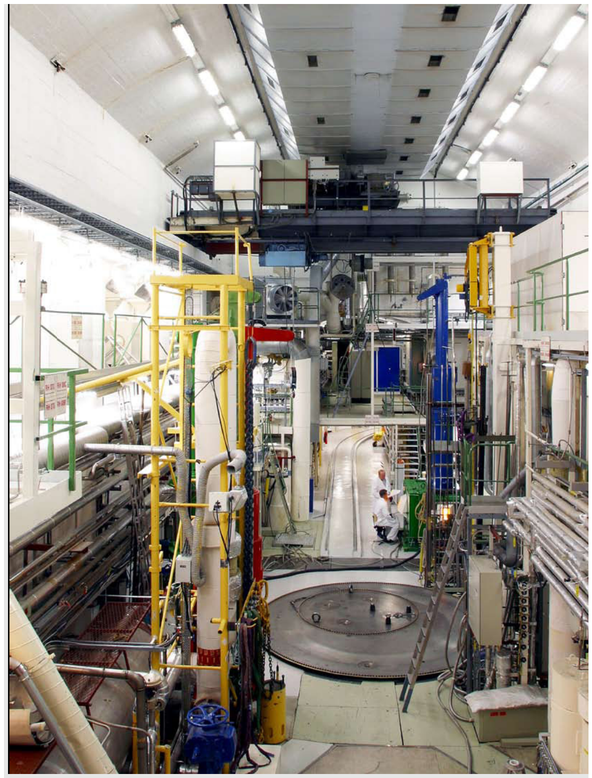
7. REACTOR FUEL

Background

Fuel integrity is an important element of nuclear safety technology. The oxide fuel pellets and their zirconium alloy cladding are the initial physical barriers against inadvertent release of radionuclides to the environment in the event of an accident. The ability of nuclear fuel to operate without failure in the demanding thermal, mechanical, radiation, and chemical environments of BWR and PWR cores has improved dramatically over the past three decades. The various mechanisms capable of causing fuel failure have been identified and addressed effectively by materials, design, and operational improvements. The industry's current goal of achieving zero fuel failures during normal operation by 2010 is within reach.

During this same period, performance differences among fuel designs provided by various manufacturers have become relatively small, and many utilities have developed the capability to operate their cores with fuel assemblies provided by two or more suppliers. Fuel manufacturers have focused on evolutionary design changes with the occasional introduction of new materials such as M5 and Zirlo fuel cladding. Not only have these compositional changes improved the corrosion and hydriding resistance of fuel cladding during normal operation, they have also improved resistance to embrittlement during loss of coolant accidents.

The fuel industry is now evaluating more significant design changes. Lead use assemblies are in operation in which chemical additives have been used to improve fission gas retention and mechanical properties of the UO_2 fuel pellets. Both new and evolutionary fuel cladding materials are in the



Halden Reactor Project (HRP)

Halden tests on high-burnup fuel under loss-of-coolant accident conditions supported NRC research on cladding embrittlement.

The Norwegian Institute for Energy Technology (Institutt for Energiteknikk or IFE) manages the HRP for the Organization for Economic Cooperation and Development/Nuclear Energy Agency. The HRP is based at IFE's facility in Halden, Norway. This facility includes the Halden Boiling Water Reactor (HBWR), which currently operates at 18 to 20 megawatts. The HBWR is fully dedicated to instrumented in-reactor testing of fuel and reactor materials. The NRC has used numerous research products from this internationally funded cooperative effort.

development and regulatory pipeline. In BWRs, changes in fuel bundle design from the current 10x10 arrays to 11x11 are under evaluation as are changes in the composition and thermal processing of fuel channels.

The NRC must maintain the capability to evaluate the adequacy of the testing and analyses supporting these new developments and the impact of these changes on nuclear safety.

Current Research Activities

The overall NRC fuel behavior research program is summarized in Figure 2. As shown, the research is categorized from left to right by sources of experimental and analytical data, regulatory issues of interest, analytical models in which new information will be incorporated , and finally by users and user needs.

The suitability of new cladding materials is determined by extensive vendor testing in laboratories and test reactors, and by performance evaluations of lead-use assemblies in operating power plants. Fuel suppliers typically perform laboratory tests to evaluate cladding embrittlement during loss of coolant accidents. However, fuel suppliers rarely validate predictions of fuel performance during design basis accident conditions with experimental data. In the past, the industry relied on NRC to provide information on fuel behavior under accident conditions. Because of the cost, complexity and requirements for specialized test facilities such data are currently obtained from internationally funded programs such as those at CABRI or Halden.

Experimental data on the performance of fuel and cladding systems under severe accident conditions are not provided by licensees. Severe accidents, of course, pose the bulk of the risk to the public health and safety. Experimental data obtained in research sponsored by the NRC show that cladding interactions with the fuel play a central role in the rate and extent of core degradation and fission product release under severe accident conditions. For example, recent single rod loss-of-coolant accident (LOCA) experiments at the Halden reactor project have demonstrated conditions under which very high burnup fuel can fragment, relocate into the ballooned volume and subsequently be ejected from the cladding. These data will be invaluable in determining future high burnup limits.

The NRC is in the process of completing experimental studies of high-burnup fuel and cladding behavior under design basis accident conditions. The NRC has sponsored in-pile tests of fuel behavior during reactivity insertion events and out-of-pile tests of fuel behavior under design basis loss of coolant conditions. These experimental investigations have been conducted using an impressive combination of national and international collaborations. Results of the research have led to well considered proposals for changes in the regulations. The proposed changes would make the regulations more realistic and could decrease burden on both the staff and licensees, especially as new fuel and cladding systems are proposed. Implementation of these proposed changes to the regulations has been slow. Acceptance of the changes to the regulations now awaits results of additional testing that could be regarded as confirmatory in nature.

The staff has also completed revisions of its fuel performance computer codes FRAPCON and FRAPTRAN. These computer codes are used to independently confirm analyses done by the vendors and other licensees. The modifications allow the computer codes to be used to evaluate fuels taken to burnups of up to 62 GWd/t and to evaluate the performance of mixed oxide (MOX) fuels. MOX fuels have been irradiated in the Catawba reactor as part of a DOE program to dispose of excess weapons-grade plutonium. The NRC is also examining the severe accident behavior of high-burnup and MOX fuels in its severe accident research program.

Assessment and Recommendations

The upgraded fuel performance models appear to meet most near-term agency needs. There is ample capability to assess risks of fuel failure due to fuel center melting or departure from nucleate boiling events. However, the NRC still does not have empirical or analytical capability to quantitatively assess the more likely risk of Pellet Cladding Interaction (PCI) fuel failures during anticipated operational occurrences (AOOs) on current or future fuel designs. The stress corrosion fuel failure mechanism that is active during some AOOs may compromise fuel integrity as licensees continue to uprate power, increase fuel burnup, and introduce new fuel and cladding materials. Despite these needs, NRC is not developing any quantitative capabilities to assess the fuel vulnerability to PCI during operational transients. There is a wealth of experimental data on the PCI phenomenon available to the NRC to develop and validate a practical PCI failure model for use in regulatory decision making.

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

Lead-test assemblies of MOX fuels have emerged from their second cycle of irradiation in the Catawba reactor. These MOX fuels are being tested as part of a DOE program to dispose of excess weapons-grade plutonium by using it as reactor fuel. It appears that the NRC does not have plans for any research examinations of these novel fuels either after the first cycle of irradiation or after subsequent cycles of irradiation. In light of the limited NRC experience with MOX fuels, the ACRS recommends that there be a research program to follow closely the post-irradiation examination of the lead-test assemblies planned by DOE.

NRC must maintain the expertise in the area of reactor fuel to respond to future design changes. Because of license extensions, power uprates, and the prospect of additional new reactors, it is anticipated that the vendors will introduce new fuel and cladding systems. The challenge in maintaining what amounts to an essential core competency of the agency arises because of the limited availability of expertise outside of the agency that is independent of licensees. Current manpower working for the agency in this field either directly or by contract is experienced and there is a need to groom newer generations in the field. A major difficulty in doing so is the decline in the United States of in-pile test facilities and hot cells for examinations of irradiated fuels and cladding. Long-term collaboration with international partners having the facilities and staff capable of undertaking pertinent in-reactor studies may well be essential for NRC to maintain an adequate level of expertise in reactor fuels. One option available to the NRC for the development of fuel staff expertise is the stationing of promising staff members at appropriate international projects such as the Halden project for one to two years. This opportunity has been widely used by industry and should be considered by the NRC.

Reprocessing of nuclear fuels may be an emerging technology that merits research attention by NRC. Aqueous reprocessing of irradiated fuels has been done for many years within the nuclear weapons community using the PUREX process. Though familiar, this process has not been trouble free and there are many known hazards. NRC has gained some exposure to the associated safety issues of "red oil," hydroxylamine nitrate, ammonium nitrate and the like through its review of the construction authorization application for the DOE's MOX Fuel Fabrication Facility. Modifications of the PUREX process – the so-called UREX processes – were being considered by DOE in its now abandoned Global Nuclear Energy Partnership initiative. Undoubtedly, safety issues similar to those of the PUREX process and new safety issues will arise if these new processes are pursued.

Even more challenging will be the pyrometallurgical reprocessing of spent nuclear fuel which was also being considered by DOE as a longer-term development. In the current political environment, it is unlikely that the NRC will face the challenge of regulating technologies other than modest changes to the PUREX process.

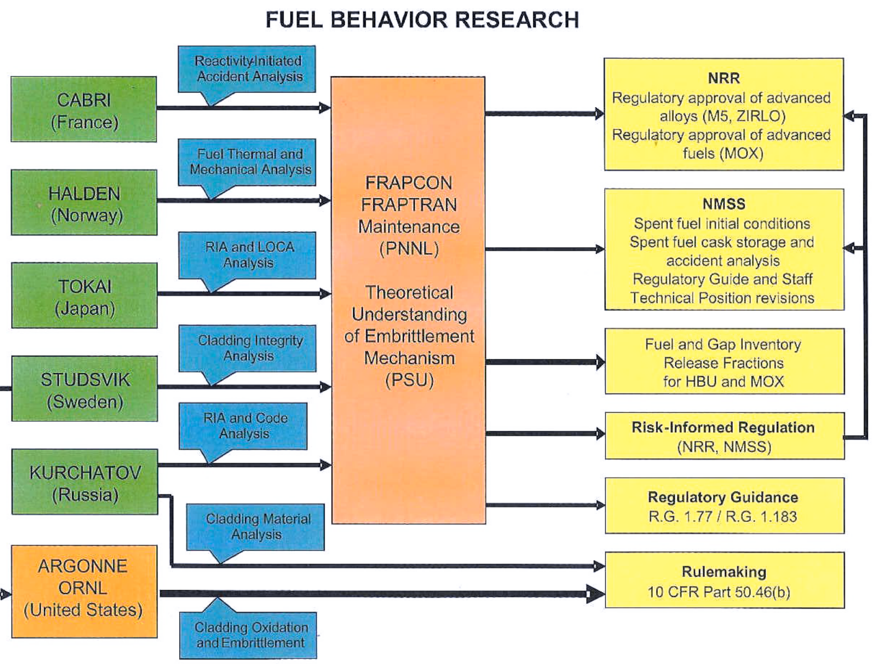


Figure 2. Current NRC Research Activities in Reactor Fuel

8. HUMAN FACTORS AND HUMAN RELIABILITY

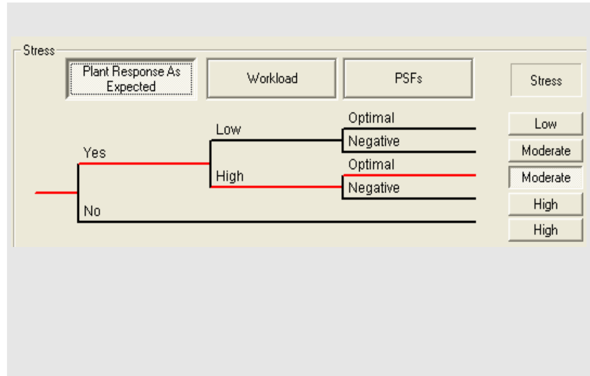
Human performance continues to be a significant element of the safe operation of nuclear facilities. The challenge NRC research faces is how to respond using the current state of the practice to short-term needs emanating from program offices while at the same time maintaining a research program that is long-term and aims at improving our understanding and modeling human performance. To achieve an appropriate balance between these two competing objectives the staff has organized its activities around four goals:

- ensuring that Human Factors (HF)/Human Reliability Analysis (HRA) methods and guidance have sound, up-to-date technical bases;
- expanding the HF/HRA infrastructure for anticipated changes made by the nuclear industry;
- improving HF/HRA methods to reduce uncertainty and promote the state of the art; and
- maintaining the infrastructure of expertise, facilities, capabilities, and data.

These goals of the staff's research activities are well directed towards meeting the agency needs (see also Figure 3). The staff's participation in domestic and international collaborations strengthens these activities and should be encouraged. These collaborations include various OECD groups, EPRI, NASA, and professional organizations such as ANS, IEEE, and AIChE.

Human Factors Research

10 CFR Part 26, "Fitness for Duty Programs" (FFD), was updated and published in the Federal Register on March 31. 2008. To ensure consistency among various offices, the staff is developing training materials for personnel involved in the implementation of the updated Part 26.



Recognizing that the methods for assuring personnel 'fitness for duty" are evolving, the staff is exploring the use of new methods in a project titled "Methods for Assessment of FFD." Both of these activities are important to the regulation of existing nuclear power plants (NPPs) and should continue.

The new generation of nuclear power plants will differ from the current plants in three important ways:

- the plant design,
- the extensive use of digital instrumentation and control (DI&C) systems,
- and human-system interface (HSI).

