NUREG-1635, Vol. 4

Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program

A Report to the U. S. Nuclear Regulatory Commission

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

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at www.nrc.gov/NRC/ADAMS/index.html.
Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402–0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: Office of the Chief Information Officer,
Reproduction and Distribution
Services Section
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: DISTRIBUTION@nrc.gov
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address www.nrc.gov/NRC/NUREGS/indexnum.html are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG–0750).

NUREG-1635, Vol. 4

Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program

A Report to the U. S. Nuclear Regulatory Commission

Manuscript Completed: April 2001
Date Published: May 2001

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

CONTENTS

LIST OF FIGURES

LIST OF TABLES

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AEOD	Analysis and Evaluation of Operational Data
ANO-2	Arkansas Nuclear One, Unit 2
ANSI	American National Standards Institute
APEX	Advanced Plant Experiment
ASCE	American Society of Civil Engineers
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
BEIR VII	Biological Effects of Ionizing Radiation
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CFD	Computational fluid dynamics
CDF	Core damage frequency
CFAST	Consolidated fire and smoke transport
CFR	*Code of Federal Regulations*
COMPBRN	Computer Code for Modeling Compartment Fires
COTS	Commercial Off-the-Shelf
CRSS	Center for Risk Studies and Safety
CSNI	Committee on the Safety of Nuclear Installations
DBA	Design basis accident
DPO	Differing Professional Opinion
EMI	Electromagnetic Interference
FIVE	Five Induced Vulnerability Evaluation
FRAPCON	Fuel Rod Analysis Package-Gap Conductivity
FRAPTRAN	Fuel Rod Analysis Package-Transient
GSI	Generic safety issue
GV	Garner Valley
GWD/t	Gigawatt day/ton
HSSI	Heavy-section steel irradiation
IAEA	International Atomic Energy Agency
ICRP	International Committee on Radiation Protection
I&C	Instrumentation and control
IGR	Inert gas receiving
IPE	Individual plant examination
IPEEE	Individual plant examination of external events
INPO	Institute of Nuclear Power Operations

JCCRER	Joint Coordinating Committee on Radiation Effects Research
LERF	Large, early release frequency
LET	Linear energy transfer
LOCA	Loss of coolant accident
LOFT	Loss-of-fluid test
LNT	Linear no-threshold
LWR	Light water reactor
MAAP	Material access authorization program
MACCS	MELCOR Accident Consequence Code System
MC	Master curve
MITI	Ministry of International Trade and Industry
MOX	Mixed-oxide fuel
MOV	Motor operated valves
MRP	Modifications/Rework Package
NDE	Non-destructive examination
NEA	Nuclear Energy Agency
NIST	National Institute of Standards and Technology
NMSS	Office of Nuclear Material Safety and Safeguards (NRC)
NOAA	National Oceanic and Atmospheric Administration
NPP	Nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation (NRR)
NUPEC	Nuclear Power Engineering Corporation
ODSCC	Outside-diameter stress-corrosion cracking
OECD	Organization for Economic Cooperation and Development
OERAB	Operating Experience Risk Analysis Branch
OSHA	Occupational Safety and Health Administration
PRA	Probabilistic risk assessment
PTS	Pressurized thermal shock
PUMA	Purdue University Multidimensional Integral Test Assembly
Pu	Plutonium
PWR	Pressurized water reactor
RELAP5	Reactor Leak and Analysis Program
RES	Office of Nuclear Regulatory Research (NRC)
RFI	Radio frequency interference
RHR	Residual heat removal
RPV	Reactor pressure vessel
SAPHIRE	Systems Analysis Program for Hands-On Integral
SPAR	Simplified plant analysis risk
SSC	Structure, system and component
TRAC-BWR	Transient Reactor Analysis Code—Boiling Water Reactor
UCSB	University of California, Santa Barbara

I. INTRODUCTION

I.1 . Objectives

This report to the U.S. Nuclear Regulatory Commission (NRC) presents the findings and recommendations that emerged from an assessment conducted from March 2000 through March 2001 by the NRC's Advisory Committee on Reactor Safeguards (ACRS).

In conducting this assessment, the ACRS sought to fulfill the following objectives:

- Assess the ongoing and planned reactor safety research being carried out by the Office of Nuclear Regulatory Research (RES).
- Recommend additional research that will be needed to promote the efficient and effective execution of the agency's mission in the future.

The attentions of the report focuses on the research addressing issues of power reactor safety. Research dealing with the permanent disposal of radioactive waste in geological repositories will be discussed separately in a report by the NRC's Advisory Committee on Nuclear Waste (ACNW).

I.2 Approach

The ACRS assessment of the research programs focused on the following questions, rather than on the initial need for the research results:

- Has the research sufficiently progressed to a point at which its results are adequate to support regulatory decisionmaking?
- Does the research need to be expanded or otherwise modified to better meet the spectrum of agency needs?
- Does this research need to be performed independently by the NRC, rather than depending on information supplied by the licensee or nuclear energy institutions?

In addressing these questions, the ACRS recognized that the agency's needs go beyond the immediate, identified regulatory decisions that might motivate research. Foremost among these are the need to maintain the technical competencies of the NRC staff, and to leverage the agency's participation in collaborative international research. The recent experience with safety issues associated with high burnup reactor fuels has demonstrated that certain technical competencies are crucial to the NRC, although they may not be continually used by the agency's line organizations and are difficult to acquire by contract. The difficulty can arise because there are few contractors who possess such episodically used expertise, or because the expertise that can be obtained is not sufficiently independent of licensees to meet the NRC's needs. When these circumstances arise, it is necessary for the NRC to maintain the competency perhaps by sponsoring research in the given technical area. The ACRS believes that there are very few areas that require this type of resource commitment by the NRC to maintain competencies; however, competency in reactor fuels is certainly such one area. It would be useful if RES explicitly identified any others. Unfortunately, past efforts to identify such needs have used criteria that were too broad. The selection criteria should be limited to crucial importance to the NRC mission and unavailability outside the agency. The time required for outside experts to "spin up" on the peculiarities of regulatory processes and issues ought not be a criterion, since the agency's current staff is quite capable of providing this context for any technical issue.

Reactor safety research is becoming a collaborative enterprise, with many countries joining together to carry out high-cost research initiatives, particularly in the area of experimental research. In order to maintain access to ongoing or planned collaborative projects that address the agency's critical research needs, it may well be necessary for RES to participate in other collaborative research projects that do not directly apply to the agency's needs.

I.3. Research Needs and Focus

The value of NRC-sponsored research in the past has been discussed at some length in previous communications from the ACRS to the Commission,[1,2,3] and need not be reiterated here. It is noteworthy, however, that many of the highest-impact results from NRC research have come from focused projects that utilized substantial fractions of the agency's research capabilities. Such major projects included the initial development of probabilistic risk assessment (PRA) techniques for reactor safety (WASH-1400), the assessment of risk at representative plants (NUREG-1150), and the research on the severe accident source terms that led to the revised accident source term (NUREG-1465).

The great majority of research activities now being sponsored by RES have been motivated by "user need" requests from the NRC's line organizations. The ACRS still has concerns about the user need process. As now constituted, it is not evident that this process provides a thorough indication of research that could benefit the agency's line organizations. The ACRS remains concerned that the current user need process leads to a nonuniformity in the research support for improving line organization capabilities. RES should have an understanding of the needs of line organizations and the opportunities to improve regulatory processes that is as complete as possible to begin the research prioritization effort. As part of its ongoing activities, RES should have its own mechanisms for identifying needs and opportunities that would benefit from research. The identifications that RES makes should enter into the hierarchy of possible research activities with equivalent standing to the identifications made in the user need requests.

Nevertheless, motivation of the research by a user need request provides prima facie evidence that the information sought in the research is needed for the effective and efficient execution of the NRC's missions by its line organizations.

The ACRS foresees continuing needs for NRC research, and disagrees with those claiming that nuclear power generation and its safety regulation have been sufficiently established such that no new safety issues requiring research will arise. Indeed, one has only to look at the substantial changes taking place in the industry, as well as the changes within the NRC to forecast that research capabilities will still be needed. Within the industry, the important changes include power uprates, extended fuel burnup, "best estimate" safety analyses, licensing renewal, and shortened outages with greater use of online inspection and maintenance. Within the regulatory agency, the major change is the move to greater use of risk information in regulatory processes. This transformation of nuclear power reactor regulation has yielded processes for risk-informed changes to licensing bases, risk-informed oversight and monitoring, risk-informed enforcement of regulations, and new efforts to make the regulations themselves risk informed. The continuation of these changes in the regulatory processes, as well as the need to respond effectively and efficiently to changes in the nuclear industry, are very likely to require research to provide information and methods that are not currently available.

The ACRS does see, however, qualitative changes in the focus of NRC research. In the past, much of the research has been on discovering, characterizing, and redressing vulnerabilities of nuclear power plants that might adversely affect the health and safety of the public. Certainly, the development of risk assessment methods was initiated explicitly for this purpose. Much of the research on severe reactor accidents, containment integrity, reactor vessel integrity, fire protection, and accident source terms was initiated in response to suspected vulnerabilities. The success of this research is largely demonstrated by the fact that we now believe that it is possible to quantitatively evaluate the risks that are posed by nuclear power plants in many circumstances.

Although the identification and resolution of safety vulnerabilities should remain as the highest priority, the focus of research should be evolving more in the direction of developing improved methods and databases for the efficient and effective execution of the NRC mission. Because some research on safety vulnerabilities is not as complete as might be desired, there is today an expected tension and competition for resources between research with a focus on vulnerabilities and research directed toward improving and modernizing the capabilities of NRC line organizations. Such tension and competition becomes apparent in the discussions of both the ongoing research programs and the definitions of future research for the NRC.

I.4 Research Initiatives for the Future

RES seems to have a good appreciation of the trends and challenges that it will face in the future and the research that will have to be done to achieve the NRC's mission. These challenges include the following examples:

- aging of the existing fleet of nuclear power plants and the need to make license renewal increasingly risk informed
- the increasing number of plants that are entering into decommissioning, and the need to explore alternatives to the current options for decommissioning
- technological changes in digital electronics for instrumentation and control of power plants and new fuels and cladding

- technological change that will take the form of new types of power plants that may be quite different in design than the current fleet

In the many research programs discussed in Part II of this report, one can see elements of research that are beginning to address these challenges for the future. It can also be anticipated that there will be continuing pressure on the NRC as a whole to carry out its mission more economically. As the current workforce retires, the NRC will be pursuing its mission with less-experienced staff. Thus, demands will arise for computational tools and information aids developed by the NRC's research to assist line organizations.

The following sections offer additional suggestions regarding issues that should be considered by the research program in the future:

I.4.a. New Power Plants and a Revised Regulatory Structure

A substantial research effort is currently underway to examine the possibility of risk-informing individual regulations. As part of that effort, the staff is working on risk informing Title 10, Section 50.46 of the *Code of Federal Regulations* (10 CFR 50.46) and the design-basis loss-of-coolant accident (LOCA). This laudable effort is challenging. Even if the piecemeal revision of regulations is finally successful, the results may still not be applicable to plants with designs that are significantly different from those of plants that are now licensed and in service. As a result, the Commission needs to determine, on a somewhat urgent basis, the requirements for a set of regulations that are not design dependent.

The ACRS expects that the most viable option would be a major revision of the regulatory approach to be fully risk informed. For this to be a viable option, however, the staff must first explicitly define the full spectrum of regulatory objectives expressed in terms of risk acceptance criteria. These could include such things as prompt fatalities, latent fatalities, total deaths, injuries, land contamination, worker exposure, and releases of all magnitudes of radioactivity. The possible role in the new framework of the cornerstones that have been defined in the revised oversight process should also be explored.