Consequently, Human Factors Engineering (HFE) must adapt to this new reality. Recognizing this need, the staff convened a group of experts from research organizations, vendors, and utilities to prioritize research needs. NUREG/CR-6947, "Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants," contains the results of this process, i.e., the prioritization of the research areas and the rationale for the ranking. These results are used by the staff to develop a long-term research plan. The ACRS commends the staff for taking such a systematic approach to the prioritization of research topics.

Of the twenty topics identified as having high priority, the staff is currently pursuing two. The project "Automation and Complexity" aims at developing guidance for reviewing the operators' interface with automation displays. The project "Human Factors under Degraded I&C" investigates how degraded or failed I&C systems may affect operator situation awareness and performance. Both of these projects will contribute to the expansion of the human factors infrastructure that will enable the agency to deal with anticipated changes in the industry. This research is supported by experiments performed at the Halden Man-Machine Laboratory (HAMMLAB) in Norway. HAMMLAB uses a simulator control room that can be configured as a prototype advanced control room with an integrated surveillance and control system.

A testing capability similar to that at Halden's HAMMLAB does not exist in the US. Some have advocated that such a capability be developed in this country. It is not evident that such a capability would be cost effective. It is noteworthy that the next human reliability benchmark test (discussed below) will be conducted at a simulator for a US plant.

Human Reliability Analysis

Human Reliability Analysis (HRA), as an accident sequence evolves, is an area where several models exist. For example, the EPRI software tool "HRA Calculator" recommends the Cause-Based Decision Tree Model and THERP as the default methods for all post-initiating-event human failure events. The NRC itself is using ATHEANA and SPAR-H. There is clearly a need to investigate and evaluate the validity of the assumptions behind these models and to recommend classes of problems for which a model is appropriate.

This need has been recognized by the Commission. The Commission has asked the ACRS to "work with the staff and external stakeholders to evaluate the different HRA models in an effort to propose either a single model for the agency to use or guidance on which model(s) should be used in specific circumstances".

In response, the staff has initiated two research projects. The project "HRA Method Benchmarking" is utilizing HAMMLAB to simulate accident sequences. It is co-sponsored by Halden, the Swiss Federal Nuclear Inspectorate, and EPRI. The objective is to collect and analyze crew performance data, to apply HRA models to predict crew performance, and to evaluate these models on the basis of a comparison of the simulator data with the model predictions.

The project "HRA Model Differences" will build on the conclusions from the benchmarking project mentioned above and will recommend appropriate models as the SRM requires. Insights gained from operating experience will play an important role in the formulation of this recommendation. The staff is continuing the development of the Human Event Repository and Analysis (HERA) system. HERA is developing taxonomy for collecting human performance data, collecting real and simulated events, and developing tools enabling the use of HERA to inform human reliability analyses. If successful, these HRA projects will offer a significant advance in the evolution of methods.

In addition to nuclear reactor applications, the staff's HF/HRA research activities include several worthwhile projects aimed at nuclear facilities including:

- "HRA for Byproduct Materials'"
- "Qualitative HRA for Spent Fuel Handling'"
- "AREVA Fuel Cycle Facility Review,"
- "GEH Fuel Cycle Facility Review," and
- "MOX Fuel Cycle Facility Review."

Overall, the ACRS finds the HF/HRA research program and activities to be well based on sound rationale and to be focused on both near- and long-term needs of the agency.

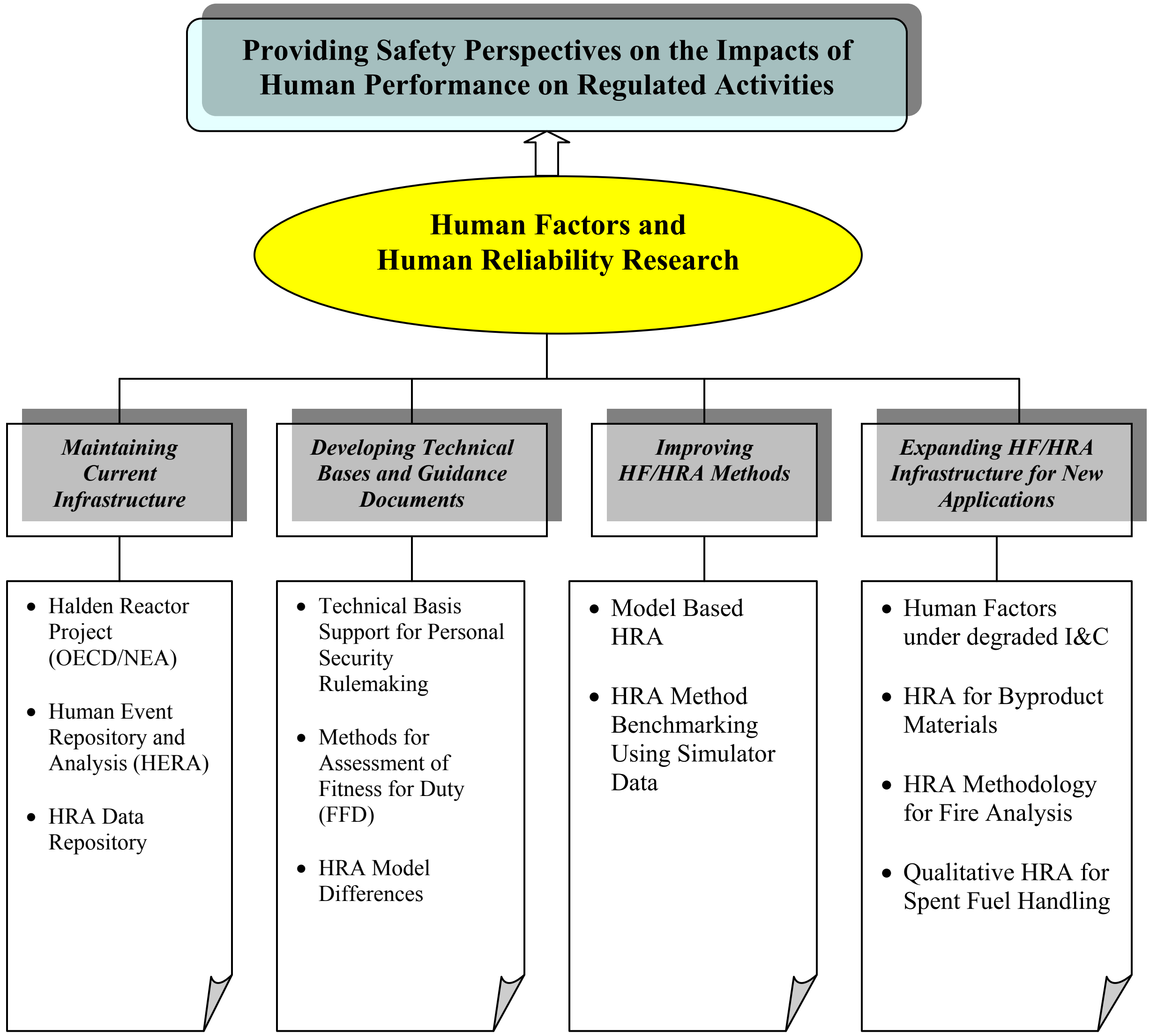


Figure 3. Current NRC Research Activities in Human Factors and Human Reliability

9. MATERIALS AND METALLURGY

Background

Materials and metallurgy continues to be the most active area of research within the NRC. This is appropriate in light of the efforts required by the agency to address known and emerging materials degradation phenomena in aging LWRs, and to monitor the effectiveness of licensees' aging management programs. As plants age, known degradation mechanisms will continue to affect components important to safety, and new degradation mechanisms may develop. The agency must develop the technical capabilities to minimize the risk of material degradation surprises and to assess the effectiveness of the industry initiatives to deal with materials degradation.

Sectioning of reactor vessel head penetrations from Washington Nuclear Power, Unit 1, a canceled plant

Current Research Activities

Current materials and metallurgy research activities are conducted by the Component Integrity Branch and the Corrosion and Metallurgy Branch within the Division of Engineering of RES. The former directs research focused on fracture mechanics, non-destructive examination (NDE), and safety assessments, and the latter on corrosion, metallurgy, and advanced reactors. Topical areas of research are grouped into five broad categories:

- Component Integrity Assessments
- Non-Destructive Evaluation
- Environmentally Assisted Cracking
- Proactive Management of Materials Degradation
- Steam Generator Tube Integrity

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

<u>Component Integrity Assessments</u>

This area of research addresses the integrity of dissimilar metal welds in piping, control rod drive mechanisms, and pressure vessels; develops probabilistic and deterministic fracture mechanics tools; evaluates effectiveness of primary water stress corrosion cracking (PWSCC) mitigation methods; and validates methods used to determine weld residual stresses (WRS) in dissimilar metal welds. In addition to research on metallic components, work is being done to confirm the acceptability of high density polyethylene (HDPE) piping for long term safety related applications.

A new research project has been initiated to develop and validate a probabilistic fracture mechanics tool to assess extremely low probability of rupture (xLPR) in piping systems susceptible to degradation mechanisms such as PWSCC. These systems have rupture probabilities of $< 10^{-6}$ per reactor year. As shown in Figure 4, this code will incorporate flaw distributions, fracture mechanics, fluid mechanics, corrosion, and metallurgical variables in a manner similar to that used in development of the new pressurized thermal shock rule.

The PWSCC mitigation program will evaluate the effectiveness of industry developed mitigation strategies such as full structural weld overlay and mechanical stress improvement. The objective of this work is to ensure that the probability of rupture of current leak before break (LBB) systems remains extremely low in PWR components subject to PWSCC. An important input to this research is the validation of estimated weld residual stresses in dissimilar metal welds. This research will include determinations of weld residual stresses in simple plates and cylinders and in complex mockups of PWR components. Weld residual stresses will be measured using x-ray diffraction, neutron diffraction, and incremental deep hole drilling techniques.

High density polyethylene (HDPE) materials have attractive properties for service water and buried piping applications, but they are subject to aging and degradation mechanisms that differ from those of metal piping and other types of piping conventionally used. A growing number of licensees propose to use HDPE in safety-related applications. Failure mechanisms and service life as affected by temperature, time, and flaws need to be researched. The research will include fracture-mechanics-based flaw tolerance evaluations of base materials as well as fusion joints.

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

Major progress has been made in completing and closing several research projects on reactor pressure vessel integrity and incorporating remaining research in the component integrity assessments program. This research has led to updates in several NRC regulatory documents as well as ASME and ASTM codes and standards. Revisions in the Pressurized Thermal Shock (PTS) screening criterion in the newly issued PTS rule and the associated regulatory guides and Appendices G and H to 10 CFR Part 50 provide great benefit by quantifying excessive conservatisms in prior regulations and allowing longer service life of reactor pressure vessels.

The staff is now focusing pressure vessel research under two programs. The “Pressure Boundary Materials” program will develop and validate predictive material property models aimed at the refinement and generalization of structural integrity assessments. The “Probabilistic Pressure Boundary Safety Assessment” program will develop more realistic flaw evaluation tools and fracture mechanics models to replace deterministic failure criteria with probabilistic criteria.

Non-Destructive Evaluation

Various NDE methods are relied upon to monitor the integrity of reactor coolant systems. These include ultrasonic testing, eddy current testing, penetrant testing, radiographic testing, and visual testing. The reliability and effectiveness of these methods can vary considerably depending on component geometry, materials, and types of defects. There are two ongoing projects addressing NDE technology. These have been expanded to address the effectiveness of NDE methods applied to high density polyethylene piping materials and joints.

"Reliability of NDE for Nuclear Power Plant In-Service-Inspection" is a project focused on quantifying the reliability of a broad range of NDE techniques used in power plant in-service inspection (ISI) programs. This task was initiated in 2007 to evaluate the accuracy and reliability of the NDE methods and to provide recommendations to the staff to improve the effectiveness and adequacy of ISI programs. This research, which is expected to be completed in 2012, covers the entire spectrum of NDE techniques used in reactor construction, in-service inspection, and repairs. Specific evaluations of the effectiveness of ISI techniques will be provided for:

- PWSCC in Alloy 600, 82, 182 dissimilar metal welds and J-groove penetrations,
- IGSCC in austenitic stainless steel welds,
- degradation in cast stainless steel and weldments,
- repairs including overlays, cladding and in-lays,
- reactor internals examinations,
- vessel penetrations, and
- HDPE piping

The project "Assess Emerging NDE for Dissimilar Metal Welds" is intended to provide data and correlations necessary for NRC staff to independently evaluate licensee programs for assessing the integrity of dissimilar metal welds. In addition, the program will continue to evaluate emerging NDE techniques such as phased array UT and synthetic aperture focusing techniques for the inspection of coarse grained materials such as cast stainless steels.

Environmentally Assisted Cracking

Environmentally assisted cracking (EAC) is a generic term for the various stress corrosion cracking mechanisms that can be active in BWRs and PWRs. These complex phenomena are influenced by applied and residual stresses, water chemistry, radiation exposure, temperature, material composition, microstructure, and fabrication history. In BWRs, these mechanisms are known as intergranular stress corrosion cracking (IGSCC) and irradiation assisted stress corrosion cracking (IASCC). The intergranular stress corrosion cracking occurs on stainless steel welds outside the core regions and the irradiation assisted stress corrosion cracking occurs in radiation hardened stainless steel core internals. In PWRs the dominant mechanisms are primary water stress corrosion cracking (PWSCC) and IASCC. These phenomena occur on susceptible nickel base alloys and dissimilar metal welds.. In recent years, EAC has occurred in components internal to the vessels in BWRs and PWRs, and in reactor vessel penetrations and dissimilar metal welds. In addition to primary system leakage, EAC can lead to serious secondary damage. At Davis-Besse PWSCC cracks in control rod drive mechanism (CRDM) nozzles led to leakage of borated coolant and accelerated corrosion of the pressure vessel head.

Licensees have implemented improved inspection methods, resistant materials and repair procedures, and improved water chemistries to prevent EAC. These actions have been effective in reducing the frequency of EAC events, but they have not eliminated the problem. Environmentally assisted cracking continues to occur in nuclear power plants as metal components age and radiation exposure increases.

The NRC staff must maintain capabilities to evaluate licensees' analyses of the active degradation phenomena in their plants, and the effectiveness of implemented or proposed mitigation methods. The research projects now under way are designed to ensure that the NRC has the necessary technical understanding of the root causes of the various environmental degradation phenomena, their underlying mechanisms, and the long-term reliability of mitigation methods.

The effectiveness of the various materials, fabrication and water chemistry changes introduced to mitigate EAC in BWRs and PWRs should be confirmed by long-term and aggressive confirmatory testing. Some mitigation methods may lose effectiveness over time, while others (individually or in combination) may demonstrate the ability to protect critical components from stress corrosion cracking for the 60 year life of the uprated plants

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

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

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

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

<u>Proactive Materials Degradation Assessment</u>

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

The staff completed Phase 1 of the project in 2008. A comprehensive assessment of the likelihood and safety significance of possible environmental degradation mechanisms was completed for approximately 1900 BWR and PWR components, and NUREG/CR-6923 documenting this work was issued. The objective of Phase 2 of the project was to establish agreements with industry and international organizations to define research tasks addressing the identified issues of greatest concern. The Zorita Internals Research Program was such an activity in which high fluence core internal components which had operated for over 30 years would be examined. Today the program has evolved to:

- establishing a refined plan
- establishing international collaboration
- targeting items of low knowledge and high/medium susceptibility to degradation, and
- creating an information tool.

The ACRS enthusiastically supported the initial vision of the Proactive Materials Degradation Assessment Program and has recommended its continuation. However, the program seems to have lost its momentum and focus. Current research appears to concentrate on conventional reactive management instead of anticipating and preventing materials degradation surprises. The work published in NUREG/CR-6923 identified target items of low knowledge and high-to-medium susceptibility to degradation. These target items were essentially predictions of possible materials degradation problems which had yet to occur. They represented areas of greatest uncertainty and concern to the experts involved in the identification process. It would seem that the next step in the proactive process should be the validation of one or more of these predictions by experimental means or by focused inspections at operating plants. In the absence of a quantitative confirmation of the predictions made in NUREG/CR-6923, it is doubtful that the program can attain its primary goal. If such confirmations are not possible, continuation of the program should be reconsidered.

Steam Generator Tube Integrity

Rupture of steam generator tubes in PWRs can lead to accidents that allow radioactive materials released from the core to bypass the reactor containment and enter directly into the environment. Severe accidents involving containment bypass can be risk dominant in some PWRs. Degradation has been observed on both primary and secondary sides of steam generator tubes, and many different phenomena have been observed including: general corrosion and wastage, denting, crevice corrosion, pitting, intergranular attack, and stress corrosion cracking.

Research and development by the industry and by the NRC under the Steam Generator Action Plan has been very effective in reducing the number and severity of steam generator materials degradation events This has been achieved by improved water chemistry control, development of reliable NDE methods, and the introduction of improved tube support designs and tubing materials such as Alloy 690 in replacement steam generators.

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

With the closure of the Steam Generator Action Plan, the vast amount of knowledge gained during this long term research program is to be thoroughly documented.

Research will continue, to assure that that the safety of steam generators will persist over the life of these systems. A variety of advanced NDE and signal analysis techniques are being evaluated for inspecting original or repaired steam generator tubes.
An analytical model based on improved understanding of the mechanisms of initiation and propagation of stress corrosion cracks in nickel based alloys, and the influence of crevice conditions will be developed and validated.

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

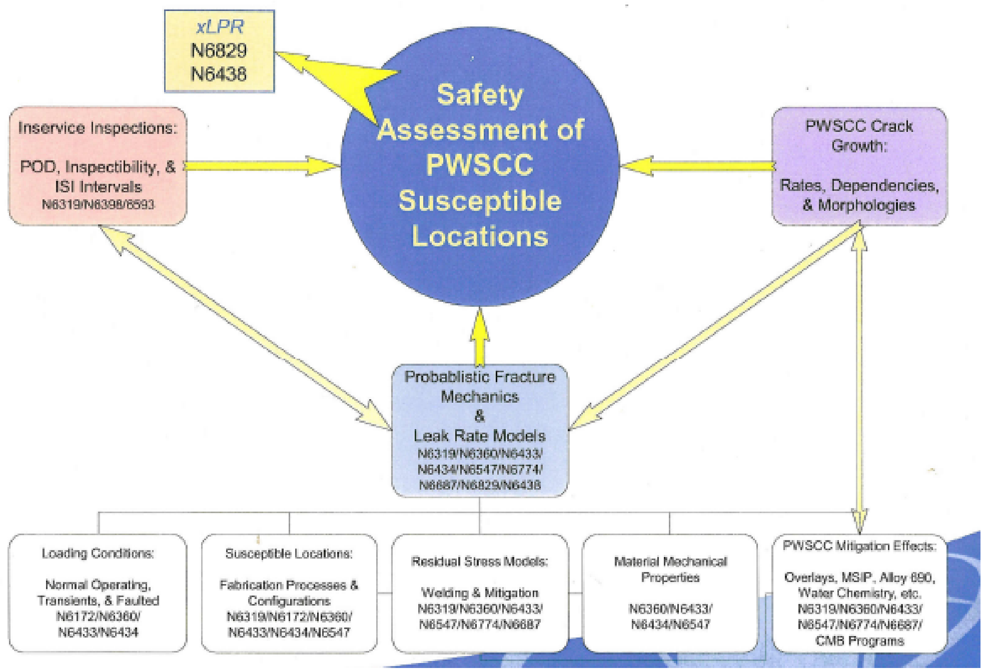


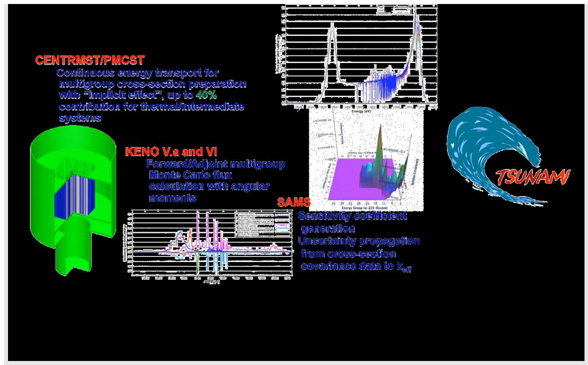
Figure 4. PWSCC Safety Assessment Elements

10. NEUTRONICS AND CRITICALITY SAFETY

Background

Neutronics analysis and criticality safety are among the core competencies essential to the pursuit of the NRC mission. Neutronics and criticality issues are unique to the nuclear sciences. The licensees and, indeed, the public expect that the agency to have capabilities in these areas that are at or very near the state-of-the-art. There is limited availability of independent expertise from the academic community or the private sector in these areas. It is, then, difficult for the agency to rely on external contractors to provide the needed expertise to independently analyze neutronics and criticality issues. NRC must, then, have a strong, in-house, capability in these areas to make technically sound regulatory decisions without imposing conservatisms that are actually unnecessary when the state-of-the-art is examined. The in-house capability certainly must include the ability to execute analyses with the available computational tools. It must also include a respectable understanding of core physics that computer codes attempt to estimate.

Currently, the NRC relies on the SCALE (Standardized Computer Analysis for Licensing Evaluation) suite of computer codes to analyze neutronic issues associated with fuel loading in nuclear reactors. This suite of codes is maintained by Oak Ridge National Laboratory. The SCALE computer codes were developed in the era of batch computer codes and do not take advantage of modern, multiprocessor computational architecture with large memory machines. The SCALE code suite, then, provides an adequate capability for analysis of the current fleet of operating reactors. The cores for the current reactors are simple and have significant symmetry. Such cores do not challenge the computational capabilities of the SCALE code suite.



SCALE/TSUNAMI

SCALE Tools for Sensitivity and UNcertainty Analysis Methodology Implementation (SCALE/TSUNAMI) utilizes first-order-linear perturbation theory to produce the sensitivities of a computed k_{eff} value to constituent cross-section data. The energy-dependent sensitivity data for each reaction of each nuclide in a system model can be quickly computed using SCALE/TSUNAMI's 1-D and 3-D analysis tools. These sensitivity data can be coupled with cross-section-covariance data to produce an uncertainty in k_{eff} due to uncertainties in the evaluated nuclear data.

In addition to the SCALE code suite, NRC has the PARCS (Purdue Advanced Reactor Core Simulator) neutronics analysis code used in conjunction with the TRACE thermal-hydraulic code. PARCS does derive some of its input from the SCALE analyses.

There are economic driving forces that push reactor designers to wring from reactor cores as much profitable power generation as possible. As a result, nuclear cores for advanced light water reactor designs being submitted for certification are not so symmetrical or simple to analyze as the cores of existing light water reactors. Local asymmetry, great variability in fuel enrichment, partial length rods, and burnable poisons complicate the neutronics analysis of these cores. Core analysis is becoming more time consuming. Complexity of analysis coupled with greater demands for multiple analyses to quantify uncertainties in computed results are taxing the capabilities of NRC's resources for neutronics analysis. The labor involved in individual analyses limits the opportunities for exploration of uncertainties and low probability events that may have high consequences.

Challenges to the current capabilities for neutronics analysis will grow as more advanced reactor designs based on technologies other than conventional light water are submitted for certification. The "pebble bed" reactor core for gas-cooled reactors certainly appears to pose challenges for neutronics analysis that will not be easily met with current computational resources. Any move to fuel enrichments greater than 5% ^{235}U will pose challenges to the currently available data bases supporting computations.

Current Research Activities

The current NRC research activities in neutronics analysis and criticality safety are depicted in Figure 5. Research activities are focused on maintenance and application of current capabilities. There are also modest efforts to apply these current capabilities to issues that the agency will have to face in the future.

Assessment and Recommendations

NRC has established an adequate neutronics analysis capability. It has done well maintaining this activity and is trying to find ways to apply the capability developed in a previous era to regulatory issues that arise today. There is not a crisis of need now, but the application of existing capabilities will grow more challenging. Labor intensive work to analyze the core of the ESBWR design is indicative of the challenges that will grow more common as the light water reactor technology advances. The demand for more extensive, quantitative analysis of uncertainties (in the main, parametric uncertainties) for neutronics analysis can be met by Monte Carlo methods, but these methods require at least hundreds of calculations for each core configuration. Such extensive calculations are not practical with existing computational capabilities when applied to complicated core designs.

NRC needs to establish a long term strategy to upgrade its neutronics analysis capabilities to take advantage of modern computer architecture in a way that facilitates the analysis of low symmetry reactor core including quantitative analysis of parametric uncertainty.

A move to reactors based on technologies other than light water is not imminent. There are, however, DOE initiatives in both gas-cooled reactors and sodium-cooled reactors that will challenge NRC neutronics analyses capabilities. Some of the challenges are associated with data bases. NRC can encourage the DOE and foreign counterparts to cooperate in the development of data bases adequate for safety analyses of advanced reactor concepts and reactors using higher fuel enrichments and new fuel compositions such as mixed oxide (MOX) fuels enriched in other actinides.