The safety goals (as re-interpreted in terms of the need for increased safety for new plants) could provide a starting point for this revision of the regulatory approach; however, the current safety goals are incomplete, and do not provide sufficient guidance with respect to uncertainties and defense-in-depth. To provide a proper perspective with respect to defense-in-depth and uncertainties, the risk-acceptance criteria need to be developed in terms of confidence limits.

For such a regulatory system to work, there must be quality, peer-reviewed, acceptable PRAs available for the various new reactor concepts. This will require revisiting the concepts of neutronics, thermal-hydraulics, fuel behavior, severe accidents, and fission product source terms. New models and new data are very likely to be required. Consequently, the ACRS recommends that the staff begin now, and continue with early interactions with designers and developers to identify the needs in this area.

The needs for traditional defense-in-depth must be clarified in terms of the unacceptable uncertainty in meeting the risk acceptance criteria. Those events that are of high consequences, but with high uncertainty for their frequency, must be identified, and decisions must be made regarding how to deal with them in a traditional defense-in-depth manner. The current requirement for a containment must be re-examined for new reactors. The need for containment must be predicted on the level of confidence to be attached to the design's ability to meet the various risk acceptance criteria without a containment.

One likely constraint on the potential for licensing new reactors is early site permitting. The Commission may be called upon to create a bank of approved sites. As a result, the agency needs to develop criteria and guidelines for how this can be done, particularly in view of the fact that there may be intentions to place multiple modular units on a single site. Particular needs are to identify dominant accident sequences and associated source terms for the new reactor designs. These needs raise two main questions:

- How will the multiple units on a site affect the risk-acceptance criteria?
- What will the NRC's position be with respect to multiple, automated, modular units that are managed by relatively few operators at a central facility?

Another challenge to the agency with respect to licensing the new designs will be to determine the role to be played by the concept of "licensing by test" of a prototype. Again, this challenge raises several questions:

- How many and what kinds of tests are necessary?
- What data are required?
- Who is to perform the tests, and who is to participate?
- What are the needs for validating the new computer codes for thermal-hydraulics and severe accidents for these reactor designs if the tests are not conclusive?

Finally, some advance thought should be given to the needs for surveillance and inspection of new plant designs. Plants (such as the Pebble Bed Modular Reactor) that utilize continuous refueling will undoubtedly propose "online" continuous calibration of instruments, rather than periodic shutdowns for "hands- on" calibration. Automated surveillance and diagnostic systems, as well as artificial intelligence-based systems being developed at DOE national laboratories and universities, will undoubtedly be proposed. Even "noise" and vibration detecting systems monitoring core structures and mechanical equipment, traditionally operated off line, may have to be automated and integrated into an overall monitoring system.

I.4.b. Risk Implications of License Renewal and Power Uprates

Although the NRC has set upon a course of making greater use of risk information in the regulatory process, two of the most important regulatory processes now underway, namely license renewal and power uprate, are being addressed by deterministic rules. Furthermore, the evaluations of plants for license renewal and power uprates, as well as evaluations for other significant regulatory actions such as extended fuel burnup, are essentially done independently of one another. Although the NRC can seek risk information on each of these types of regulatory actions for a plant, and can consider the synergisms that arise among these actions, this is not often done in detail because of the lack of data and computational tools that are well suited for evaluations of these regulatory actions or any synergisms among them.

After the implementation of each of these changes, plants will still meet regulatory requirements. However, there is little question that the useful life of systems and components is being consumed by the higher demand imposed by these licensing actions. Vessels are more embrittled at 60 years of life than at 40, fuel cladding is more embrittled at higher burnups, mechanical components are closer to their fatigue limits, and the burst pressure of flawed steam generator tubes is decreased.

Probabilistic risk assessments do not explicitly account for aging phenomena. A PRA that could account for aging of structures, systems, and components would provide measures of increases in risk metrics such as core damage frequency (CDF) and large early release frequency (LERF) as a plant ages from 40 years to 60 years of operation. These increases would result from a higher failure probability of long-lived components that are subjected to 20 additional years of service. These higher failure probabilities would be reflected in PRA results in several ways:

- increased frequency of accidents caused by rupture of passive components

- increased probability of cascading failures resulting from physical interaction of ruptured components with age-degraded components

- increased probability of failure of engineered safety systems

- reduction in the structural capability of the reactor coolant system and the containment during severe accidents

The magnitude of the increase in risk that would be calculated in such a hypothetical PRA cannot be estimated at this time. The risk increase may well be found to be small because the license renewal process is intended to provide assurance that aging management programs preserve regulatory margins in structures, systems, and components throughout the period of extended operation. Although licensees are not required to assess the changes in their risk profiles as a consequence of license renewal, the ACRS believes that the NRC should undertake an assessment of the impact of the rule on plant risk measures to demonstrate that the governing deterministic rule is effective in managing risk and ensuring that any increases are small and consistent with the safety goals.

Assessments of the risk increases associated with license renewal and power uprates may need to consider risk metrics other than CDF and LERF. A study[4] sponsored by the Swiss Federal Nuclear Safety Inspectorate assessed the risk associated with a 14.7% power increase of a Swiss BWR-6 with a Mark III containment. This study showed that the power upgrade would result in only minor increases in CDF and LERF, but the reactor would experience a 30% increase in the frequency of releases of radioactivity that were not sufficient to result in an increase in prompt fatalities, but could increase latent effects and land contamination. This risk increase is directly attributable to the increased radioactive inventory and the acceleration of events associated with the increased decay heat level at the uprated power conditions. This risk increase may be acceptable but, again, the NRC should undertake research to explicitly demonstrate the risks and their acceptability to ensure public confidence in upcoming power uprates. This work should also reconsider the criteria used to judge changes in risk, which are currently limited to CDF and LERF.

Extended burnup of fuel, as noted in Part II of this report, has both societal and economic benefits. There seems to be little doubt that there will be interest in continuing the historical trend of ever increasing fuel burnups in power reactors. This trend has been arrested by recent regulatory fiat, but this interruption is likely to be temporary. The nuclear industry is marshaling the technical basis, independent of plant aging and power uprates, to justify burnups that go well beyond the current regulatory limit of 62 GWd/t. At the same time, events are being recorded of operational difficulties associated with high-burnup fuels, and research information is being accumulated regarding unusual fuel and cladding responses to upset conditions.

Concurrent licensing actions (such as power uprates) also consume margins existing in the design of structures, systems, and components, and compete with aging for these margins. The proposals for power uprates are being considered without explicit consideration of aging. The ACRS believes that the NRC will have a growing need to be able to evaluate concurrent licensing actions such as license renewal, power uprates, realistic accident analyses, and extended fuel burnups. The exclusion of explicit consideration of aging risk from the evaluation of plants for license renewal does not mean that the aging effects that are allowed by license renewal should be excluded from consideration in assessing the risk posed by other licensing actions that may be affected by the aging of structures, systems, and components. For example, the risk associated with a power uprate could be significantly higher if the increased radioactive inventory and the time acceleration of events associated with the uprated power conditions were to occur in a containment with its ultimate capability degraded by aging.

In principle, PRA could account for the individual and synergistic effects of concurrent licensing actions, such as license renewal, power uprates and extended fuel burnup. However, the supporting risk and phenomenological models are not yet sufficiently developed to do this. Model uncertainties are expected to be large, and the metrics that are commonly used in PRAs may not adequately represent the risks that are associated with these changes taking place in nuclear power plants. Therefore, the ACRS recommends that the NRC undertake, as part of the move toward risk-informed regulation, a research program to investigate the feasibility of improving PRA methods and the supporting phenomenological models to explicitly deal with aging, power uprates, extended fuel burnup, and other major changes taking place in nuclear power plants. This undertaking is likely to be a multidisciplinary challenge akin to the effort that the agency is pursuing in connection with the Pressurized Thermal Shock Rule. Certainly, this undertaking will involve the agency's considerable expertise in risk analysis, materials and metallurgy, reactor fuel, and thermal hydraulics.

I.4.c. Decisionmaking Method

With the exception of some isolated instances such as the decisions regarding backfits and possibly the significance determination process used in the Revised Reactor Oversight Process, the decisionmaking processes that are used in the regulatory process have a subjective character with no assistance from formal decisionmaking methods. As an example, consider the integrated decisionmaking process discussed in Regulatory Guide (RG) 1.174[5]. Little guidance is provided regarding how to integrate the results of PRAs, defense in depth, and safety margins. Consider also the term "increased management attention," which is employed in RG 1.174 with very little explanation of what it means in practical terms. As another example, NRC decisions usually involve interactions among several affected parties both within and outside of the agency. Yet, these interactions lack structure. Similarly, the justification of the "action matrix" in the risk-informed reactor oversight process remains unclear.

Some licensees have begun using methods to structure their decisionmaking deliberations. For example, the Expert Panel that categorized the structures, systems, and components according to their respective risk significances for the South Texas Project exemption request employed questions that were assigned weights of relative significance as well as rating scales. The staff's review of this exemption request could have been facilitated if the staff had been more familiar with the substantial body of literature regarding decisionmaking processes.

The subjective nature of many regulatory decisions is sometimes justified by noting that the decisions have to be made in the face of significant uncertainties. To be sure, there is a body of literature on making decisions in the face of uncertainty. This literature covers a spectrum of approaches ranging from formal, mathematical formulations to behavioral approaches. The common objective among all of these methods is to rank the desirability of decision options according to well-defined criteria. Many of the decisionmaking methods have matured to the point that relevant computer aids are marketed.

The benefits of using more formalized methods for making decisions are similar to the advantages that have come from the use of PRA in safety analysis:

- Decision trees or influence diagrams help the decisionmaker structure the decision process, much as event trees and fault trees help structure safety analyses.

- Decision trees and influence diagrams serve as common communication tools for the disparate parties affected by the decision, much as event trees and fault trees communicate the nature of risk analyses.

- Decision criteria are explicit, and this increases the probability that a consensus will be reached on a decision.

- It is possible to conduct understandable sensitivity studies to clearly see how the order of preferences of options changes as the inputs to the process are varied.

The staff argues that it is engaged with other Federal agencies and foreign institutions in examining the ways that risk information is used in making decisions. This is too timid an approach. The NRC leads the world in the use of risk information, and must accept a leadership role. Therefore, the ACRS recommends that the staff initiate a program of research to investigate how best to use formal decisionmaking methods in regulatory decisions.

II. ONGOING RESEARCH PROGRAMS

The ongoing research programs have been divided into 13 technical areas. The programs in these areas are discussed in the subsections that follow.

II.1 Probabilistic Risk Assessment Research and Applications

II.1.a. Risk Assessment Technologies in a Risk-Informed Regulatory System

The Commission has set upon a course of making greater use of risk information in the regulatory process and has made enormous strides toward achieving this goal. The Commission has set Safety Goals cast in the language of quantitative risk assessment. Risk information and changes in risk are used as input to decisions to change the licensing bases of nuclear power plants (Regulatory Guide 1.174). Risk importance measures are used in categorizing the various structures, systems, and components for maintenance (10 CFR 50.65). Risk information is used in the development of inservice inspection programs and in specifying the allowed outage times for plant safety systems. Risk information has been used to design the Commission's new Reactor Oversight Program and the Commission has instituted a process of risk-informed enforcement of its regulations. The Commission has a risk-based process (Accident Sequence Precursor Program) for evaluating risk-significant operating plant events. The Commission is now looking at revising the regulations for nuclear power plants using risk information.

Quantitative risk information that is at the heart of these evolutions in the regulatory structure comes from PRA. One would expect that the Commission and the line organizations of NRC that are responsible for implementing the risk-informed regulatory process would have available comprehensive, state-of-the-art risk-assessment tools capable of analyzing site-specific regulatory issues. In fact, the NRC has quite good risk assessment tools, but still it could be improved substantially.