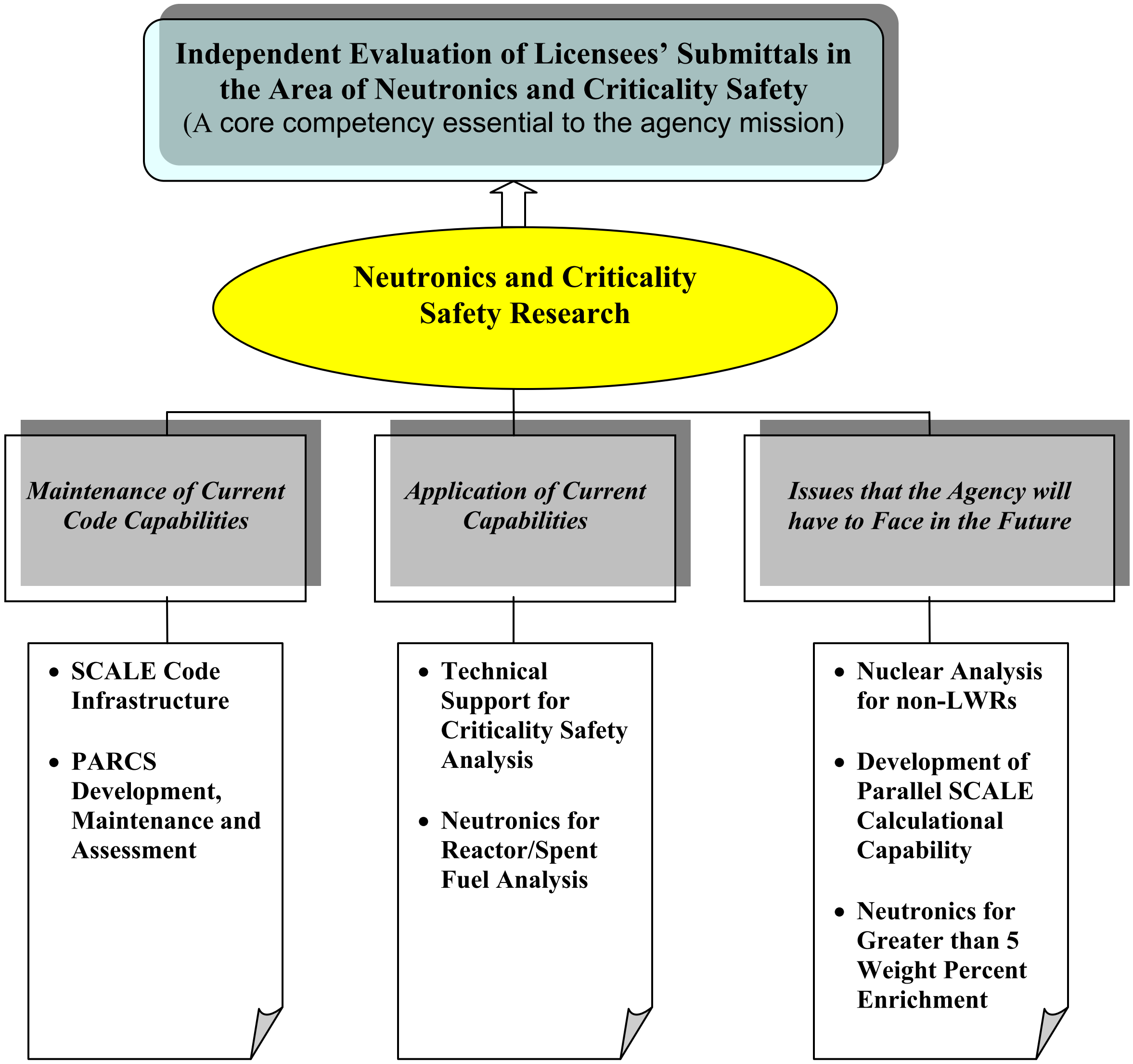


Figure 5. Current NRC Research Activities in Neutronics and Criticality Safety

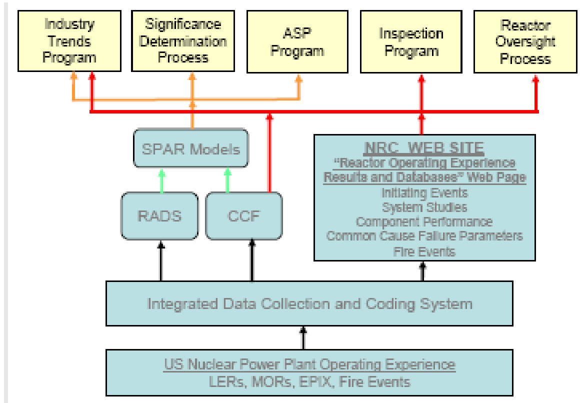
11. OPERATIONAL EXPERIENCE

Background

When properly documented and analyzed, operational data provide an invaluable source of information that can be used to refine the regulatory process and improve its effectiveness. With the increased use of risk information in the regulatory process, operational data become even more important to assure the risk information has an adequate basis for use in decision making. In the operational experience research area, there are several programs aimed at capturing appropriate data, analyzing the data, and providing the tools necessary to use the data in the regulatory process. In addition, the operational data is used as a measure of the regulatory effectiveness and inputs to the annual report to Congress on significant operating events.

Current Research Activities

The RES Data Collection, Analysis, and Trending Programs and their relationship to each other are graphically shown in Figure 6. For example, activity N6448, "Access to INPO's EPIX System," is providing a mechanism whereby NRC staff has access to some of the Institute of Nuclear Power Operations (INPO) operational data through EPIX so it can be used by the NRC. The data is then coded and integrated with other sources of information under research activity N6632, "Reactor Operating Experience Data." This integrated data can then be used as inputs to PRA models, special studies under activity N6890, "Special Reliability Studies," and risk-based operating experience analyses under activity N6631, "Computational Support for Risk Applications." All of these ultimately provide input to programs and processes such as the Significance Determination Process, Industry Trends Program, Inspection Program, and the Accident Sequence Precursor Program.



RES Data Collection, Analysis, and Trending Program

The operational experience research program is an integrated system of data collection, data coding, and tools for use of the data. Although the programs do not fall within a single branch within RES, the coordination between the various groups within RES and NRR appears to be good and results in products that improve the regulatory process. These efforts yield improved NRC PRA models for evaluating existing plants as well as new reactor designs submitted for certification. The operational research activities are providing quantitative measures for the effectiveness of the regulatory process.

Assessment and Recommendations

The Operational Experience Research program is being managed and executed in a manner consistent with the needs of the NRC. It provides data and tools necessary for regulatory decision making and for the assessment of regulatory effectiveness. There appears to be good coordination between the research staff and the user organizations. The Operational Experience Research program should be continued.

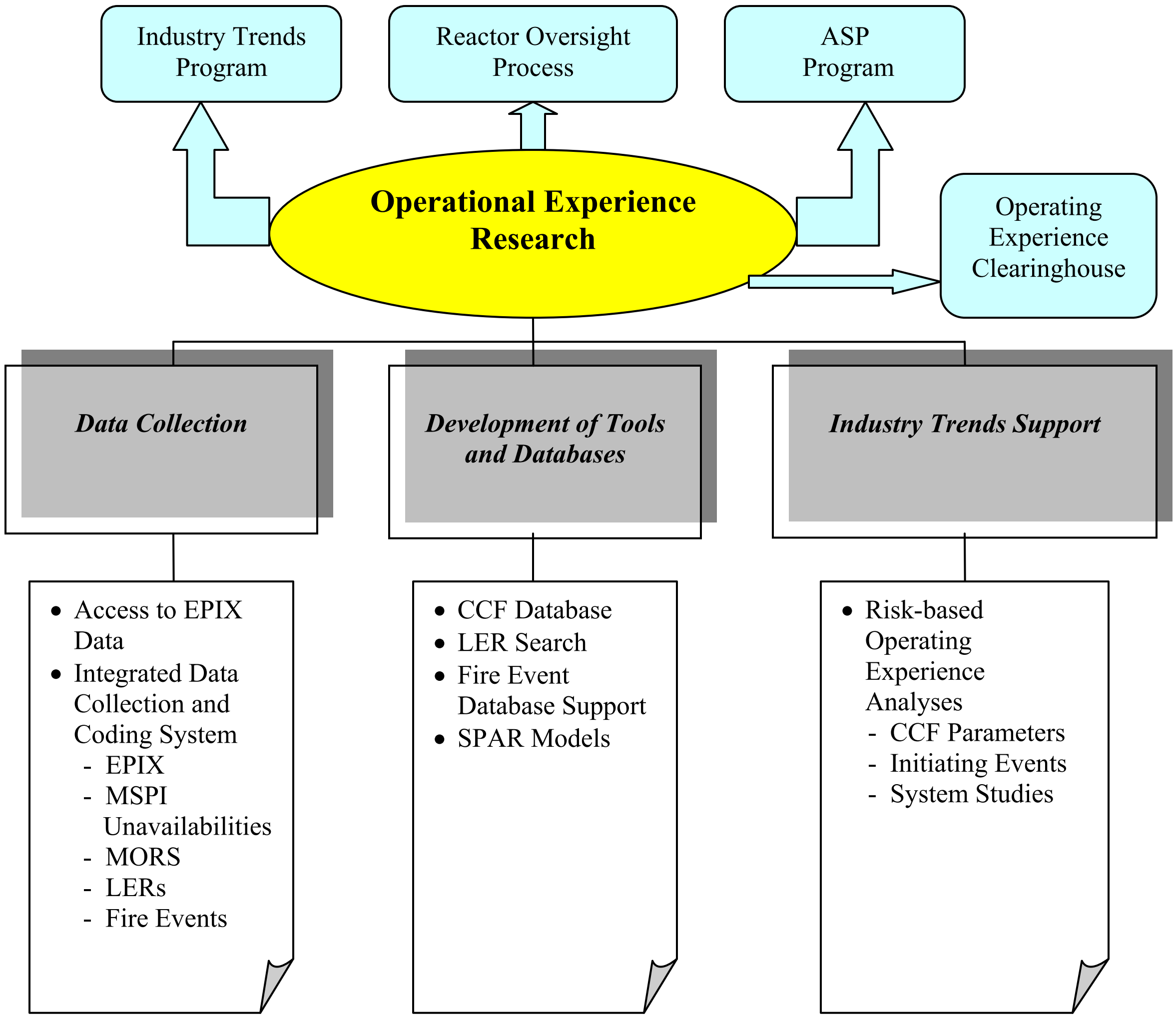


Figure 6. Current NRC Research Activities in Operational Experience

12. PROBABILISTIC RISK ASSESSMENT

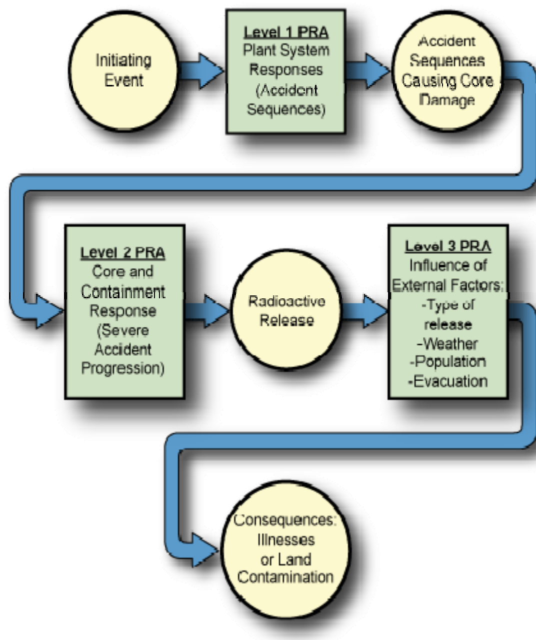
Background

Probabilistic risk assessment (PRA) is an essential technology for NRC as the regulatory system evolves to make greater use of risk information. Therefore, NRC must have state-of-the-art probabilistic risk assessment capabilities. While the agency has been responsible for the development of methods in widespread use today, much of its recent research work has been in the area of applications. Development of methods has not been a priority. The extensive use of risk information by both the industry and the staff in regulatory decision-making, the reviews of new reactor designs, and articles published in the literature have identified areas where methodological advances would be useful.

This report focuses on three aspects of the NRC research program on (PRA)—organization of the research program, technical content including recent advancements and continuing/future needs, and independent capability.

Evolving Organization: Management of the PRA research program is evolving from a reactive mode to an organized goal-driven process. A reasonable first step has been accomplished—mapping the process as it currently exists. The PRA research program managers describe the following four steps in the current process:

- Understanding, cataloging and taking advantage of the previously divergent drivers of PRA research (SRMs, User Needs, Commission mandates, ACRS recommendations, emergent needs, etc.)
- Formally stating the goals (issue resolution, application support, mission support)
- Detailing the alternative mechanisms for funding research (staff work, contracts, agreements and MOUs, and collaboration)



- Identifying and tracking anticipated outcomes

Activity is underway to move toward a more formally organized and managed process focused on meeting both long and short-term goals. One especially encouraging aspect of the process is close cooperation between Division of Risk Analysis of RES and its counterparts in other NRC line organizations (NRO, NRR). Through regular meetings, they have begun to structure current User Needs to advance long-term agency goals.

The ACRS looks forward to a future goal-driven process. More work is needed to identify, organize, and structure the long-term PRA research goals. With a clear idea of what is needed and the time frames in which those needs ought to be realized, RES can be better prepared to take advantage of new opportunities as they present themselves.

Recent Advancements: In its 2006 report to the Commission on the NRC Safety Research Program (NUREG-1635, Vol. 7), the ACRS recommended that "the staff needs to review the literature concerning Binary Decision Diagrams and evaluate the need to adopt this technology." The Committee followed up in its 2008 report (Vol. 8) explaining that "[new] calculation methods, such as those based on Binary Decision Diagrams and EPRI's Direct Probability Calculation (DPM^{TM}), are exploring the possibility of eliminating the approximations in current methods... The ACRS continues to believe that the staff should undertake such an investigation and participate actively in the international activities related to the development of the next generation of PRA software. A short-term product could be an evaluation of the potential impact of the new software on regulatory decisionmaking. The ACRS was informed that there are efforts under way to develop a standard PRA representation format which would promote independence between the logic models and the individual codes. Such a standard would facilitate significantly the review of the PRA logic, a benefit of great value to the agency." RES now is following this work and is trying to understand how such a standard can support improvements in NRC PRA tools and review capabilities.

In its 2008 report to the Commission on the NRC Safety Research Program (NUREG-1635, Vol. 8), the ACRS also pointed out that the "quantification of uncertainties is an essential element of risk-informed decisionmaking. The uncertainties in the values of input parameters to the PRA are usually handled well and there are many software packages that propagate these uncertainties to the output quantities. Uncertainties in the models themselves due to questionable or plausible alternative assumptions are still not included routinely in PRAs. RG 1.174, Revision 1, contains a discussion of these uncertainties but does not provide acceptable methods for handling them. There is a need for specific guidance on what we mean by model uncertainties, how to identify them, and how to quantify or manage them in decisionmaking."

With publication of NUREG-1885, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," the staff has taken a first step forward in providing needed guidance on the nature and treatment of uncertainty within the context of Level 1 PRA. There is still a need for a common approach for dealing with uncertainty throughout the agency.

RES has been actively attacking a number of short-term projects to resolve issues in PRA and to provide PRA support to other offices. The more significant ones include substantial efforts in the PRA standards and development of related guidance; digital I&C PRA, SPAR models, SAPHIRE 8 development, uncertainty analysis methods and guidance, and advanced reactor PRA scoping studies.

Continuing Gaps: We have been urging NRC to develop agency-wide common approaches for the treatment of uncertainties and elicitation of expert opinions, for many years. Individual research projects and regulatory applications continue to be developed and presented to the ACRS that invent "new" approaches for dealing with uncertainties or, worse, ignore them altogether.

The ACRS has discussed the need for an agency-wide common approach for the elicitation of expert opinions in the past. Development of risk information for regulatory decision-making requires often the elicitation of expert opinions. Important examples include the evaluation of the frequencies of LOCAs of various sizes in the context of the efforts to risk-inform 10 CFR 50.46 (NUREG-1829) and the study (jointly done with DOE and EPRI and reviewed by a National Research Council committee) focused on expert opinion elicitation in the assessment of seismic risk (NUREG/CR-6372). Of course, the first studies to formalize the use of expert opinions in nuclear plant risk assessments were conducted as apart of the NUREG-1150 Study. Numerous other examples involving expert opinion utilization can be found within the agency including examples in performance assessments for waste repositories and the ATHEANA model for human reliability assessment. Few of these studies build on previous NRC-sponsored approaches. We continue to urge that the staff combine the best attributes of these studies and develop a systematic methodology for the elicitation and processing of expert opinions that should be used agency-wide

Future Needs. The staff has identified a number of longer-term PRA research needs including the integrated treatment of phenomena, enhanced data, advanced level 2/3 PRA methods development including methods to analyze digital I&C systems (especially software failures) and passive systems. It has been nearly twenty years since NRC sponsored the NUREG-1150 PRAs. We think it is time for an NRC Level 3 PRA to demonstrate advances in PRA methods and suitable approaches for new reactors. Such an undertaking would require a great deal of advanced planning beginning with its scope. Would it include both internal and external accident initiators? Would it include shutdown operations and spent fuel considerations? Would it be confined to existing plants or would advanced light water reactors be included? Would mixed oxide fuel be considered?

Since September 11, 2001, the security of nuclear power plants has been a major area of activity of the agency. Many new requirements have been imposed on licensees regarding both physical and cyber security. While there is no doubt that security has been improved, there is a growing suspicion that many of these requirements are not improving the security of facilities that are already secure enough. This, of course, means that they may constitute unnecessary regulatory burden.

The Commission's PRA Policy Statement of 1995 stated that PRA use should be increased in regulatory matters, where practical within the bounds of the state-of-the –art, to reduce unnecessary conservatisms associated with the current regulatory requirements, regulatory guides, license commitments and staff practices. The ACRS believes that it is time the agency applied these principles to the security arena.

PRAs contain a wealth of information regarding the ways undesirable plant states, such as core damage and large release of radioactivity, can occur. This information is not utilized in formulating security requirements to evaluate their benefit and their impact on safety. The ACRS recommends that RES establishes a research project to explore the possibility of risk-informing security requirements and building on PRAs to create a unified framework for the evaluation of both safety and security. It is likely that security would be improved, with a reduction in added regulatory burden.

Independent Capability: Staff describes its primary role as conducting independent confirmatory research. The ACRS agrees and suggests that the following key elements of the goals identified by the staff and depicted in Figure 7 are required in order to develop and maintain cutting edge capability to conduct independent confirmatory research to ensure that the regulator's perspective on safety is adequately supported in the PRA arena:

- Supporting the reactor oversight and operating experience programs
- Facilitating the implementation of risk-informed regulation
- Expanding PRA infrastructure to encompass new reactor designs

- Supporting continuous advancement in PRA state-of-the-art and state-of-practice

The structure shown in Figure 7 could provide a basis for developing long-term research goals that can provide a constant vision of where the PRA research program should be headed. Continued and expanded international agreements and active domestic MOUs can support the long-term research program and leverage NRC's research budget. The ACRS supports the growing participation in these activities.

PRA issues related to digital I&C and HRA are discussed in Chapters 5 and 8, respectively.

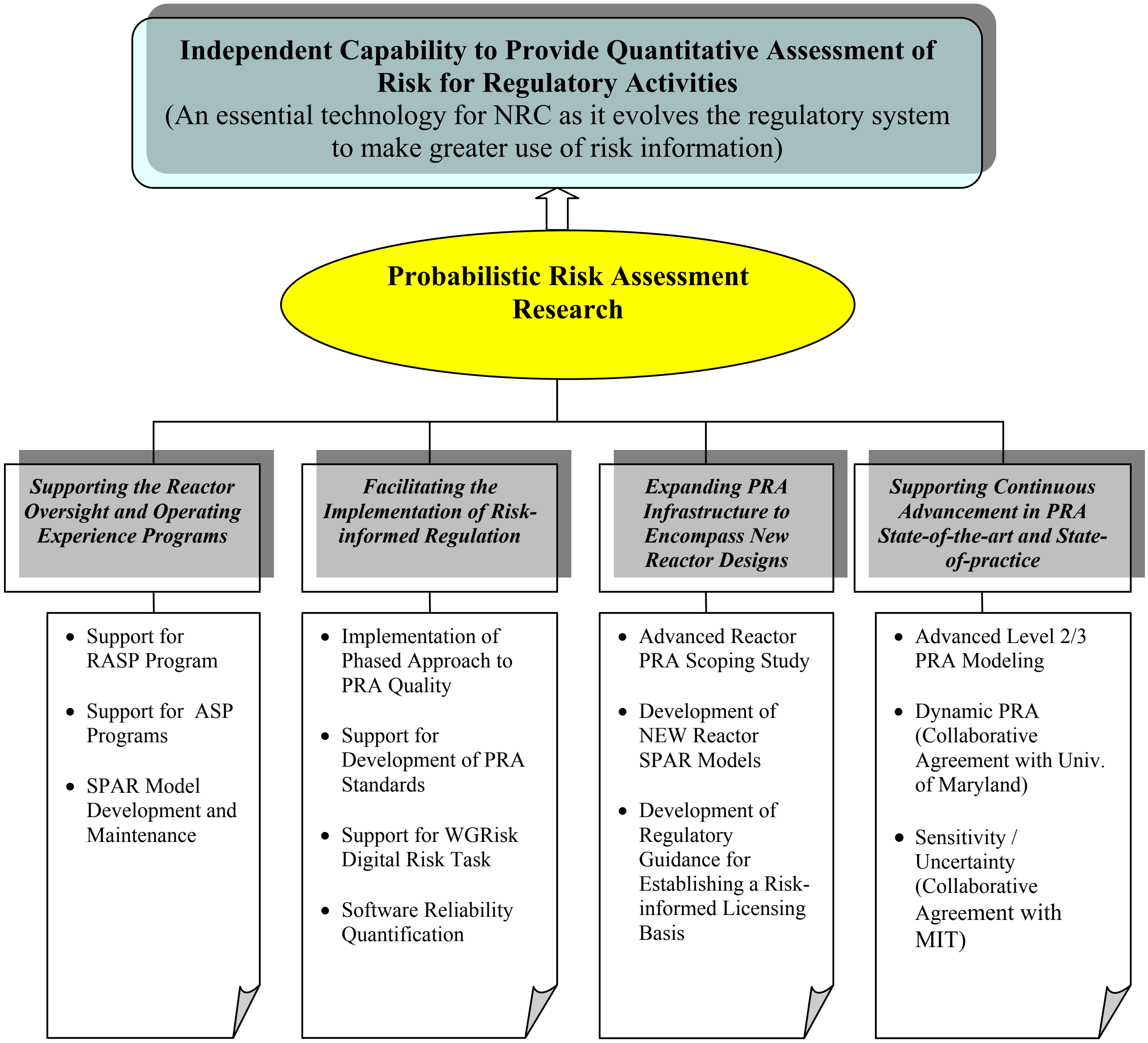


Figure 7. Current NRC Research Activities in PRA

13. RADIATION PROTECTION

Background

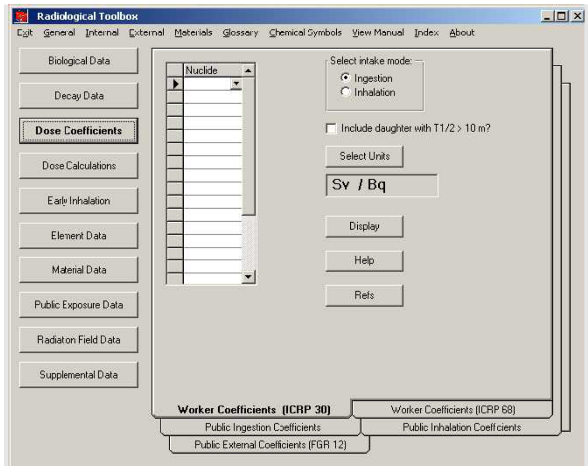
The NRC has maintained a program of research related to radiation protection in the areas of:

- risks from radiation,
- the sciences of internal and external dosimetry, and
- the fate and transport of radioactive materials in the human body and in the environment.

A major thrust of this research is to collect, analyze, and disseminate information on occupational exposures reported to NRC by licensees. This information is used to track the effectiveness of licensees' As Low As Reasonably Achievable (ALARA) programs and will form the basis for future studies to evaluate the health effects of this group of workers. Another important thrust is to develop and maintain tools for assessments related to licensing, siting, environmental performance, and the decontamination and decommissioning of licensed facilities.

Current Research Activities

The current NRC research activities in the area of Radiation Protection are depicted in Figure 8. These activities are leveraged in several ways. First, these activities are based on continued staff participation on national (e.g., The National Council on Radiation Protection and Measurements, NCRP, the National Academy of Sciences, NAS) and international committees (The International Commission on Radiological Protection, ICRP, and Committees of the International Atomic Energy Agency, IAEA). NRC research is also well leveraged by working with a number of collaborating agencies, including the U.S. Environmental Protection Agency.



Radiological Toolbox Graphical User Interface

The staff is engaged in a number of collaborative activities that support the assessment of the behaviors of radioactive material in the environment and ultimate dose consequence analysis.

These are all essential activities and need to be sustained.

<u>Health Effects/Dose Calculation Tools</u>

Health effects/dose calculation tools are used to model and assess the health implications of radioactive exposure and contamination.

VARSKIN**:** The NRC sponsored the development of the VARSKIN code in the 1980s to assist licensees in demonstrating that they have approved radiation protection programs that include established protocols for calculating and documenting the dose attributable to radioactive contamination of the skin. Since that time, the code has been significantly enhanced to simplify data entry and increase efficiency.

Since the release of VARSKIN 3 in 2004, the NRC staff has compared its dose calculations for various energies and at various skin depths, with doses calculated by the Monte Carlo N-Particle Transport Code System (MCNP) developed by Los Alamos National Laboratory (LANL). That comparison indicated that VARSKIN 3 overestimates the dose with increasing photon energy. For that reason, the NRC is sponsoring a further enhancement to replace the existing photon dose algorithm, develop a quality assurance program for the beta dose model, and correct technical issues reported by users.

Radiological Toolbox: The NRC developed the radiological toolbox as a means to quickly access databases needed for radiation protection, shielding, and dosimetry calculations. The toolbox is essentially an electronic handbook with limited computational capabilities beyond those of unit conversion. Further revisions of the toolbox are planned as the need for additional data is identified by NRC staff and other users. The toolbox contains radioactive decay data, biokinetic data, internal and external dose coefficients, elemental composition of a large number of materials, radiation interaction coefficients, kerma coefficients, and other tabular data of interest to the health physicist, radiological engineer, and others working in fields involving radiation. The toolbox includes a means to export the tabular data to an Excel worksheet for use in further calculations. It operates in a Windows environment.

Radionuclide Transport Codes (for License Termination and Decommissioning)

Radionuclide transport codes provide dose analyses in support of license termination and decommissioning:

DandD: DandD is a code for screening analyses for license termination and decommissioning. The DandD software automates the definition and development of the scenarios, exposure pathways, models, mathematical formulations, assumptions, and justifications of parameter selections documented in Volumes 1 and 3 of NUREG/CR-5512.

Probabilistic RESRAD 6.0, RESRAD-BUILD 3.0, and RESRAD-OFFSITE Codes: The existing deterministic RESRAD 6.0 and RESRAD-BUILD 3.0 codes for site-specific modeling applications were adapted by Argonne National Laboratory (ANL) for NRC regulatory applications for probabilistic dose analysis to demonstrate compliance with the NRC's license termination rule (10 CFR Part 20, Subpart E) according to the guidance developed for the Standard Review Plan (SRP) for Decommissioning. The deterministic RESRAD and RESRAD-BUILD codes are part of the family of codes developed by the U.S. Department of Energy. The RESRAD code applies to the cleanup of sites and the RESRAD-BUILD code applies to the cleanup of buildings and structures. Current RESRAD work is focused on RESRAD-OFFSITE and improving the treatment of complex source terms

Cancer Incidence and Mortality Data for Population Living Near NRC-Licensed Nuclear Power Facilities

The objective of this proposed study is to provide the NRC with the latest cancer mortality data for populations living near NRC-licensed nuclear power facilities and to examine the feasibility of studying cancer incidence among these populations. This study will provide the NRC with the most current scientific information available for responding to stakeholder concerns related to cancer mortality rates of populations that live near past, present, and proposed nuclear power facilities.

The Committee supports the staff’s decision to establish an external review committee to review and comment on study method, protocol, and final deliverables.

Assessment and Recommendations

The Committee believes that the staff has developed an appropriate and robust research program in the areas of radiation protection. This program includes radiation protection of workers and radiological assessments related to decommissioning and waste management.