To a very real extent, the risk information available to the Commission and the staff that has been used to construct the technical bases in developing a risk-informed regulatory system has come from risk assessments done more than a decade ago for five representative nuclear power plants, and the individual plant examination (IPE) submittals from licensees that are widely regarded to be of variable quality and, in many cases, out of date. The risk information available for the design and implementation of the risk-informed regulatory process focuses heavily on risk during normal plant operations and not shutdown operations during which so many events requiring special investigations by the NRC staff occur.

II.1.b Ongoing and Planned PRA Research Programs

There is a mortgage on the NRC's steps toward a risk-informed regulatory process. This mortgage must be paid in research to further develop the agency's capabilities to understand and characterize risk in order to carry out the Commission's mission. Table 1 lists the research programs that are currently under way and planned in the area of PRA. Comments on these research programs and others that should be under way in the agency are presented below. Human performance and human reliability are so important to the quantitative assessment of risk that a separate section of this report (Section II.2) is used to comment on research in these areas. Fire protection risk that appears from the individual plant examination of external events (IPEEE) submittals to be a significant aspect of the risk profiles of nuclear power plants is also addressed in a separate section of this report (Section II.3).

- Risk During Plant Transitions and Shutdown Operations

Following directions from the Commission, the staff is not developing the capability to estimate risk during low-power and shutdown modes of operation. As a result, the staff must rely for risk information on its own scoping assessments of shutdown risk at two representative plants and a limited number of shutdown PRAs of varying scope and completeness. These scoping risk assessments are widely regarded as conservative in the sense that they overestimate risks during low-power and shutdown modes of operation. There is no doubt that the scoping assessments are out of date in the sense that they no longer reflect the nature of operations studied at the plants.

The recent revisions to the Maintenance Rule make clear the regulatory expectation that licensees will manage risk during shutdown operations. The nuclear industry has made substantial efforts to improve operations during planned shutdowns and has greatly reduced the frequency of unplanned shutdowns. The industry has developed several quantitative and qualitative risk assessment tools to guide the planning of shutdown operations. In a workshop sponsored by the NRC, licensees acknowledged that there is still substantial risk associated with shutdown operations experience.

Experience shows that the systematic search for accident sequences produces a far more complete picture of the way failures can occur in complex systems. That statement is true for normal operations; however, we do not have a similarly comprehensive and complete understanding of the risks during shutdown operations. Without this understanding, we cannot assess the effectiveness of current NRC regulations to deal with shutdown risk. The staff cannot independently evaluate the safety of plants for shutdown operations and interpret results of licensees' qualitative risk assessments. The staff relies on the licensees to evaluate the risk importance of the equipment that is available and not available during shutdown operations, the safety benefits of online inspection and maintenance versus inspection and maintenance during plant shutdown, and the risk significance of plant changes that will affect shutdown modes of plant operations.

In addition, the categorization of the structures, systems and components that are needed in a risk-informed special treatment process (or Option 2) will be incomplete as long as the risks from plant transition processes and shutdown operations are not quantified. The risk management tools used by the industry are not suitable for the shutdown risk determinations needed by the NRC. Specifically, the NRC needs the capability to realistically estimate the risk that arises from a set of plant configurations during shutdowns that are projected to occur over a plant's operating lifetime. The ACRS, therefore, continues to believe that the agency should undertake an effort to develop the capability to assess risk during low-power and shutdown modes of operation. We recommend that the Commission authorize the staff to undertake such an effort.

- Standards for Probabilistic Risk Assessments

As the NRC makes greater use of risk information in the regulatory process, it will be necessary for licensees to submit risk information in conjunction with regulatory actions. It will be impractical for the staff to repeatedly review site-specific PRAs associated with these processes. Consequently, the staff must have confidence that PRAs used by licensees in developing of their proposals meet some minimally acceptable standard. An effort, supported by the NRC, is under way to develop a consensus industry standard regarding the quality of PRAs for normal plant operations. This effort competes, in some sense, with an industry effort to peer review and certify for application PRAs done by individual licensees. The development of a standard for PRA quality and the approval of the industry's certification process have been a struggle. Difficulties arise in this process because the NRC has not attempted to definitively define the necessary risk assessment features that must be available to support regulatory decisions. The ACRS believes that there is a pressing need for the NRC to undertake a research process that will develop definitions of necessary PRA features. As the ACRS has noted many times, the sufficiency of information derived from PRAs for decisionmaking depends very much on the nature of the decision. It is probably not possible at this time to attempt to develop definitions of the sufficiency of PRA features for regulatory purposes.

- The SAPHIRE Code

The NRC risk-assessment code (SAPHIRE) continues to undergo development, and revision of this code appears at somewhat regular intervals. There is, however, no readily apparent strategy in these development efforts. An agency effort to define the necessary features of PRAs to support the regulatory process might well provide the agency with a more scrutable strategy for the development of this code. In any event, the ACRS believes that the SAPHIRE code has reached a stage of development that the public deserves to see a comprehensive peer review of this code that plays a substantial role in the risk-informed regulatory system.

- Common-Cause Failures

Common-cause failures are of crucial importance in PRAs. The NRC has completed a comprehensive study of common-cause failures. Data presented to the ACRS shows that the frequency of events involving common-cause failures has been decreasing (See Figure 1). No developments in the methodologies for analyses of common-cause failures are planned. The current program is one of maintaining and updating databases. The ACRS believes that this program has served its regulatory purpose, and that the work can be brought to an orderly close.

- Risk Importance Metrics

An important part of the effort to focus regulatory attention is a more 'risk informed' classification of structures, systems, and components. Processes have been developed to do this in the context of the Maintenance Rule, graded quality assurance, and currently reassessment of the need for 'special treatment.' Such classifications can be expected to be important if attempts are made to risk inform license renewal or the 10 CFR 50.59 process. Risk importance measures are critical tools for such classifications. Currently, the most widely used risk importance measures are the Fussell-Vesely and Risk Achievement Worth. The ACRS has commented in another report[6] on the advantages and limitations of these metrics. Although the deficiencies in these metrics can probably be overcome in particular cases through the use of more qualitative assessment by an expert panel, such approaches require additional review by the staff and add to the burdens already placed on expert panels. Because of the importance of the classification process to risk-informed regulation, the ACRS believes that the NRC should have a research program that searches for metrics that do not suffer the disadvantages of those now in common use. The Top Event Prevention methodology and importance measures that are predicted on partial derivatives of CDF and LERF are examples of alternative approaches that should be investigated for the determination of risk significance.

- Improved PRA Capabilities

The prominent role that quantitative risk assessment must play in a risk-informed regulatory system suggests that there should be continuing research to improve and expand PRA capabilities. The ACRS recommends three areas of such expansion, including plant aging, safety culture, and latent human errors. Inclusion of plant aging seems to be an inevitable need. Although the license renewal process is now highly deterministic, it will need to become more probabilistic and risk informed as plants requesting license renewal adopt risk-informed systems. There are some indications that it may be possible to quantify the effects of safety culture on human performance. Of even greater interest is the evidence that there could be an optimal level of regulatory involvement to encourage safety cultures. Latent errors (human errors that have no immediately observable impact, but later affect a plant) have been found in a study by the NRC staff to have been four times more common in risk-significant plant events than overt human errors.

- Quantification of Uncertainties in PRA Results

Results obtained from PRAs are uncertain because of uncertainties in both the models used in the risk assessments and in the parameters adopted for these models. It is widely acknowledged that quantitative assessments of uncertainties are essential inputs to the regulatory decisionmaking process. Yet, carefully quantified uncertainties seldom appear in risk-informed regulatory discussions. When uncertainties are mentioned, they are typically the results of propagating parameter uncertainties through the analyses. Even with parameter uncertainties, there are questions especially with regard to plant-specific distributions, are typically produced by combining generic distributions with plant- specific data using Bayes' theorem. The issue of concern is that the Bayesian updating process will eliminate low-probability tails of the generic distribution, yet these tails may be applicable to real accidents. The ACRS believes that the staff should research the quantification of parameter and model uncertainties for plant-specific applications. (See also Section II.6.)

- Risk-Based Performance Indicator Development

The new risk-informed oversight process is a major agency initiative. This process relies on a combination of performance indicators and baseline inspections. The existence of a good set of risk-based performance indicators will allow the agency to have a process that will be more performance based, will recognize plant-specific attributes, and will help define risk-significant trends. The ACRS is very supportive of this research.

Table 1. Ongoing and Planned Probabilistic Risk Assessment Research

	Task Description
J8263	Risk-Based Performance Indicator Development
Y6211	Development of Risk Based Performance Indicators
L1429	SAPHIRE Maintenance and User Support
W6224	Risk-informing 10 CFR Part 50
W6241	Plant Database Development for SAPHIRE
W6528	Root Cause Investigation Improvements
W6970, W6971	Support in Development of Consensus PRA Standards
Y6036	Modify 10 CFR Part 50
Y6194	Common Cause Failure Database
Y6184	Revised Reactor Oversight Program Support
planned	Develop standards for the application of risk-informed and performance based regulation in conjunction with national standards committees
planned	Develop improved methods for calculating risk
planned	Develop and maintain analytical tools for staff risk applications
planned	Risk inform NRC regulations
planned	Develop guidance on PRA applications to ensure uniform comprehensive application of PRA models
planned	Improve risk-informed decisions associated with natural hazards
planned	Preliminary results of a study using finite element modeling of coastal flooding to resolve concerns with the NOAA model predicting beyond design basis storm surges
planned	Develop PRA methods for NMSS regulated facilities and devices

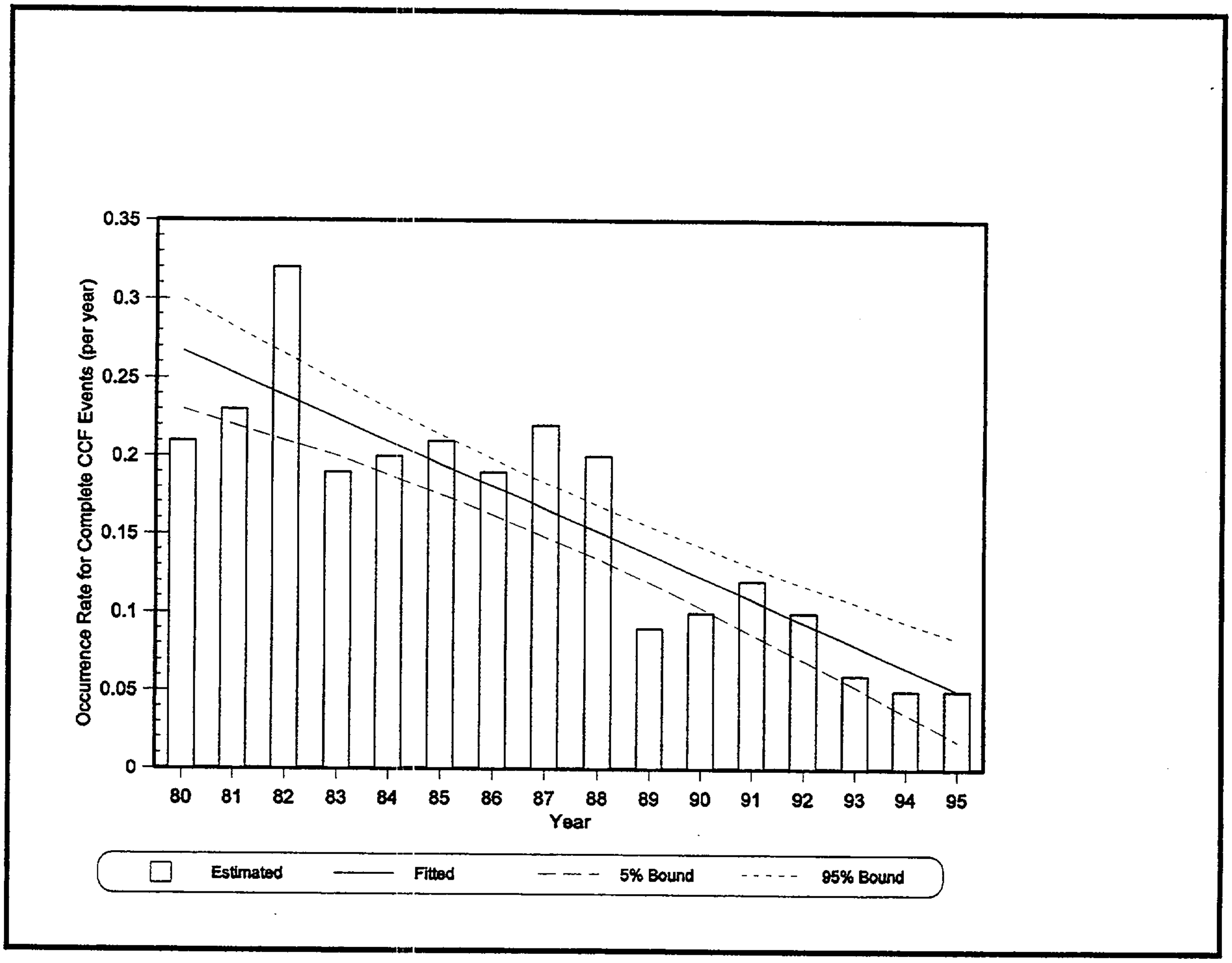


Figure 1. **Trend in the Occurrence Rate of Common Cause Failure Events***

* U.S. Nuclear Regulatory Commission NUREG/CR-6268, "Common-Cause Failure Database and Analysis System," June 1998.