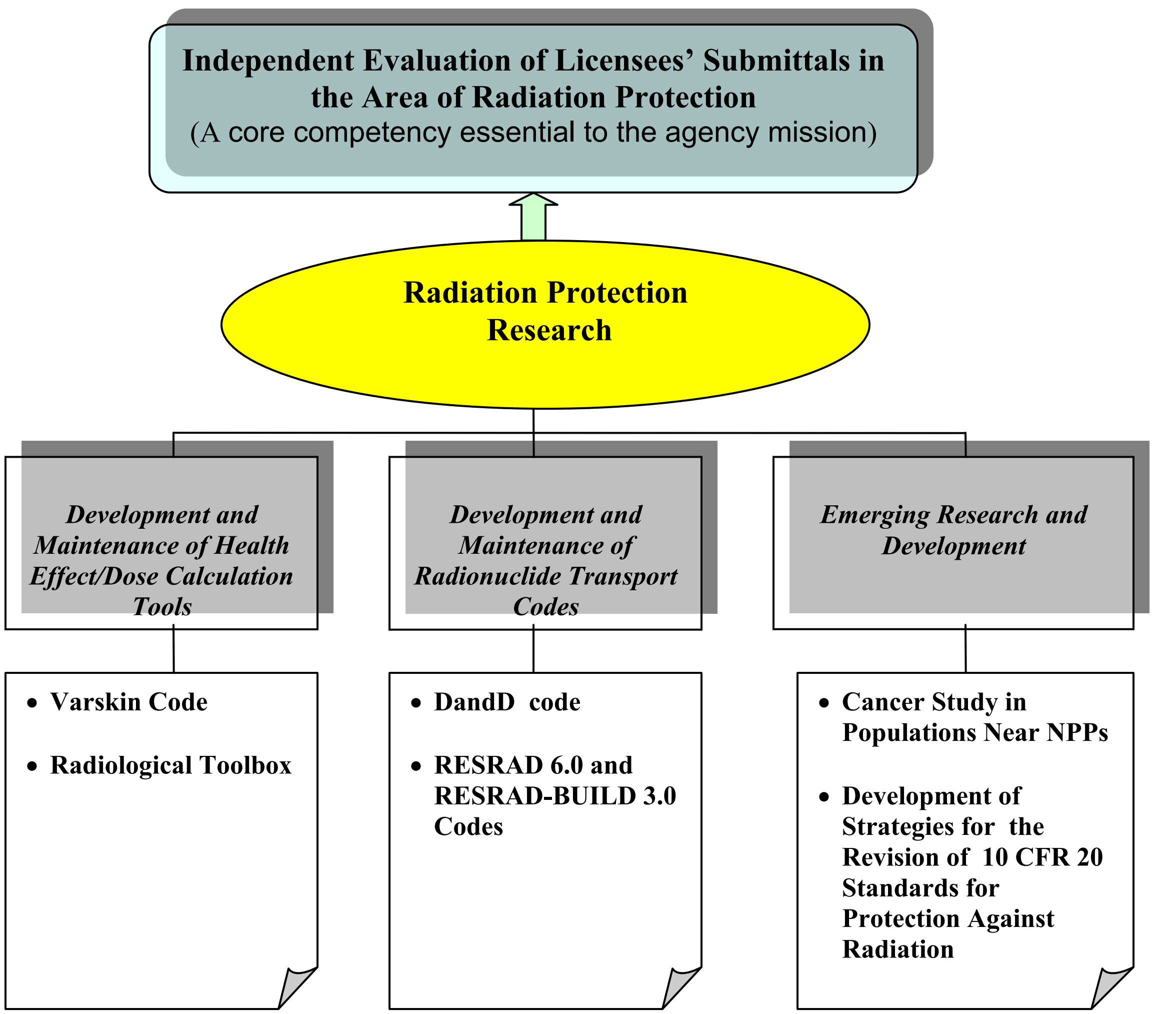


Figure 8. Current NRC Research Activities in Radiation Protection

14. NUCLEAR MATERIALS AND WASTE

Background

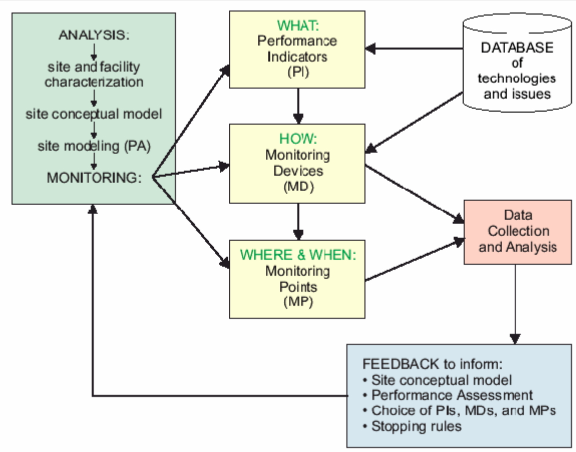
The NRC continues to maintain a robust program of research related to nuclear materials and radioactive waste topics related to licensing, facility siting, facility environmental performance, and the decontamination and decommissioning of licensed facilities.

Current Research Activities

The staff is engaged in a number of research and development projects. These projects cover a wide range of topics from fundamental studies on the fate and transport of radioactive materials in the environment to modernization of computer codes (GALE) used in risk-informed analyses and regulatory decisionmaking. These activities support both specific licensing actions and generic issues. Three projects are summarized below.

Integrated Groundwater Modeling and Monitoring

NRC research is well engaged in a collaborative effort with Pacific Northwest National Laboratory (PNNL) and the U.S. Geological Survey (USGS) regarding uranium sequestration using *in situ* bioremediation. The goal of this effort is to determine the long-term efficacy of the use of bioremediation technologies to capture and sequester uranium *in situ*. Research includes field and laboratory studies of geochemical, hydrogeologic, and microbial processes, and the performance indicators needed to confirm long-term sequestration. Through this collaboration, the NRC has been able to participate in research that is being used by Office of Federal and State Materials and Environmental Management Programs (FSME) in regulatory decision-making regarding pending licensing actions for decommissioning sites, and for evaluating proposed future remediation strategies for *in situ* uranium leach mining.



Flow Chart of the Integrated Monitoring Strategy

GALE Code Updates

Research is coming to closure on two projects to update both the computing platform and parameter values that are used in the GALE codes. Data reflecting current plant operating experience have been added to the code. Further plans are being developed for additional updates to incorporate new reactor information as it becomes available.

Flood Hydrology

Research is underway regarding flood assessments in collaboration with the U.S. Army Corps of Engineers, Bureau of Reclamation (U.S. Department of the Interior), and PNNL regarding analytical tools and guidance on hurricane storm surges, probable maximum precipitation estimates, and flood assessments, respectively. The NRC senior technical staff representative now chairs the Federal Interagency Work Group on Extreme Storm Events [under the Federal Subcommittee on Hydrology (SOH) under the Advisory Committee on Water Information (ACWI)]. Participation in the SOH ensures broad coordination of information among participating Federal agencies on flood hydrology data sources, methods, and guidance. Both the research efforts and interagency activities will provide the technical basis for updating Regulatory Guide 1.59, “Design Basis Floods for Nuclear Power Plants.” Both are contributing information to NRO licensing staff in their hydrology reviews for new nuclear power plants.

Assessment and Recommendations

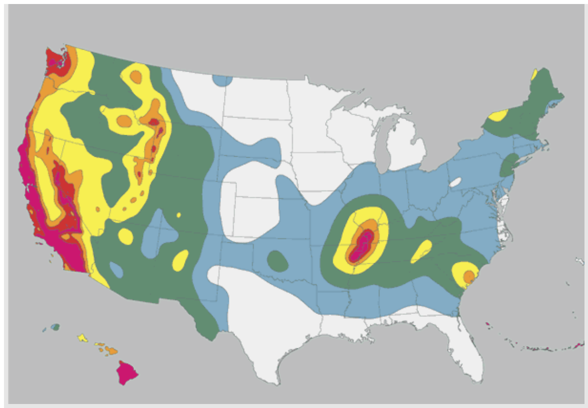
The staff has been particularly successful in leveraging its efforts with those of other federal agencies and other organizations. These collaborations are productive and provide significant synergy with NRC research activities. These collaborative activities should continue and be encouraged. Research has also been effective at producing results that support a range of topics important to regulatory decision making.

15. SEISMIC AND STRUCTURAL ENGINEERING

Seismic events come to mind commonly when discussions of nuclear power plant safety turn to the topic of accident initiation by events external to the plant. Nuclear power plants are designed, of course, to cope with substantial seismic events – up to at least the safe shutdown earthquake (SSE). Earthquakes of greater magnitude have the potential to damage the power plant and barriers to radionuclide release. Earthquake damage can be a common cause mechanism for the failure of methods to prevent accidents and methods to mitigate accidents – extending even to the final line of defense in depth of public evacuation and emergency planning.

The seismicity of the Western US is relatively well known. Research attentions are properly placed on the seismicity of the country east of the Rocky Mountains where events are less common and most of the nation's nuclear power plants are located. There are seismic centers in the Central and Eastern US that have the potential to produce earthquakes that could damage nuclear power plants. The New Madrid, Charleston, Northeast, and Eastern Tennessee seismic centers are examples. Recently, the US Geological Survey has revised expected "return frequencies" for large earthquakes at some of these seismic centers to values higher than used for seismic hazard analysis in the past. Events at these seismic centers can affect large areas since seismic attenuation is thought to be less in the Central and Eastern US than in the Western US.

The US NRC has revitalized its program of research into seismic events and the responses of nuclear power plants to these events. The research program has been developed to support regulatory activities especially in the areas of certification of new plant designs and combined license applications. The seismic research program



Seismicity of the USA

Seismic centers in the Central and Eastern US are of particular interest because the hazard these centers pose are not as well understood as seismic centers in the West and most US nuclear power plants are located in areas affected by these centers.

has elements broadly categorized as:

- seismic hazard characterization,
- earthquake engineering,
- international collaborations in seismic research, and
- development of regulatory guides.

Within these broad categories, the NRC staff has identified 40 activities that need to be pursued to support regulatory processes. The staff has had to prioritize these needs to align with available resources. The staff has done much to solicit input both from within the agency and from the larger technical community involved in seismic research and engineering to assist in this prioritization of its proposed research. The staff has also been diligent in its efforts to establish collaborations with other interested agencies and institutions to leverage resources in pursuit of 17 priority research activities currently underway.

Within the broad research category of "seismic hazard characterization," research is being conducted to:

- characterize the seismicity of the Central and Eastern US, generally, and the East Tennessee seismic zone in particular;
- develop practical procedures for implementation of the guidelines for updating probabilistic seismic hazard assessments – the so-called "SSHAC guidelines" on the use of expert opinion in analysis of seismic issues;
- develop improved models of seismic attenuation in the Central and Eastern US; and
- characterize the threats posed by possible tsunamis to nuclear plants on the Atlantic and Gulf coasts of the US.

Work being done on the implementation of the SSHAC guidelines, while very important for seismic hazard analysis, may also have utility for other agency activities that make extensive use of expert opinion to bound and describe physical phenomena not now completely understood. (See also comments on the issues of expert opinion elicitation made in connection with review of the Probabilistic Risk Assessment research – Chapter 12.)

Work done by the staff in collaboration with the US Geological Survey to characterize tsunami threats to coastal nuclear power plants sites is a notable accomplishment. The threat is probably not negligible and comes primarily from tsunamis initiated by underwater landslides along the continental shelf.

Within the broad category of earthquake engineering, the staff is conducting research on:

- random vibration theory for use in the review of combined license (COL) applications,
- methods for two-dimensional site response analysis, and
- time-domain analysis of soil-structure interactions during seismic events.

These research efforts hold the promise of substantial modernization of staff capabilities for independent review of power plant site applications and changes in plant licensing basis.

The NRC staff is continuing its collaboration with Japan on the large-scale testing of structures, systems, and components during shaking that simulates earthquakes. Further collaborations are taking place in understanding the impacts of the Kashiwazaki-Kariwa earthquakes that exceeded plant design bases in some cases. NRC staff is also engaged with the IAEA on seismic and tsunami threats to nuclear power plants.

Efforts by the NRC staff to revitalize the seismic safety research in support of regulatory activities have been exemplary. There is a well developed research plan that has been broadly reviewed for both technical quality and programmatic impact. Credentials and expertise of those pursuing the research are quite impressive. The research is having impact on the regulatory processes and is expected to have even greater impact in the future. The active research involving broad collaborations with the larger seismic community is assuring the agency has at hand expertise to help resolve novel issues that arise as new plant designs are submitted for certification and new license applications are submitted for staff review.

16. SEVERE ACCIDENTS AND SOURCE TERM

Background

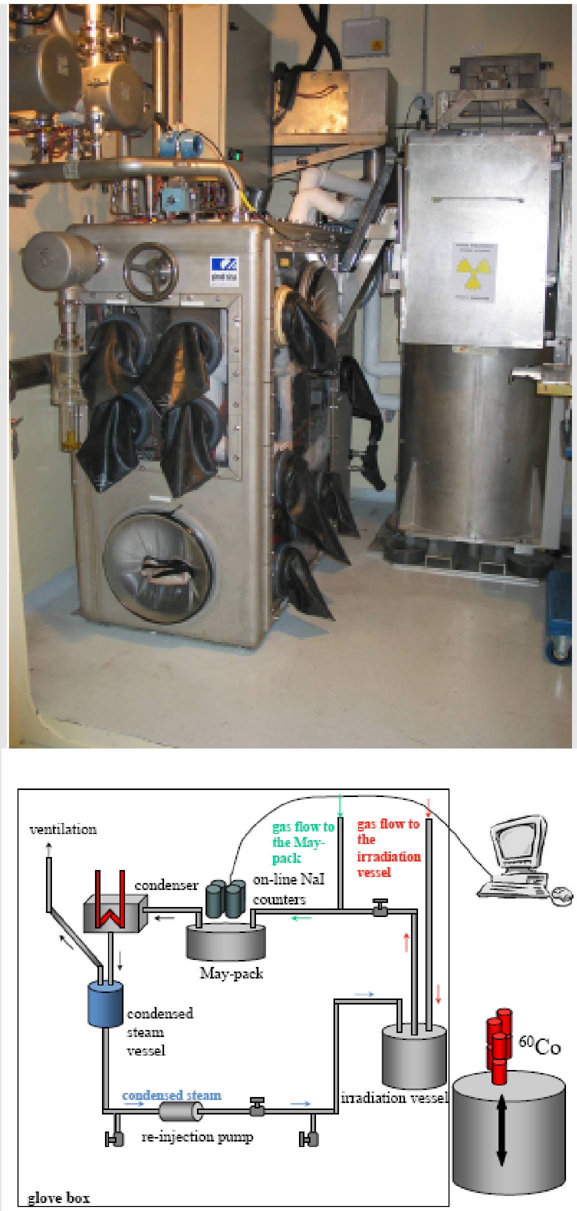
Significant research was undertaken by NRC following the accident at the Three Mile Island (TMI) Unit 2. Over the last three decades, major experimental and model development studies provided an improved understanding of progression of severe accidents and the radiological consequences of such accidents. As a result of these studies, the system-level computer code, MELCOR, has emerged as an integrated tool for predicting severe accident progression and source term magnitude. The NRC requires such expertise and analysis capabilities to help support regulatory decisions for operating nuclear power plants as well as advanced nuclear plant designs. The tools help the staff in its transition to a more risk-informed regulatory framework.

Support of regulatory needs implies a level of understanding of severe reactor accidents to assure adequate protection to the health and safety of the public. The NRC made major investments in severe accident research to achieve the needed understanding. Once this immediate need was met, the NRC adjusted its investments in severe accident research to required levels for on-going analysis and risk-informed activities.

Current Research Activities

The current NRC research activities in severe accident and source term are depicted in Figure 9. Severe accident research activities are focused on development and usage of MELCOR code. Much of the research includes international collaborative severe accident experimental programs.

Severe accident research is continuing internationally, with on-going programs in the Pacific Rim (Japan and Korea) as well as



EPICUR Facility for the study of radioactive iodine behavior in reactor containments under accident conditions

Europe (France, Switzerland, and Germany).The NRC has an effective strategy to maintain its leading position in severe accident analytical capabilities as well as to maintain its competence in understanding key accident phenomena, by collaborations in international research programs. The body of knowledge coming from NRC's past experimental work and current ongoing international efforts are systematically incorporated into the MELCOR accident analysis code. The NRC Cooperative Severe Accident Research Program (CSARP) is an annual forum used for the exchange of severe accident findings, both experimental and analytical. One significant outcome of this effort is the adoption of the MELCOR code by many countries and institutions as the key analytical tool for severe accident analyses.

MELCOR Code Development and Usage

The MELCOR code is a fully integrated, engineering-level computer code whose primary purpose is to model the progression of accidents in current nuclear power plants, advanced reactors and more recently non-reactor systems (e.g., spent fuel pool behavior). The MELCOR code was first developed as a PRA tool and thus, modeled the reactor core, coolant system and safety systems as well as containment systems in a less detailed manner than more mechanistic thermal-hydraulic and fuel rod models. A major current activity is to consolidate the physical models and capabilities of more detailed severe accident codes into MELCOR. This effort will provide an efficient state-of-the-art code for severe accident analyses. In addition, the MELCOR code has been adopted by a number of international nuclear safety organizations (e.g., IBRAE-Russia, PSI – Switzerland, IRSN – France) and universities (e.g., University of Michigan, University of Texas A&M, and University of Wisconsin) for use in severe accident analysis.

The MELCOR code and its fission product source term transport and consequences model (MACCS) are being used as an integral parts of the SOARCA program. MACCS is a widely accepted tool used for consequence analysis, and its continued development and support is important to the agency for both safety and security regulatory applications. More recently, the NRC staff is developing a Graphic-User-Interface as part of the long-term development strategy for MELCOR. Such an advance will allow NRO and NRR staff to routinely do safety analyses for specific plants (e.g., source term analyses as in Regulatory Guide 1.183 for control room habitability and license amendments). Such an advance will allow staff to do analyses themselves on their desktop. The ACRS is supportive of these efforts.

Collaborative Severe Accident Experimental Programs

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

***ARTIST*:** This program involves experimental work underway in Switzerland at PSI. The original test objective was to measure the aerosol removal on the secondary side of steam generators during accidents in a PWR, that involved containment bypass due to spontaneous or induced steam generator tube failure. Such bypass accidents are often risk dominant for PWRs. The risk ascribed to such accidents may stem from the conservatism in the aerosol decontamination assumed in accident models.

ARTIST tests of aerosol transport through secondary sides of steam generators is complete and has shown that overall retention is low (DF ~6) but dependent on particle size. ARTIST-II experiments are examining the issues of particle bounce and breakup discovered in doing the ARTIST work and not now modeled in the MELCOR computer code.

***OECD-BIP*:** The OECD/NEA Behavior of Iodine Project (BIP) has been initiated to provide separate effects and modeling studies of iodine behavior in a containment following a severe accident. As a part of this project the results of three Radioiodine Test Facility (RTF) experiments will be provided by Atomic Energy of Canada Limited (AECL). These test data will be used for validation of improved models of radioactive iodine now being developed for the MELCOR computer code

OECD-FCI SERENA**:** The OECD is sponsoring the SERENA (Steam Explosion Resolution for Nuclear Applications) Phase II program, which evaluates the capability of existing codes to predict if fuel-coolant interactions could lead to threatening, dynamic pressure loads in the reactor vessel or containment cavity. Experiments are being supported and performed at CEA (KROTOS) and KAERI (TROI) as part of the SERENA II program. Steam explosion models are then used by other international participants to perform analyses of KROTOS and TROI experiments. One specific effort by the OECD deserves note; i.e., a web-accessible database that documents steam explosion experiments and associated publications in the public domain.

OECD-MCCI**:** The OECD-MCCI program is an international collaborative experimental study being conducted at the Argonne National Laboratory. The original focus was on investigating the heat transfer to an overlying water layer to cool molten core materials interacting with concrete and its viability for long-term coolability in currently operating reactors. More recently, the program has been addressing the long-term coolability in advanced light water reactor designs using core retention devices; e.g., the Evolutionary Power Reactor (EPR). The MCCI project has entered a second phase of experimentation and consists of two experimental efforts: Small-scale Water Ingression and Crust Strength Tests (SSWICS) and integral Core Concrete Interaction Tests (CCI). The SSWICS tests focus on quantifying the heat loss to an overlying water layer by mechanisms other than conduction limited heat transfer, and to measure the strength of the crust formed during flooding of the melt. The CCI tests are focused on resolving uncertainties in axial versus lateral power splits and respective concrete ablation rates. Most recently, the CCI integral tests are examining the melt behavior with underlying cooled refractory basemat, similar to the core retention concept in the EPR. Test results will be used to update the capabilities of the MELCOR computer code.

PHEBUS-FP**:** Phebus-FP has completed its major, in-pile experiments and is preparing the final report on the last test. The follow-on program, Phebus-ISTP, is a collection of separate effects projects to pursue specific aspects of Phebus-FP findings. The separate effects tests are:

- **EPICUR:** chemistry of iodine in the containment
- **CHIP:** fission product chemistry in the reactor coolant system
- **MOZART:** air oxidation of fuel cladding
- **BECARRE:** interaction of steel and cladding with boron carbide in steam
- **VERDON:** fission product release from MOX and high burnup fuel in steam and air

The EPICUR project is making progress and has resolved some issues. All the technical issues will not be completely resolved by the conclusion of the effort and it is likely that some follow-on will be needed. The CHIP has encountered some technical issues and it is not clear that all planned experiments will be done by the conclusion of the project next year. The MOZART project is completed and appears to have yielded high quality kinetic data. The BECARRE project is entering its last phase with larger scale tests underway. It has yielded useful data for modeling the degradation of boron carbide control rods that may be used in future PWRs. The VERDON project is a large undertaking to redevelop the VERCORS facility and its capability to study fission product release from irradiated fuel pellets under controlled temperature and gas composition conditions. This facility will not be ready at the end of the project, but the work should continue and some mechanism for providing NRC with data is being developed.

Assessment and Recommendations

The ACRS supports the strategy that the NRC staff has developed to support regulatory decisions for severe accidents via computer code development and experimental data analysis and evaluation. This approach can successfully maintain and update its modeling capabilities for severe accident analyses. The planned program extensions and continuations of these collaborations are well worth the investment.

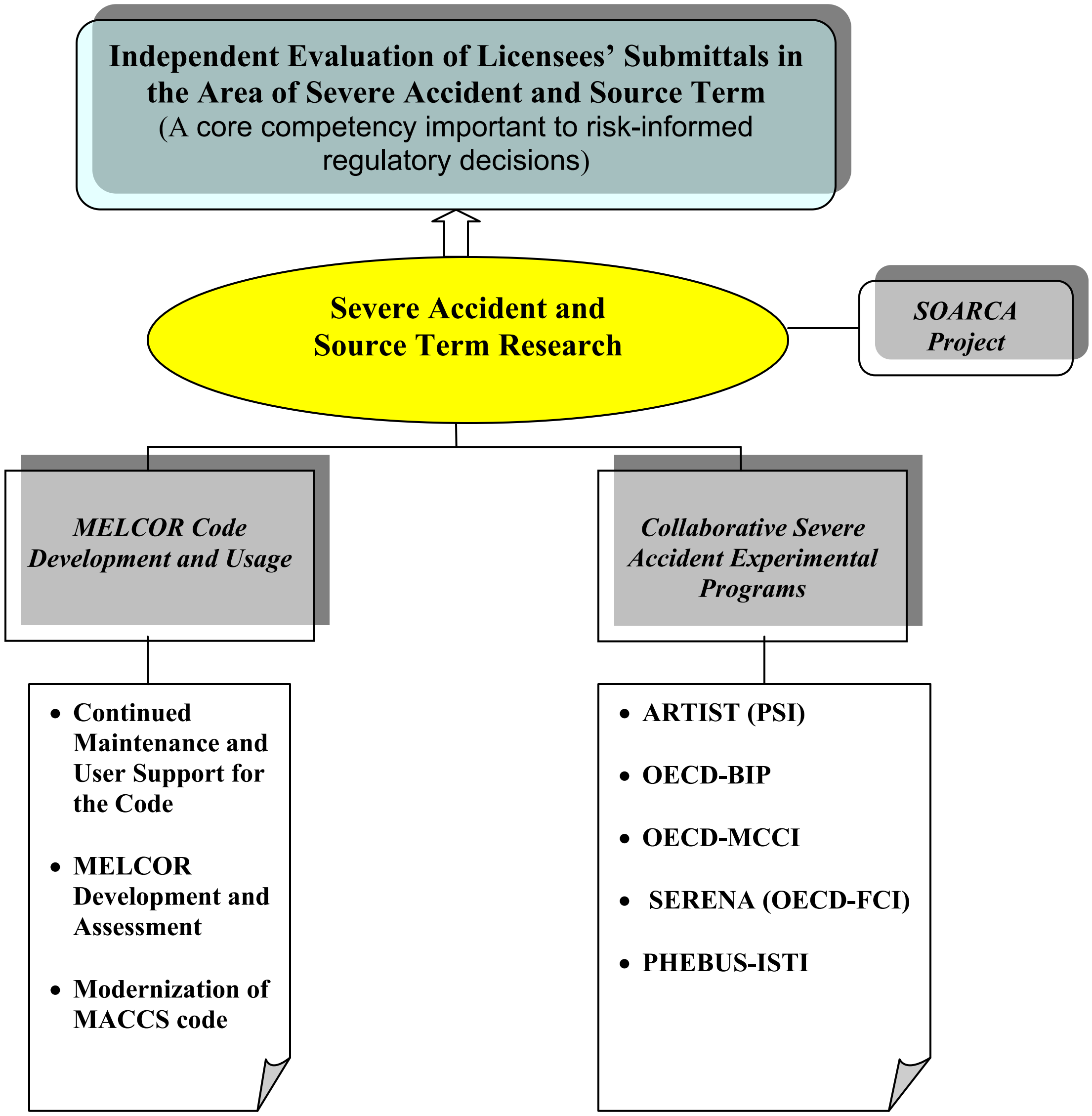


Figure 9. Current NRC Research Activities in Severe Accident and Source Term

17. THERMAL HYDRAULICS

Background

Evaluation of the effects of thermal hydraulic phenomena on nuclear safety has always been a central element in the conduct of NRC's regulatory mission. Of particular importance has been and continues to be the capability to independently confirm thermal-hydraulic analyses in licensees' submittals.

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

Upper View of the Primärkreislauf-Versuchsanlage (primary coolant loop test facility) PKL Test Facility in Erlengen, Germany

Under the auspices of the OECD-NEA, the NRC has joined 13 other international organizations to conduct thermal-hydraulic experiments in the PKL facility. PKL is an integral test facility simulating a typical western-type pressurized water reactor (PWR) and is used for the investigation of the thermal hydraulic system behavior under accident conditions.

Current Research Activities

The current NRC research activities in thermal-hydraulics are depicted in Figure 10 The research activities focus on the development of the TRACE computer code for confirmatory analyses of a wide range of safety-significant thermal-hydraulic phenomena, and supporting experiments. There is also a modest effort to develop capability in multi-dimensional computational fluid dynamics (CFD).