II.2. Human Factors Research

Human performance and human factors are currently viewed as important, al beit unquantified (in some cases), contributors to the risks posed by nuclear power plants. It is thought that the importance of human factors may well increase as competitive pressures grow within the nuclear power production industry. The issues of interest are not solely confined to the plant operators; rather, they include the entire workforce involved in the design, operation, and maintenance of a power plant. Issues that are cited in this regard include changes in staffing, management, and organization. The issues of interest include overt failures in human performance, as well as latent errors that have no immediately observable impact, but later contribute to events at a plant. The issues of human factors and human performance may well be of even greater interest if generation of nuclear power at future plants is achieved by multiple, automated, modular units that are managed by relatively few operators at a central facility.

In the discussions of human factors and human performance presented here, human reliability analysis is included along with those projects that the NRC has typically included in this research area. This is done because the ACRS believes that there needs to be a closer integration between human reliability analysis and other human factors research at the NRC. On the other hand, discussion of human factors associated with the introduction of digital instrumentation and control (I&C) systems is deferred to the discussion of these digital systems later in this report.

a. History of Human Factors Research at the NRC

The NRC established its first Human Factors Program following the accident at Three Mile Island-Unit 2. The NRC commissioned the Human Factors Society to develop a plan to assist in establishing this first human factors program. The agency adopted some of the recommendations of the Society, and published its first Human Factors Program Plan in 1983.

In 1985, the human factors research function was discontinued within RES. The function was reestablished in 1987 on the basis of a recommendation in the National Research Council's report, "Revitalizing Nuclear Safety Research." That report recommended a research agenda that included human-system interface design, personnel subsystem (training, licensing, work scheduling), human performance (measurement, prediction, and modeling of human error), management and organization, and research on the regulatory environment. In 1989, published a report entitled 'Human Factors Regulatory Research Program Plan" was published, in part, to respond to the recommendations of the National Research Council's report. The plan addressed all of the recommended areas of research, with the exception of research on the regulatory environment.

In 1994, a Human Factors Coordinating Committee was formed to coordinate activities within the various NRC offices that deal with human factors. The Human Factors Coordinating Committee published the "Human Performance Program Plan" in 1996. The ACRS[7] was critical of this plan, since it was more an inventory of activities than a plan. A revision to the plan, published in 1998, included in its mission statement:

- Identify human performance issues important to public health and safety.

- Increase understanding of the causes and consequences of inadequate performance by humans.

- Implement appropriate regulatory responses to such issues.

The ACRS agreed with the mission statement of the revised plan, but found that the staff did not have a systematic approach for achieving the three goals of the mission statement.[8] A revision was published in 1999 and the ACRS[9] commented, " ...the staff described a disciplined strategy for future development of a more technically justified [human performance plan]. We believe the following two elements of this strategy are valuable: review of the Accident Sequence Precursor (ASP) data to identify the contribution of human performance to significant events and interaction with other organizations, such as the Institute of Nuclear Power Operations (INPO), that have a strong focus on human performance. The NRC conducted a workshop in April 2001, and to utilize input obtained in this workshop to produce a detailed, 5-year, human performance implementation plan.

The NRC's human performance plans include the acquisition of experimental data to validate human reliability analyses. This is being done largely through the NRC's participation in the Halden Program. Human reliability analysis research done at the NRC is not a part of the plan. This research has focused on the issue of evaluating human errors of commission in PRA and has been developing A Technique for Human Event Analysis (ATHEANA) methodology in what appears to be substantial isolation from the rest of the human performance activities at the NRC.

The evolution to a Revised Reactor Oversight Process at the NRC has included an acknowledgment of the importance of human performance. The staff has not identified specific indicators for human performance in the operation of a nuclear power plant. Instead, it has asserted that any degradation in human perform- ance will be revealed by degradations in other areas that are addressed by performance indicators. The ACRS[2] has termed assertions concerning the ability to detect degradation of human performance from performance indicators associated mostly with plant hardware an "untested assumption." The ACRS[10] has noted that results of the staff's analyses of ASP data have shown that latent human errors with no immediately observable impact were four times as common as overt errors.

b. Ongoing and Planned Research Programs

Table 2 lists the ongoing and planned research programs in the area of human performance and human factors. Some comments on these programs are provided below:

- Corrective Action

The intention of this program is to develop a guidance document to assist in the review of human performance aspects of a licensee's corrective action program. This will be an important aid to inspectors in the Revised Reactor Oversight Process. This work should be completed, and allowances should be made in future research programs to evaluate the effectiveness of the guidance and any lessons learned from its use.

- Halden Experiments

The NRC has long been a partner in the international cooperative effort known as Halden project. A much proclaimed feature of the Halden project is the study of operator performance in an instrumented simulator. Unfortunately, there is not a close tie between the NRC's need to develop methods of human reliability analysis and the experiments being done with the Halden simulators. Despite several attempts, the ACRS was unable to identify instances where information obtained in the Halden simulator experiments was ever used in the human reliability analysis program. This is not to say that the information could not be used. There must, however, be some greater coordination of the activities. Furthermore, there must be some demonstration that data obtained in a program conducted by the Committee on the Safety of Nuclear Installations (CSNI) using a simulator for a VVER Russian reactor and Finnish operators will yield results that are transferable to U.S. nuclear power plants.

- Task 20: Determining the Impact of Human Performance on Risk
- Task 21: Evaluation of Human Performance in the Revised Reactor Oversight Process

This activity has involved the examination of ASP results, and has yielded a determination that in 20 events with conditional core damage probabilities between $1x10^{-5}$ and $5x10^{-3}$, "... the average contribution of human performance to the event importance was over 90%."[10] The study also found that latent errors were four times more numerous than overt or active errors. This work is being continued to examine the issues of human performance in support of the Revised Reactor Oversight Process, and should test assumptions that have been made in the design of the process related to the influence of human performance on plant safety. The ACRS believes that this type of work which attempts to quantify the impact of human performance on the basis of operational data is of critical importance to the design of a useful NRC research program in the area of human factors. The ACRS believes that the work should be continued—and even expanded—answers to the following questions:

- What is the risk significance of the human element in the operation of a nuclear power plant?

- When is human performance at a nuclear power plant "good enough"?

- What are the significant latent errors, and why do they occur?

- Are current regulatory staffing requirements having a meaningful impact on safety?

- Root Cause Investigations

The Revised Reactor Oversight Process has altered the role of inspectors with respect to investigating of human performance events. Specifically, root causes of the events will be examined by the licensees, and for findings of very low safety significance (green findings) NRC inspectors will no longer be involved in these detailed examinations of the events. Instead, they will examine the licensee's corrective action program to ensure that its methods and executions are adequate. This work provides the necessary revisions to the human performance protocol. As such, it is a necessary activity to support the agency's evolution to the Revised Reactor Oversight Process.

- Develop Review Guidance (NUREG-0700)

The detailed design review guidance for control rooms provided by NUREG-0700 has become anachronistic in the face of the NRC's move toward more risk-informed and performance-based reactor oversight. In the current regulatory environment, the prescriptive guidance provided in NUREG-0700 is more likely to inhibit innovation in the design of control rooms than it is to promote safety. Furthermore, the performance of hardware and operators in the control room, rather than the details of design, are to be the focus of regulatory attention within the Revised Reactor Oversight Process. This will be even more true for the designs of control rooms for future reactors. Consequently, the work in this project no longer appears to be consonant with agency needs and should be brought to a close. The NRC should be looking for performance-based guidance in this area.

- Develop a Human Performance and Reliability Plan

This program involves the development of a more detailed, 5-year plan for human performance activities and research at the NRC, as mentioned above. The ACRS is supportive of the development of this plan, since the directions of the efforts now seem to be along systematic avenues consistent with the objectives in the mission statement for the current plan and supportive of the NRC's evolution toward risk-informed and performance-based regulation and the Revised Reactor Oversight Process. The ACRS notes that work such as that discussed above in connection with Tasks 20 and 21 is more likely to provide defensible technical bases for planning than the NRC will obtain by continuing to collect unquantified opinions at workshops.

The ACRS reiterates its advice given in regard to a previous incarnation of the human performance program plan, the development of a plan for research on human factors is certainly not a simple task. This task would be made easier and the recommendations more convincing if the task were guided by a high-level model that identifies the important elements that influence the likelihood of unsafe human acts. The need remains for this high-level model as the basis for planning. Human performance will not receive the attention that its importance merits in the regulatory scheme if it is treated at the margin as seems to be the case in the Revised Regulatory Oversight Process. There must be an overarching strategy that properly places human factors in the overall approach to safety. There is no regulatory model that establishes the place of human factors issues in the regulatory structure. Are human factors more important than, less important than, or of equal importance to equipment reliability? Without agreement on the importance of human performance in the regulatory scheme, there will be no agreement on the research needs, no basis for allocating resources to meet these needs relative to other needs, and no agreement on when these needs have been met.

- ATHEANA Development and Application

The ATHEANA effort was initiated with the laudable, if ambitious, goal of expanding the capabilities of PRA to include human errors of commission, as well as human errors of omission that are currently considered. ATHEANA has attempted to use the concept of error-forcing context in its development and the ACRS has applauded this effort. At the same time, the ACRS has identified important elements that are missing from the identification of error-forcing contexts, such as the safety-conscious work environment. Applications of ATHEANA have failed to demonstrate that the contextual approach has provided any benefits when compared to traditional methods. A major problem with ATHEANA is its failure to model the relationship between error-forcing contexts and the probability of unsafe acts. The staff has acknowledged the need to initiate quantification in ATHEANA. After several years of development aimed at expanding the capabilities to quantify risk, ATHEANA is only now beginning to address the issue of quantification. The development team is now investigating the applicability of other approaches that are not necessarily consistent with the much-publicized idea of error-forcing context. ATHEANA has not been evident in any of the NRC's programs dealing with human performance, especially mission-critical programs such as the Revised Reactor Oversight Process. The ACRS has been told that ATHEANA will be applied to the human performance aspects of the Pressurized Thermal Shock program now under way at the agency (see Materials and Metallurgy, later in this report). The ACRS concludes that the greatest achievement of the work to date is to sensitize people to the concept of error-forcing contexts. The ACRS, therefore, recommends terminating the ATHEANA effort, and developing a new plan to quantify the probability of unsafe human acts.

- Develop Human Factors Regulatory Guidance

This program of future research activities in the area of human performance will be affected by the planning efforts that the staff now anticipates. Current plans are to focus on four areas:

- effects of deregulation and organization changes, including consolidation of the nuclear utilities
- quantification of experience and data to improve human reliability
- analysis for events such as steam generator tube ruptures or anticipated transients without scram
- addressing the issue of latent errors in probabilistic risk assessments

Although the individual elements of these future plans may have some merit, the ACRS is concerned with the approach that is being adopted to develop these future plans. The planning process for the human performance program seems to be driven by the identification of possible or expected changes in the regulatory environment, examining the human dimension of the regulatory issues involved and then defining a project to address specific human performance concerns. This approach can produce a list of plausible research activities, but it treats human performance at the margins of the regulatory process. There is no articulation of the place of human performance in the regulatory structure relative to all of the other elements in the structure that compete for support. Thus, there is no context for judging either the importance of individual issues or the completeness of the program.

Table 2. Ongoing and Planned Research Activities in the Area of Human Performance

	Task Description
Y6315	Crew Systems Ergonomics Research Information Analysis Center* Support
Y6315	Corrective Action
B7488	Halden Experiments
E8238	**Task 20:** Determine the impact of human performance on risk **Task 21:** Evaluation of human performance in the NRC Revised Reactor Oversight Process
W6528	Root Cause Investigations
W6546	Develop Review Guidance (NUREG-0700)
W6994	ATHEANA
planned	Develop human performance and reliability plan
planned	Develop human factor regulatory guidance
planned	Develop the technical bases for review guidelines that will be used to evaluate human performance

*Now called "Human Systems Information Analysis Center."

II.3. Fire Protection Research

In the aftermath of the fire at the Browns Ferry Unit, the Commission established additional fire protection requirements for nuclear power plants. These highly prescriptive requirements seemingly have been effective. Although significant fires continue to occur at nuclear power plants, none has posed a threat to the public health and safety comparable to the fire at Browns Ferry.

Because the fire protection requirements were "backfits" to the safety profiles of many nuclear power plants, a complicated regulatory quilt of Appendix R plants and license-condition plants exists. All these plants were reviewed against the guidance contained in the Branch Technical Position (BTP) applicable to its vintage. There are many hundreds of variances, inaccurately called exemptions, to specific provisions of the requirements. As a result, monitoring and inspecting fire protection programs at nuclear power plants is a technically complicated process for the NRC. Furthermore, the protection achieved by the prescriptive requirements has been obtained at substantial costs to the licensees. As commonly found with prescriptive safety requirements, the current condition probably does not take optimal advantage of the engineering and managerial creativity available to the licensee, and does not encourage the taking of fire protection measures beyond those that are mandated by the regulations.

The NRC has asked licensees to estimate in the IPEEE effort the residual risk of core damage associated with accidents initiated by fires as part of its vulnerability assessment. Although insights have been slow to emerge from this effort, anecdotal accounts suggest that it has often been found that the core damage frequency associated with fire initiators can be commensurate with the core damage frequency from all other operational initiators. Such high estimates of the core damage frequency despite the current, comprehensive fire protection requirements may be accurate and not a reflection of excessive conservatism in the fire risk assessment methods. Fire protection systems are not safety related, and do not have the diversity and fault tolerance requirements of other safety-related plant protection systems. If, indeed, anecdotal accounts concerning the risks of fire at nuclear power plants are accurate, although the NRC has invested resources in fire-related issues, it may be argued that fire has not received the attention (relative to other safety issues) that is merited by its risk significance.

Today, the NRC finds itself with substantial limitations in its abilities to apply risk concepts to the regulation of fire protection at nuclear power plants:

- Fire growth models used by the NRC do not have the technical sophistication or capabilities of models developed at the National Institute of Standards and Technology (NIST) and in other industries.

- Inspections of licensees' safe shutdown programs in response to fires are specialized activities that cannot be totally done by the NRC's regional staff, and require attentions by specialists from the headquarters staff.

- Neither the onsite inspectors nor the Senior Reactor Analysts have fire risk assessment tools that allow them to independently evaluate fire hazards and qualitative fire hazard assessments done by licensees.

- The staff disagrees with licensees regarding the assessment of electrical circuit failures during fires. Computational methods should be developed to facilitate the circuit analyses and risk inform the entire process.

These burdens on the staff and licensees could be relieved by research in the area of fire protection. The ACRS has benefitted from an excellent document prepared by the staff entitled, "NRC Fire Risk Research Plan: Fiscal Years 2001–2002." This plan is an example of the type of document that is needed to define research objectives, identify programs, estimate costs, set milestones, and identify future goals.

- Fire Risk Assessment Tool Development

This task is intended to develop many of the technical tools and databases that are needed to do a risk analysis:

- improved estimates of challenging fire frequency
- fire effects database including characteristic heat release rates
- guidance for identifying scenarios for which smoke effects may be important
- improved estimates of the probability of fire and fire containment
- configuration and condition dependent fire protection system reliability estimates
- improved estimates of the probability of spurious actuation of equipment during fires
- adequacy of current methods for analyzing fires in turbine buildings
- risk methods for prioritizing fire barrier penetration seals

The ACRS understands that these needs for improvements to fire risk assessment methods have been selected from a more extensive list of risk assessment weaknesses, and the ACRS is supportive of the proposed research. The ACRS believes that the plans for this research could be improved in the following ways:

- The proposed improvements of fire risk analysis tools should be part of a systematic assessment of the scope and quality of fire risk assessment technologies that the NRC needs (and will continue to need) to execute its mission now and in the future.

- The proposed improvements should be predicated on an explicit understanding of whether fire risk assessment at the NRC will remain the function of headquarters, technology used by Senior Reactor Analysts, or technology that is more generally available for use by line organizations.

The ACRS is concerned about an interminable code development effort with episodic revisions that are not targeted to meet regulatory needs. Augmentation of the effort with a careful examination of the fire risk assessment information that is needed for the regulatory process would allow better planning of the needed research. Similarly, quantitative goals for improvements in the technology are useful, as long as they are treated as goals and not requirements, to aid researchers in the identification of productive paths for their work.

The ACRS notes that the plans include efforts to characterize realistic heat release rates and other features of prototypic fires. Successful completion of this part of the research will be most useful for many of the technical issues concerning fire that have arisen in the regulatory process such as fire barriers and electrical circuit damage. This research could allow the NRC to use realistic estimates of fire characteristics that are appropriate for nuclear power plants, rather than bounding fire characteristics that are intended for generic situations found in standards such as those advanced by the American Society for Testing and Materials (ASTM).

- Fire Risk Requantification Study

This program will attempt to develop the kinds of insights concerning fire risk that were derived by the NUREG-1150 study for plants during normal operations. The ACRS is enthusiastic about the concept behind this research, but details of the undertaking are not currently available. It is important that the level of effort needed for this work, including the detailed analysis of uncertainties not be underestimated. The insight derived from this study could lead to changes in the priority, or even the need for other research in the plan.

- Fire Model Benchmarking and Validation

This task is part of a collaborative program that will compare the capabilities of codes like the Computer Code for Modeling Compartment Fires (COMPBRN) and the Fire-Induced Vulnerability Evaluation (FIVE) method with more sophisticated fire models such as the CFAST model developed by the National Institute of Standards and Technology (NIST) and with experimental data. The ACRS feels that this is a step in the process of defining the types and technical sophistication of fire growth models that the NRC needs to carry out its regulatory mission. The collaborative effort cannot be the entire scope of work for defining improvements to fire effects and growth models. There are aspects of fires important to the safety analysis of nuclear power plants that are not treated by the NRC's current model, and these need to be taken into account in the definition of fire modeling needs. Eventually, the NRC will need models of fire growth and behavior that are in the public domain. This work could be expanded by planning for the development of such models.

- Fire Risk Assessment Guidance Development

This task is to develop guidance from the results of the fire risk requantification study mentioned above. This may be a first step in defining a standard for fire risk assessments, especially if it builds upon a bottom-up assessment of the fire risk information needs of line organizations to carry out the NRC mission.

- Fire Protection for Nuclear Power Plants

The focus of this effort is to model fire detection methods. The ACRS agrees that there must be confidence in the modeling of fire detection systems, but it is not apparent why this work is not part of the Fire Risk Assessment Tool Development effort. The need for this work would be better appreciated if it were placed in the context of a complete description of the scope and precision of fire risk assessment capabilities that the NRC needs for its mission. Again, any plans for the improvements in modeling would be helped by the establishment of the goals for the improvements, preferably in quantitative terms.

- Fire Protection for Gaseous Diffusion Plants

Fire is very likely to be a risk-dominant accident in gaseous diffusion plants. This task is to develop analytical tools to support the placement of sprinklers for fire suppression, fire properties of combustible liquids, and unspecified fire protection and risk information needed for the regulation of gaseous diffusion plants. The plans do not seem to address issues of smoke and the transport of toxic, corrosive, and radioactive materials within plants during fires. Again, the ACRS believes that the planning in this activity could be improved by defining the complete scope of fire risk assessment needs for regulation of gaseous diffusion plants. The ACRS assumes that much of the work done for the gaseous diffusion plants will be applicable to the facility that will be licensed by NRC for the fabrication of mixed-oxide (MOX) fuel elements, as further discussed elsewhere in this report. Fire is very likely to be the risk-dominant accident at this facility, and transport of radioactivity in smoke will be the mechanism by which fire events have significant consequences.

- Fire Significance Determination Process Support

The ACRS believes that the fire-significance determination process must be corrected to have technically defensible bases (which it does not currently have) and, to minimize the dependence that now exists on subjective evaluations. This research should be directed toward these ends.

- Fire Risk Assessment Tools for Precursor Analysis

The ACRS is very supportive of this element of the proposed fire research. If, indeed, fire poses the risks suggested in the IPEEE results, then it will be important to have a process for evaluating fire precursors similar to that available for examining precursors during normal plant operations (ASP).

An area of continuing controversy is the analysis of safe shutdown circuits. It is known that the industry is trying to develop a probabilistic model to address this issue and to provide the data needed to evaluate the model. The NRC needs to have in hand technical information to define what would be prototypic fires that could yield meaningful probabilistic data for the resolution of this issue. The NRC may even need to carry out its own confirmatory experimental studies.

Because manual fire suppression is often an important element of a nuclear power plant's fire protection plan, there are human factors that need to be considered in the assessment of fire risks. The NRC needs a systematic method to assess how smoke will affect the performance of fire fighters, and how the Occupational Safety and Health Association (OSHA) two-in and two-out rule will affect the effectiveness of manual fire suppression strategies. It is our expectation that eventually experimental data will be needed to resolve these issues.

A challenge to the staff that is now on the horizon is the licensing of a facility for the fabrication of MOX fuel assemblies for use in light-water reactors as a means to dispose of some of the world's excess weapons-grade plutonium. Fire will be a major safety concern at this facility. Although most of the requirements for the facility can be drawn from industrial fire protection standards, the peculiar element of plutonium aerosol dispersal in fires will require specialized attention by the staff. Indeed, there are issues concerning smoke aerosol dispersal in nuclear power plants, since corrosive components in smoke can affect electrical and electronic circuits long after the fire has been extinguished. The ACRS believes that the staff needs to have the technical capability to predict the dispersal of aerosols associated with smoke and fires both in the MOX fuel fabrication facility and in nuclear power plants.

In summary, the ACRS is supportive of the current research activities in fire protection being undertaken by the staff. The ACRS feels, however, that in light of the failure of the attempt by the National Fire Protection Association to produce a useful, performance-based, risk-informed fire protection standard, and in light of results that seem to be coming from the IPEEE effort, it is time for a comprehensive re-examination of the agency's needs with respect to fire protection and the resources it takes to monitor licensees' fire protection programs.