<u>TRACE Computer Code Development and Validation</u>

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

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

TRACE is intended to serve as the main tool for the confirmatory analyses of a broad range of thermal-hydraulic problems for current and new reactor designs. It has the potential to offer significantly enhanced capabilities for state-of-the-art analyses of thermal-hydraulic issues. Several important technical issues, such as core stability and ATWS behavior, involve coupling between neutronics and thermal hydraulics and require that TRACE be properly coupled to a neutronics code like PARCS, an activity that is completed now. The integration, validation, and testing of the PARCS-TRACE coupled code is currently under way so that it can be reliably used for confirmatory analyses. TRACE also has the capability to interface with the CONTAIN code for containment response analysis as well as with other computational tools, including MATHCAD.

In response to ACRS recommendations, the staff commissioned and completed a detailed peer review of TRACE. The main objective of the TRACE code peer review was to identify the strengths and deficiencies of the code, and provide recommendations for code changes and improvements. The peer review found no major deficiencies that would introduce significant errors or preclude the use of TRACE for analysis of postulated LOCAs in current LWRs. Certain improvements were recommended for treatment of the momentum equations, which could be particularly important for passively cooled systems, though TRACE was not explicitly reviewed for applicability to such systems.

Thermal-hydraulic codes, including TRACE, solve an intertwined structure of approximate conservation equations and empirical correlations. The uncertainties and biases inevitably introduced by such empirical procedures need to be properly addressed. Because of these uncertainties, predictions of such codes are adequately accurate only within certain ranges of parameters. The codes cannot be given blanket approval for all situations to which they might be applied. In practice, a code, such as TRACE, must be qualified by assessment against a range of data that cover the phenomena that dominate the prediction of figures of merit, such as peak clad temperature, important to the regulatory process. These dominating phenomena change with the reactor systems and accident conditions being considered. In view of this, thermal-hydraulic codes need to be assessed for analyses of a specific accident in a particular system.

RES has initiated a systematic assessment of the applicability of TRACE to analyze new reactor designs. Work on a detailed assessment of the applicability of TRACE to analyze ESBWR LOCAs, focusing on the collapsed liquid level in the reactor pressure vessel as the primary figure of merit, has already been completed. Ongoing work on assessment of the applicability of TRACE for confirmatory analyses of safety-significant thermal-hydraulic phenomena in the US-Advanced Pressurized Water Reactor (APWR) and EPR designs should be completed in a timely fashion to allow applications in the design certification process.

RES has also been participating in two cooperative international agreements to obtain high-quality experimental data to refine best-estimate calculations. The first cooperative agreement, which has recently concluded, was the NUPEC BWR Full-size Fine-Mesh Bundle Tests (BFBT) benchmark. Participation in this activity allowed the NRC to obtain a database of sub-channel void fractions, pressure drops, and critical power measurements from a representative BWR fuel assembly from the Nuclear Power Engineering Corporation (NUPEC) in Japan. The extensive database and contributions of the benchmark participants will be used to improve the predictive capabilities of TRACE.

A second international cooperative agreement, centered on a PWR database, has been recently put into place. The NUPEC PWR Sub-channel and Bundle Test (PSBT) benchmark will use a high-quality database of PWR void fraction and DNB data provided by METI/JNES. This database will be used to assess TRACE code models. Like the BFBT benchmark, the PSBT agreement will involve the participation of international experts who will take part in workshops intended to facilitate open discussion of the database and the modeling techniques used in other thermal-hydraulic codes. Under this agreement, METI/JNES will provide the experimental data, OECD/NEA will be in charge of the logistics of the event, and NRC/PSU NEP will prepare the benchmark specifications and coordinate the workshops.

Multi-Application Small Light Water Reactor (MASLWR) is a system-level test facility constructed by Oregon State University (OSU) to examine thermal hydraulic phenomena of importance to integral type reactors. MASLWR is the predecessor to the current NuScale integral reactor with 1:3 length-scale, 1:254 volume scale, and a 1:1 time scale and is operated under full pressure and temperature conditions of the NuScale design. Currently, NuScale is working with OSU to update this integral test facility to the latest NuScale module design in preparation for design certification testing. Through participation in the IAEA International Collaborative Standard Problem (ICSP), a TRACE model for MASLWR will be developed. Predictions of this model will be compared for several steady state and transient reactor states. Test results can also be used to validate new analysis models or components for the TRACE code. The research effort aims towards the future support of licensing reviews of the NuScale reactor.

<u>Experimental Studies of Thermal-hydraulic Phenomena</u>

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

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

The Purdue University Multidimensional Integral Test Assembly (PUMA) is a medium size reduced height scaled facility and has been used in the past to perform integral LOCA tests of interest for the ESBWR design. Tests are being conducted at the PUMA facility to obtain experimental data on the void fraction distribution and fluid dynamics of a BWR suppression pool during the blowdown period. The results of these tests will be used to support the resolution of Generic Safety Issue 193, “BWR ECCS Suction Concerns.”

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

The Rode Bundle Heat transfer (RBHT) facility at Penn State University was developed to address issues related to emergency core cooling, including the development of a better understanding of reflood and rewetting in realistic, bundled geometries. Currently, this facility is being used to conduct oscillating reflood tests to determine the effect of the inlet flow rate, magnitude, and frequency on peak clad temperature (PCT). There is also an option to perform steam cooling with droplet injection tests to determine the effect of dry spacer grid on droplet size and distribution.

In parallel, the NRC is collaborating with international groups in undertaking experiments in facilities abroad as noted below:

OECD/NEA ROSA-2: NRC is participating in the OECD/NEA ROSA-2 Project to utilize the Large Scale Test Facility (LSTF) of ROSA (Rig-of-Safety Assessment) Program of JAEA (Japan Atomic Energy Agency) for studying the integral response of the core and steam generator. The full-height ROSA/LSTF integral test facility, with 1/48 volumetric scaling, is designed to investigate thermal hydraulic phenomena of interest to PWRs. The ROSA-2 Project will provide both integral and separate-effects thermal-hydraulic data on intermediate break LOCA and on the recovery from Steam Generator Tube Rupture (SGTR) events. These data will be used to assess predictive capability of thermal-hydraulic analysis codes including TRACE.

OECD/NEA-PKL-2: The OECD/NEA-PKL2 Program is designed to address thermal-hydraulic safety issues for current PWR and new PWR designs.. The PKL facility is a full-height, 1:145 power and volume scaled replica of a 4-loop, 1300 MW PWR. The experiments will focus on steam generator heat transfer under shutdown conditions (e.g., loss of RHR during mid-loop operations), fast cooldown transients (such as main steam line breaks), accident situations for new PWR designs (e.g., EPR Emergency Operating Procedures), and boron precipitation. The data obtained from these experiments will be used for TRACE assessment, to complement design certification reviews, where possible, and in the case of boron precipitation, to validate licensee post-LOCA long-term cooling strategies.

ISP-50, ATLAS 50% DVI Line Break: ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation) is a 1/2-height and length, 1/288-volume, and a full pressure integral test facility in Korea. The ATLAS facility will be used to simulate a direct vessel injection (DVI) line break, which is important in several new PWR designs such as APWR and AP1000. Participants in the International Standard Problem 50 (ISP-50) will involve "blind" and open calculations for the test using various safety analysis codes including ATHLET, CATHARE, MARS, RELAP5/MOD3, as well as TRACE. Considering the very limited integral test data on DVI line breaks, this will provide the participants with the opportunity to understand the relevant thermal-hydraulic behavior and to assess existing safety analysis codes against the data. The effort will be coordinated by the Korea Atomic Energy Research Institute (KAERI).

Development of Multidimensional CFD Capabilities

The NRC currently has only a modest effort in the area of CFD and this is limited to using commercial CFD codes from ANSYS Inc. (FLUENT) and CD-adapco (STAR-CCM+).

Assessment and Recommendations

The staff is to be commended for the progress that has been made in developing and moving forward with incorporation of TRACE into the regulatory process. Much work remains to be done to enable its reliable use for the analysis of the new LWR designs, an urgent matter which should be conducted with high priority. The priorities for further development of TRACE require careful evaluation.

The international collaborative efforts are also to be commended, as they take advantage of facilities that are of a scale and capability that do not currently exist in the U.S. Furthermore, they draw on the expertise of international partners, who have continued to maintain a very high level of capability in the thermal-hydraulics field. However, complementary development of national facilities to address safety related thermal-hydraulics issues should be seriously considered. Such facilities would enable the retention of U. S. expertise and provide capability to conduct experiments for supporting confirmatory thermal-hydraulic analyses of new reactor designs.

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

It is inevitable that the licensees will increasingly capitalize on the extraordinary advances in computing power and computational science to resolve problems which the current generations of thermal-hydraulic codes such as TRACE are unable to do. A recent LWR Sustainability Research and Development (R&D) Program, prepared by DOE in close collaboration with industry R&D programs, identified examples of advanced features anticipated in the next-generation system analysis code, including:

- CFD-based Coarse-Grain Modeling for turbulent and two-phase flow;
- fully implicit, nonlinear (tightly) coupled multi-physics, using Jacobian-Free Newton-Krylov methods;
- high order, accurate, computationally effective and robust numerical solvers;
- parallel, high-performance computing;
- optimized operation for multi-processors (100-1000 CPU clusters);
- adaptive model refinement; and
- built-in sensitivity analysis and uncertainty quantification machinery, to provide guidance for model refinement and further research ("a quantitative PIRT process")

LWR Sustainability R&D Program considers features such as a next-generation system analysis code as essential components to improve understanding and utilization of safety margins.

NRC thermal-hydraulic research has to position the agency to address the research coming from the LWR Sustainability Program. Several possibilities for developing such capabilities should be evaluated taking budgetary constraints into account. First amongst these would be participation, and cost-sharing, in programs with international partners to develop next-generation multidimensional CFD simulation tools aiming at a high level of transparency, verification, and validation. Second, perhaps in conjunction with the first, consideration should be given to the formation of an NRC-US university consortium to develop such capability capitalizing on the high level of CFD expertise that now exists in several universities. Third, and perhaps again in conjunction of some subset of the other options, the possibility of building on one of the excellent open-source CFD platforms should be seriously considered.

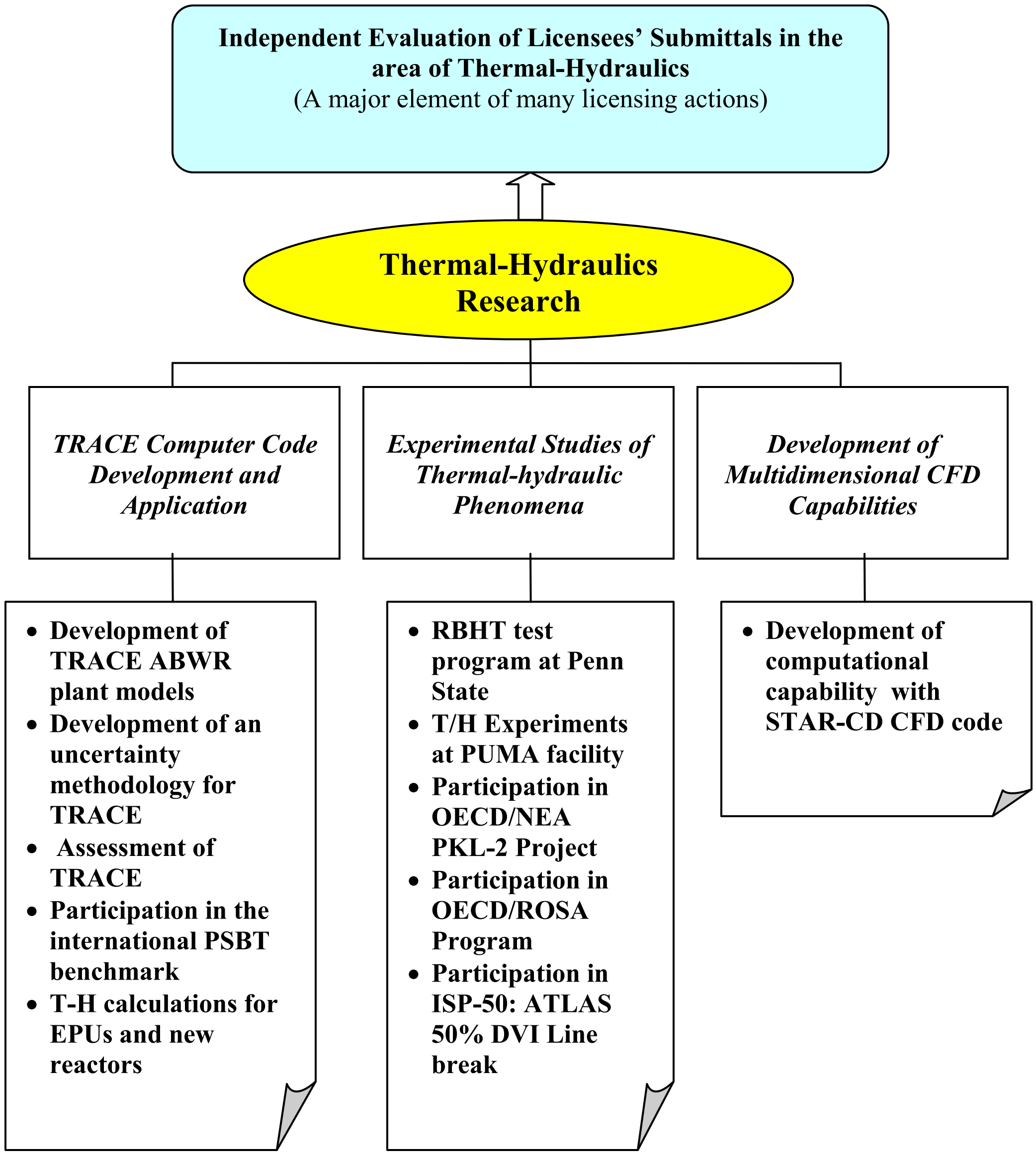


Figure 10. Current NRC Research Activities in Thermal-Hydraulics Research

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

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

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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