Table 3. Ongoing and Planned Programs in the Area of Fire Protection

	Task Description
W6733	Screening Review of Internal Fires-IPEEE
Y6037	Fire Risk Assessment
W6593	Effects of Aging and Emerging Issues on MOV Performance
Y6041	Assess Debris Accumulation on PWR Sump Performance
planned	Fire Protection for Gaseous Diffusion Plants
planned	Fire Significance Determination Process Support

II.4. Reactor Fuel Research

An important development within the nuclear industry today is the use of reactor fuels to extended levels of burnup. Until recently, 12- to 18-month fuel cycles were common. At discharge from the reactor, fuel burnups were typically about 40 GWd/t. Today, 24-month cycles are becoming common and the burnups of reactor fuels at discharge from the reactor are approaching 60 GWd/t. Some in the nuclear industry expect that competitive pressures will lead to industry interest in fuel duty cycles of up to 36 months and fuel burnups of up to 75 GWd/t.

The NRC had been approving ever higher burnups for reactor core loads. The Current agency policy is to limit the peak rod burnups to 62 GWD/t. This policy stems, in part, from the irradiation experience gained from lead fuel rod testing programs conducted in the 1980s, which involved the successful irradiation of several thousand fuel rods to burnups approaching this regulatory limit. It must be noted, however, that fuel rods in the high burnup demonstration programs of the 1980s were usually placed in nonaggressive, low-power core locations that do not reflect the more aggressive environments that are typical of present day reactors (higher power levels, higher core temperatures, and higher core flow rates for boiling-water reactors). It is not surprising, therefore, that some problems have been revealed for high burnup fuel in recent years. Although the lead rod test experience has demonstrated the feasibility of achieving high-burnup levels under normal operating conditions, there is not a satisfactory demonstration of performance of high- burnup fuel for design basis or severe accidents. For design-basis accidents (DBAs), the available database extends to certainly no more than about 35 GWd/t and is predominantly in the range of only about 17 GWd/t. This regulatory strategy was based on a confidence that there was a thorough understanding of fuel behavior after many years of research sponsored by NRC and that the database developed in the past experimental programs could be extrapolated to higher burnups and higher operating power densities that have become common because no new phenomena would emerge.

Of course, new phenomena did emerge. The discovery by French and Japanese experimentalists that high burnup fuels were vulnerable to modest reactivity insertion events is now well known. Clad embrittlement by hydride formation and the development of a high-porosity, low-thermal conductivity rim on fuel pellets are also well known. Less appreciated are the operational difficulties that have been encountered with high burnup fuel, including axial offset anomalies in the core neutron flux attributed to boron absorption on higher temperature portions of fuel rods and control rod insertion difficulties.

The ACRS regularly reviews the research program on reactor fuel. The confirmatory research program developed by RES is exceptionally well conceived, focused, and executed. It has taken maximum advantage of its limited resources by developing partnerships with the Electric Power Research Institute for the experimental studies of cladding material properties and behavior during LOCAs and other types of accidents, and partnerships with researchers abroad who still have the capabilities to conduct in-pile fuel safety tests. The program is extending the NRC's empirical fuel behavior codes, FRAPCON and FRAPTRAN, to the higher burnups that are allowed today.

The program designed by the staff is very economical. The limitations on resources has meant some approximation have been made. Among these was the assumption that properties of the various types of zirconium cladding now in use could be inferred by investigation of only a subset of these claddings. Recent events have suggested that the embrittlement of cladding may be more sensitive to the alloy composition and clad fabrication methods than supposed when this approximation was adopted. As a result, it may be prudent to consider expanding the testing program to include all the major types of cladding, and even multiple examples of these clads. The ACRS remains concerned that current resource constraints on the program may limit the ability to demonstrate the repeatability and to quantify the experimental errors of the various tests.

The program is constrained by the user need process to its confirmatory role to examine fuel only up to (or close to) currently permitted fuel burnups. We know, however, that the nuclear industry is interested in fuel burnups that are much higher than 62 GWd/t. There are both societal and economic advantages associated with use of fuel to higher burnups. It is the NRC's position that licensees will have to supply all the information needed to justify use of fuel to very high burnups. The ACRS challenged the research staff to find ways within their limited research program to identify definitively the type and quality of data that licensees should make available to support applications for yet higher fuel burnups. Especially important is to identify where experimental data must be used to substantiate predictions of analyses. The research staff responded with a brilliantly conceived program of phenomena identification and ranking. They assembled a prestigious panel of world experts in fuel behavior to carry out this effort. The results of the effort are providing continuing guidance to NRC research and the efforts to develop regulatory guidance concerning higher burnup fuels.

Still, the constraints on the program are significant. A broader range of investigations is merited in light of the societal and economic incentives to use fuel to higher burnups. Furthermore, the NRC should be very wary of again abandoning core competency in the area of reactor fuel. An active, exploratory research program could go a long way toward sustaining this core competency. The following areas could merit investigation:

- Develop predictive models of fuel behavior as a function of burnup capable of predicting rim effects and other new phenomena that become apparent only in reactor fuel that is taken to higher burnups.

- Investigate the degradation of high burnup fuel, which may involve fuel swelling and slumping processes that are not currently considered in the NRC's severe core damage models.

- Investigate fission product release from high burnup fuels which is thought to differ from the release rates from lower burnup fuels that were the basis for the NRC's Revised Accident Source Term as well as the NRC's severe accident models.

- Investigation of boron interactions with cladding, and the propensity of boron adsorption to distort the neutron flux profiles in operating reactors.

The new element in the NRC's fuel research program involves mixed plutonia-urania fuel that the Department of Energy proposes for use in selected light-water reactors as a means for disposing of some of the world's excess weapons-grade plutonium. Existing U.S. fuel regulations do not specifically address the use of MOX fuel, although prior to the U.S. moratorium on MOX, several plants loaded partial MOX cores and LTAs so there has been some U.S. experience using MOX fuel. Following the moratorium in the 1970s, other countries decided to continue to pursue MOX fuel; therefore, experience exists in Europe and Japan on the use of MOX fuel. The fuel used in these countries originates from reprocessed spent fuel and therefore, the isotopic mixture of the plutonium and uranium differs from the MOX fuel proposed for the U.S. mission reactors. The European and Japanese experience consists of operational data and testing of the fuel. This experience and testing has shown that the fuel composition does create some operational and accident behavior differences between uranium and plutonium based fuels. Because the industry in Europe and Japan has not provided data to support a higher burnup limit, the current limit remains at the initial licensing limit.

There are differences between MOX fuel produced from recycled fuel and MOX fuel that is fabricated for burning weapons-grade plutonium, Weapons- grade plutonium is almost entirely Pu^{239}, while reactor-grade plutonium contains a mixture of the isotopes Pu^{239}, Pu^{240}, and Pu^{241}. There are also morphological differences. Plutonium in MOX fuel made from recycled fuel is distributed in a uranium oxide lattice at the atomic scale. MOX fuel made from weapons-grade plutonium will be fabricated by blending particles of plutonium dioxide and uranium dioxide so that there are macroscopic islands of plutonia distributed within a matrix of urania. Because plutonium fissions preferentially, islands of burnup and porosity become distributed in the fuel as it is consumed. The limited testing that has been done suggests that there are substantial differences in the retention of fission products in the matrix of MOX fuel. There are also differences in the mechanical and thermal properties between MOX fuel and conventional uranium based fuels. The biggest differences between MOX fuel and conventional fuels and, indeed, differences between MOX from weapons-grade plutonium and MOX from reactor-grade plutonium, are likely to be neutronic differences. The NRC neutronic models are not currently able to model these differences, and substantial upgrades are very likely to be needed. The NRR MOX user need letter to RES identifies the changes needed to the codes.

It is evident that there are technical issues to be resolved in the safety regulation of MOX fuel for use in light-water reactors. The burden of resolving these issues will fall primarily on the applicant. If experience with high-burnup fuel is applicable, the licensee will provide persuasive evidence of acceptable fuel behavior under normal operating conditions. Paper analyses substantiated with little experimental data will be provided to argue for adequate fuel behavior under off-normal conditions of design basis accidents. Severe accident behavior will not be addressed at all. The NRC will need a strong technical foundation to critically review the licensees submittals to ensure that analyses are both plausible and correct. The NRC research program will have to provide this strong technical foundation.

The research program that the staff will pursue in connection with MOX fuel is only now being defined. Early indications are that it also will make use of the Phenomena Identification and Ranking process that has been successfully applied in the research on high burnup fuel. The program will have to investigate how the fission products differ from conventional fuels. The information will provide the technical bases for specifying accident source terms for the safety evaluation of plants that will use MOX fuel. Additionally, developing the neutronics capabilities identified in the NRR user need memo on MOX fuel to the Office of Research will provide the necessary tools for analyzing a heterogeneous mixed uranium and MOX fueled core. These analytical tools in addition to the collaborative programs that the Agency participates in and the knowledge gained from other regulatory partners who have experience with MOX, will provide the resources and information necessary to make a safety determination on the use of MOX fuel in current LWRs.

Table 4. Ongoing and Planned Research on Reactor Fuel

	Task Description
W6200	Code Development and Analysis for High-Burnup Fuel
W6500	IGR High Burnup Fuel Tests
Y6367	High Burnup Cladding Performance
Y6403	Reactor Core Analysis
planned	Provide reports that serve as the basis for confirming or revising criteria for high burnup fuel in Regulatory Guide 1.77 and Standard Review Plan Chapter 4.2
planned	Develop Agency plan on MOX fuel
planned	Provide updates to the FRAPCON code for predicting fuel performance

II.5. Materials and Metallurgy

The major ongoing and planned research programs in the field of materials and metallurgy are summarized in Table 5. These programs fall into four important topical areas:

- improved evaluation of the risk of pressurized thermal shock
- reactor vessel integrity
- steam generator tube integrity
- environmentally assisted cracking

These programs are discussed in the subsections that follow:

- Pressurized Thermal Shock

The NRC has investigated the susceptibility of reactor pressure vessels to pressurized thermal shock (PTS). On the basis of this research, the agency established a rather conservative screening process that imposes constraints on the startup and shutdown of nuclear power plants. As the plants age and the reactor vessels accumulate more radiation damage, the constraints imposed on the heatup and cooling of the plant become more onerous. It can be anticipated, especially in light of the enthusiasm for license extension, that a point will be reached at which licensees will need to have less conservative pressurized thermal shock constraints on operations. Perhaps in anticipation of licensee requests, the NRC has undertaken a remarkable, multidisciplinary program to re-examine the phenomena and risk associated with pressurized thermal shock. This program brings together the agency's rather considerable capabilities in:

- accident frequency assessment
- thermal hydraulics
- probabilistic fracture mechanics

To prepare best estimate assessments of PTS, special attention to the development of defensible uncertainty analyses of both phenomenological and probabilistic aspects of the research.

The PTS program has been examined several times over the past year by the ACRS and its subcommittees. Although, at first blush, it might appear that the research NRC is doing in this program ought to be done by licensees seeking relief from current requirements and guidance, the NRC research program can be justified as needed to independently evaluate licensee requests for such relief. There is a strong basis for believing that the program will be successful in light of the substantial advances that have occurred in the fields of probabilistic fracture mechanics and risk assessment in recent years. The program is well organized and well-managed, and appears to be progressing toward its goals in a technically defensible way. It may well set a standard by which future multidisciplinary research programs at the NRC are judged especially if the promised detailed quantifications of uncertainties are done successfully. In any event, the program will be of generic interest, since it is the first major application of the risk-informed technologies that the NRC is developing to a rule that has its genesis rooted in adequate protection.

- Reactor Vessel Integrity

Nearly all PRAs that the frequency of spontaneous reactor pressure vessel failure is negligibly low. Regulatory oversight of the reactor pressure vessels is intended to ensure that this low probability is maintained despite the aging of the vessels. The PTS rule, 10 CFR 50.61, provides a conservative screening criterion that effectively limits the level of embrittlement of the reactor pressure vessel. This screening criterion was intended to ensure that the probability of failure due to pressurized thermal shock is less than $5X10^{-6}$ per reactor year. Research on the integrity of reactor pressure vessels has been supported by the NRC since its formation and prior to that by its predecessor agency, the Atomic Energy Commission. Despite this long history of effort, NRR issued a series of user requests in 1991, 1995 and 1997 for additional information to address the issues of pressurized thermal shock, pressure and temperature limits for startup, low Charpy upper-shelf energy, setpoints for low-temperature overpressure protection, characterization of flaw distributions, thermal annealing, fatigue and stress corrosion crack growth, and extended operation under plant license renewal.

The fundamental degradation mechanism in the reactor pressure vessel is embrittlement under irradiation. The technical issues that must be addressed to meet the overall objectives of the aging research program and provide the technical bases for NRR assessments include the availability of adequate fracture mechanics analysis methods, reactor dosimetry methodologies, understanding of the effects of embrittlement on material properties, and the capability to adequately characterize the flaw distribution in reactor pressure vessels.

The NRC research efforts have provided great insight into the regulatory questions arising from reactor pressure vessel embrittlement. The reactor surveillance program, together with Regulatory Guide 1.99, Revision 2, provide conservative estimates of upper-shelf toughness loss and transition temperature shifts in most cases. Fracture mechanics tools have been developed that can be used to demonstrate that embrittled vessels still have adequate upper-shelf toughness. The generic analyses in NUREG/CR-6023 and the submittals from the owners groups demonstrate that all the pressurized water reactors and boiling water reactors licensed by the NRC should have adequate upper-shelf toughness at least through the end of their current licensing periods. If a plant were faced with a PTS limit problem and wished to consider annealing as a solution, models for embrittlement recovery during annealing and reembrittlement rates are available. The technical expertise provided by the program was used successfully to address the recent request by the Boiling Water Reactor Vessel and Internals Project (BWRVIP) for inspection relief for boiling water reactor vessels based on fracture mechanics arguments for extremely low probabilities of failure of circumferential welds.

These results represent a substantial accomplishment in meeting regulatory needs. Today, the reactor vessel integrity research program is addressing regulatory needs in two key areas: embrittlement estimation methods and fracture toughness estimation methods for embrittled materials. While related, these areas address distinctly different approaches to estimating fracture toughness for use in pressure vessel integrity analyses. Such analyses affect continued operation of the power plant in terms of PTS, as already discussed, allowable pressure and temperature limits that govern plant startup and shutdown operations, and the acceptability of flaw indications detected during periodic inservice inspections. These operational considerations have proven to be limiting for some plants during the current license period and will be more so during license renewal periods.

Embrittlement Estimation Methods

The current management of vessel embrittlement relies heavily on materials surveillance programs. When the surveillance program includes appropriate samples of the limiting pressure vessel material, the use of surveillance materials reduces the uncertainty in estimates of vessel embrittlement. Even if new, currently unknown embrittlement phenomena should occur, the surveillance materials would still provide good estimates of the embrittlement. Unfortunately, we do not have sufficient materials in all cases to continue surveillance programs through the license renewal period. Even now, in some cases, we do not have surveillance materials (e.g., sample coupons) of the actual belt line weld material. We must rely, therefore, on surrogate materials and adjust the results through the use of models, which account for the differences in composition between the actual material and the surrogate material. Addressing this issue requires models that include the effects of all pertinent variables and assessment of potential distributions of these variables. Development and validation of these models are important elements of the research program. Although tremendous progress has been made in the development of these models over the past decade, statistical analysis of the current data shows that there are long-term irradiation/aging phenomena occurring that are not accounted for in the current models. This could indicate that, at longer times, other mechanisms of embrittlement in addition to the familiar nucleation of nanoscale copper-rich precipitates are occurring. Anticipatory research efforts indicate that as irradiation time increases, other microstructural changes such as the nucleation of copper-catalyzed, manganese-nickel-rich precipitates can produce additional embrittlement.

The surveillance program only provides estimates of the embrittlement at the inner wall of the vessel. For reactor pressure vessel (RPV) integrity analyses, such as PTS, in which a flaw initiates and propagates deeper into the wall, the attenuation of embrittlement is of importance for determining whether cracks that initiate will arrest before penetrating the wall and whether failures will be small leaks or catastrophic ruptures. Calculations of the variation in embrittlement through the RPV wall arising from attenuation of the neutron flux have substantial uncertainties, and more accurate estimates could reduce the conservatisms in estimates of the consequences of crack initiation during PTS events.

Changes in core loading could potentially lead to a significantly harder neutron spectrum at the vessel wall, even if shielded assemblies are used to reduce the total flux. Such a shift in spectrum would require reevaluation of the embrittlement correlations and the throughwall attenuation of embrittlement damage.

Fracture Toughness Estimation Methods

The current methods for estimating fracture toughness use the knowledge of the material chemistry, unirradiated properties, neutron fluence, and the embrittlement estimation methods to adjust generic fracture toughness curves for both crack initiation and crack arrest. Because of the considerable uncertainties in each step in this process, relatively large margin terms are included, making the overall estimate of fracture toughness quite conservative. An alternative approach to determining fracture toughness, the so-called Master Curve (MC) has been evaluated under the NRC's research program and currently has gained significant licensee interest.

This approach has a better physical basis than the current empirical approach of relating the shift in the fracture toughness to the shift in 30 ft. lb. Charpy transition temperature, and could reduce the conservatism that was introduced to address the larger uncertainties associated with the empirical approach. However, a number issues must be addressed in implementing of the MC approach. One is the surrogate material problem noted in the discussion of the surveillance program. Others include the shape of the MC, its applicability to the determination of crack arrest toughness, the use of small specimen tests to integrate MC with the existing database, confirmation that the shape of the MC does not change with irradiation, and a predictive method so that changes in fracture toughness due to through-wall fluence attenuation can be estimated.

Current regulatory requirements for pressure vessels do provide assurance of the vessel integrity. However, the conservatisms in these requirements impose significant regulatory burdens, operational constraints, and large costs on the industry. These conservatisms are detrimental to safety because they may distort the allocation of resources, and because they directly increase exposures to personnel and restrict the operational flexibility of the plant during heatup and shutdown. Although reductions in the conservatisms in the assessment of vessel integrity can and should be made, there is danger in removing some of the conservatisms without an overall understanding of all the sources of conservatism and uncertainty. Excessive conservatism in one aspect may be compensating for nonconservatism in another aspect. The ongoing research program is addressing important issues in our understanding of embrittlement. The understanding of the technical issues associated with vessel embrittlement has reached a level of maturity that justifies a reduction in the effort that has historically been associated with this area. It can be argued that the elimination of any excess conservatism should be the responsibility of industry. However, because of the fundamental importance of vessel integrity for reactor safety and public confidence, the NRC must be able to independently verify that any changes to regulatory criteria will not increase the probability of failure to unacceptable levels.

The ACRS feels that the need for independent verification of regulatory criteria and the technical complexity of the issues require additional information and a level of technical expertise that the agency can maintain only through the continued support of a research program in this area. RES should work with NRR to review the projected needs and ensure that the program is directed to meet those needs.

- Steam Generator Tube Integrity Program

Steam generators have been the most troublesome of the major components in pressurized water reactors around the world. Industry efforts have been successful in managing the degradation of steam generator tubes due to wastage, pitting, and denting. Fretting, stress corrosion cracking, and intergranular attack have proven to be more difficult to control. Even for those plants that have not yet had to replace their steam generators, inspection, monitoring, and repair are very expensive. As a result, there is substantial industry interest in operating pressurized water reactors with modestly degraded steam generators without compromising safety.

To ensure the structural and leak integrity of degraded steam generators, it is necessary to be able to evaluate and characterize degraded tubes, evaluate structural integrity and leakage associated with degraded tubes, and characterize the progress of degradation over future inspection cycles.

The activities of the ongoing research are intended to address these technical issues. The industry has developed inspection technologies, performance demonstration and qualification programs that have improved the effectiveness and reliability of steam generator inspection programs. The research program is carrying out an independent assessment of steam generator inspection reliability through an industry round-robin on a steam generator mockup developed as part of the research program. The mockup contains hundreds of cracks and simulations of artifacts such as corrosion deposits, tube support plates and the like that make detection and characterization of cracks more difficult in operating steam generators than in most laboratory situations. An expert group has reviewed the signals from the laboratory grown cracks used in the mockup to ensure that they provide reasonable simulations of those obtained from real cracks. The number of tubes inspected and the number of teams in the round-robin are intended to provide better statistical data on the probability of detection and the crack characterization accuracy than is currently available from industry performance demonstration programs.

In addition to the round-robin activities, there is also a task on advanced technology for nondestructive examination (NDE) of steam generator tubes. There are two motivations for this work. First, the work is intended to maintain expertise in area so that industry efforts to improve the accuracy and reliability of inservice inspection can be independently examined. Second, the work on advance NDE techniques is needed for the round-robin testing with the mockup which raises the problem of determining the true state of the flaws. Although this can be done by metallographic sectioning and burst and leak testing, such methods are too expensive and time consuming to be used for hundreds of cracks, and a nondestructive assessment of the true state of the flaws is needed.

A tube failure and leak measurement facility has been constructed that can provide prototypical conditions for both normal operating and main steam line break conditions. It is being used to assess industry models for tube integrity and leak rate. The entire progression of the failure from the initiation of a small leak, stable crack growth under increasing pressure, to unstable crack growth and gross rupture can be simulated in the facility. A high temperature tube failure facility has been used to study the behavior of flawed tubes under severe accident conditions.

In most cases, degradation of steam generator tubing by stress corrosion cracking is currently managed by plug or repair on detection, because current NDE techniques for characterization of flaws are not accurate enough to permit continued operation. This is very conservative in many cases, since flaws less than 40% throughwall or even deeper short flaws have very little impact on tube integrity. On the other hand, current inspection technologies and procedures can miss flaws that will lead to steam generator tube ruptures.

The development and qualification of inspection and degradation management strategies are the responsibility of industry. Current regulatory guidance is properly more performance oriented and less prescriptive that it has been in the past. Because steam generator tubes account for well over 50 percent of the primary coolant pressure boundary and failure can lead to containment bypass, it is important to maintain the capability and technical expertise to independent assess industry developments. The program has already made important contributions to the assessment of issues not within the original proposed scope of the program such as thermally induced failure of flawed tubes, behavior of the tubes with Electrosleeve™ repairs, and the potential for tube failure propagation under design basis and severe accident conditions. It is currently addressing the issue of the effect of pressurization rate on tube failure pressures that has arisen from consideration of performance demonstration issues at the Arkansas Nuclear One Unit 2 plant (ANO-2). The NRC must be prepared to expect other similar issues to arise during continued operation with degraded steam generators. Even in the case of generators that have been replaced, laboratory studies show that Alloy 690 is susceptible to cracking in environments that can occur on the secondary side of steam generators.

The recent examination by the ACRS of technical issues arising out of the differing professional opinion (DPO) on alternative repair criteria identified a number of technical issues that need to be further addressed and which should be considered by RES in the formulation of the steam generator research program:

- the effects of the loads and vibration due to the large blowdown forces that occur during rapid depressurization such as would occur following a main steam line rupture on tube leakage and integrity

- the effect of tube support plate movement during depressurization on the tubes

- a better understanding of the heatup of the steam generator tubes associated with the countercurrent flows produced by natural convection under some severe accident situations

- a better understanding of the radionuclide releases associated with iodine spiking

- the DPO issues focused on the use of alternative repair criteria for outside diameter stress corrosion cracking at drilled-hole tube support plates. A more critical issue is whether current inspection techniques in areas of potentially high noise are adequate to identify significant tube degradation. This issue is of particular concern at locations like U-bends and the roll transition region where the tubes are not constrained by the tube support plate and where the additional loads due to blowdown could be even more significant than in the case degradation in the tube support plate region

Although the problems associated with degraded steam generators should greatly diminish as more licensees replace the current models with designs much less susceptible to degradation, assurance of steam generator integrity will remain a significant problem for some time yet.

- Environmentally Assisted Cracking

Since 1967, the NRC (and its predecessor the Atomic Energy Commission) have conducted research that addresses aging of reactor components. Environmentally assisted cracking of reactor structural materials has been a recurrent problem in operating reactors since the mid 1970's. The NRC research in this area was originally initiated to address boiling water reactor pipe cracking problems. Since that time, the focus of the research has shifted to address other problems in environmental cracking such as irradiation assisted stress corrosion cracking. This research has been used to evaluate and establish regulatory guidelines to ensure acceptable levels of reliability for light water reactor components. The products of this program have been technical reports, methodologies for evaluating licensee submittals, and other inputs to the regulatory process. These results have led to the resolution of regulatory issues as well as the development, validation, and improvement of regulations and regulatory guides.

Fatigue and stress corrosion cracking continue to be significant degradation mechanisms in piping systems of nuclear plants. Because pipe failures are included in the DBAs and effective mitigating systems are available, such failures typically do not rank high in terms of contributions to the risks posed by a nuclear power plant. The potentials for such failures have a significant impact on plant operability, worker exposure, and public confidence.

High radiation levels in a reactor core can increase the susceptibility of the core structural materials to stress corrosion cracking because of changes in the water chemistry due to the radiolytic decomposition of water and degradation of the materials themselves. Although many of the affected components can be replaced, replacement is frequently difficult or impractical. Catastrophic failure of some internal components that are needed to maintain core geometry such as the top guide in boiling water reactors could have serious safety consequences, but internal structures tend to be highly redundant and tolerant of cracking. But again, failure in such components has a large impact on plant operability, worker exposure and public confidence. Currently, NRR has accepted crack growth rate curves only for materials subject to fluence levels less than $5x10^{20}$ n/cm^2. For materials that have sustained higher fluences, they are forced to assume crack growth rates thought to be very conservative.

To ensure the integrity of the reactor coolant system boundary and the reactor internals, susceptible materials and conditions must be identified, the effectiveness of mitigating measures demonstrated, and crack growth rates must be determined to ensure that selected inspection intervals are adequate to ensure integrity.

The industry has a very substantial research effort in this area to address these issues. Much of the work is currently done through broad industry cooperative efforts such as the BWRVIP and the modifications/rework package (MRP), but individual owners groups and vendors sponsor additional work to address problems that are unique to a smaller class of systems or are proprietary such as the Noble Chem process to reduce susceptibility to irradiation assisted stress corrosion cracking in boiling water reactors.

The objective of the NRC research is to provide sufficient data and physical understanding to enable the NRC staff to assess the validity of industry analyses of the likelihood of core internal component degradation and failure as a result of irradiation assisted stress corrosion cracking, to evaluate licensee submittals concerning inspection and remediation and the industry models used for establishing inspection intervals and repair criteria. The results support licensing decisions related to operating plants and license renewal reviews. Currently, three aspects of environmentally assisted cracking are being investigated:

- fatigue of reactor structural materials
- irradiation assisted stress corrosion cracking
- environmentally assisted cracking of nickel-based alloys and weld metals

The NRC effort is leveraged through participation in the Cooperative Irradiation Assisted Stress Corrosion Cracking Research Program managed by the Electric Power Research Institute which includes vendors, utilities, and regulatory authorities from the U.S., Europe, and Japan. Participation in the program also provides access to industry irradiation programs in the BOR-60 reactor in Russia. Each member contributes funds, which are used for joint research work as well as in-kind results from their own research programs. The NRC contractors and staff have also worked with the Pressure vessel Research Council in their efforts to develop a consensus position on the importance of environmental effects on the fatigue life of reactor materials. The experimental studies on irradiation assisted-stress corrosion cracking depend on tests being performed at the Halden reactor as part of the NRC participation in the Halden Project.

Maintenance of the integrity of the reactor coolant boundary system is a fundamental aspect of defense in depth for reactor safety. It is clear that the primary responsibility to demonstrate that degradation mechanisms can be managed adequately belongs to the industry and they do appear to making the necessary investments to provide this information. The NRC research programs in this area are substantially leveraged through cooperative international programs and provide an independent capability that is important to maintaining public confidence. Dealing with degraded components is not a prospective problem. Staff must make judgments now on the operability of degraded components. The ongoing research programs help to maintain a technical capability to ensure that such judgments are valid and defensible. The additional information provided by the programs and the industry efforts will lead to more effective and efficient management of the degradation that occurs.

Table 5. Ongoing and Planned Research Programs in Materials and Metallurgy

	Task Description
Y6095	Underwater Welding of Irradiated Stainless Steel
W6275	Assessment of Reliability of Ultrasonic Testing and Nondestructuve Evaluation Methods
W6953	Heavy-section Steel Irradiation (HSSI) Program
W6878	Statistical analysis of Fracture Toughness Behavior of RPV's
Y6127	PTS Risk Assessment
Y6193	Probabilistic Fracture Mechanics for Revision of PTS Rule
Y6249	Analysis of BWR Reactor Vessel Metallic Samples
W6610	Environmentally assisted cracking of LWRs
W6631	International Pressure Vessel Technical Cooperation Program
W6212	Radiation Embrittlement Damage Analysis and Predictions II
W6986	Fracture mechanics technology for LWR Materials
W6487	Steam Generator Tube Integrity Program
planned	Develop Regulatory Guide for Implementing Leak Before Break Concept for Pipes
planned	Develop Thermal Hydraulic Conditions to assess Steam Generator Integrity
planned	International Cooperation on Environmentally Assisted Cracking
planned	Generic Flaw Density and Size Distribution for Reactor Vessel Welds

II.6. Thermal-Hydraulics Research

The ongoing and planned research programs in the area of thermal hydraulics are summarized in Table 6. The centerpiece of the research on thermal hydraulics at the NRC is the consolidation of the various thermal hydraulic codes into a single NRC code. Through its Subcommittee on Thermal-Hydraulics Phenomena, the ACRS reviews this code consolidation effort regularly. The ACRS believes that this code consolidation is an essential activity that has been well-planned and is progressing very well despite the difficulties that the NRC has had in staffing and funding this work. Should this effort not fully meet its goals in the planned time frame for the work, the ACRS believes that this simply reflects the inexact nature of research planning and not the exceptional diligence and skills of those conducting this effort to provide a consolidated code.

The ACRS strongly supports the NRC having its own thermal-hydraulic code. Because of the evolving state-of-the-art in this technical field, independent calculations must be made for reactor transients using a code that is familiar to the analysts. The availability of a familiar thermal-hydraulics model makes it possible for the staff to independently assess applicant predictions, investigate the effects of assumptions and approximate physical modeling, discover sensitivities and weaknesses, incorporate improvements in the analytical and solution methods, and provide predictions in interpretations of plant transients.

A second important component of the ongoing thermal hydraulics research is the contribution to the staff initiative in reassessing pressurized thermal shock. This initiative is discussed further in this report in the Section II-5 dealing with Materials and Metallurgy. Suffice here to say that thermal hydraulics is an important element of the comprehensive reexamination of pressurized thermal shock being undertaken by the staff and that the ACRS fully supports this work.

Several of the ongoing research programs involve maintenance and user support for thermal-hydraulic codes that will eventually be replaced by NRC's consolidated code. In general, the ACRS believes that maintenance of mature codes should be done by the NRC staff and not be a contracted activity. In the particular cases of the thermal hydraulic codes, the ACRS understands the maintenance of other thermal hydraulic codes such as RELAP5 and TRAC-BWR to be essential interim measures taken to meet the needs of the NRC line organization until the consolidated code is available for routine use by these organizations.

The following paragraph present the ACRS comments regarding other ongoing thermal-hydraulic research programs:

- PUMA Integral Test Facility

RES has a strong desire to keep at least some of the ever decreasing number of thermal-hydraulic test facilities viable. The Purdue University Multidimensional Integral Test Assembly (PUMA) facility is the only BWR test facility available to the NRC that can model the interaction between the containment and the reactor coolant system, which has some importance in the analysis of event scenarios involving an anticipated transient without scram (ATWS) and other BWR instabilities. Understanding of such issues will become more important as the industry moves to use fuel with higher burnups, increases the power of plant operation, and adopts best estimate safety analyses. It is essential that the NRC better define the role that the PUMA facility can play in improving the understanding that the NRC has of these issues, and the independent validation of licensee's contentions on the basis of proprietary data. It is important to develop and evaluate the tradeoffs between continuing to use the PUMA facility and developing a better facility that is more suited to the agency's research.

- Oregon State University Integral Test Facility

This facility is being used to address particular, current issues in thermal hydraulics. However, the future maintenance of this facility needs to be better justified in terms of specific needs to improve the NRC's understanding of thermal-hydraulic issues and to validate and improve its consolidated thermal- hydraulics code.

- Rod Bundle Heat Transfer

Research at the Pennsylvania State University into rod bundle heat transfer is intended to provide a mechanistic model of core reflood. Uncertainties in current, empirical models are too high to support the NRC efforts to make requirements in 10 CFR 50.46 risk informed. Uncertainties in the current reflood models jeopardize the ability of the NRC to reduce conservatisms in the decay heat model licensees are required to use and this may have collateral effects on regulatory activities associated with spent fuel storage both in water pools and in dry casks. The ACRS fully supports this work and has been quite impressed by its technical quality. It is imperative to coordinate the ambitious data collection in this program with the development of models for the thermal-hydraulic computer codes. Both data collection and model development may have to be modified as computational uncertainties are quantified.

- Two-phase Flow and Heat Transfer for CFD

An emphasis in this work is on the tracking of liquid-vapor interfaces. The ACRS is surprised that such research is needed. Vendors of commercial computational fluid dynamics (CFD) codes have done quite a lot of work on developing means to track such interfaces.

- CFD Code Models

This work is an example of longer term research sponsored by the NRC. The work is undertaken in anticipation of the day when thermal hydraulics codes used for reactor safety assessments will bear a closer resemblance to the commercial CFD codes that are now available. Even now, the agency is finding advantages in the use of CFD models for the analysis and resolution of local thermal hydraulic issues that arise in the regulatory process. The ACRS believes that it is, indeed, important for NRC to maintain a foothold in this important direction in the development of thermal hydraulic modeling. It is, however, also important to ensure that the work being done in this program does not digress into more academic realms rather than remaining focused on the issues of regulatory significance.

- Steam Generator Tube Rupture

This activity is directed at a current regulatory concern. The ACRS supports this work, but wonders if the work is sufficiently extensive enough to finally resolve the issues. The following examples illustrate the particular areas where additional research may be needed:

- shock and sympathetic vibrations of the reactor coolant system accompanying the system depressurization in a main steamline break or steam generator tube rupture

- identification of parts of the reactor coolant system (other than reactor nozzles, surge lines, and steam generator tubes) that are vulnerable to thermally induced failure as a result of countercurrent natural convection under severe accident conditions

- 3-Dimensional Neutronics for MOX Fuel

Neutronic analysis codes that are currently available to the NRC have been tuned to data on urania-based fuel. A number of differences complicate the use of these neutronic codes for the analysis of fuels composed of mixtures of plutonia and urania. Not the least of these is the approximate treatment of the delayed neutron fraction. Consequently, this work is viewed by the ACRS as essential and in anticipation of applications by the Department of Energy and current licensees to burn MOX fuel in power reactors.

- Graphical User Interface for Thermal-Hydraulics codes

The graphical user interface for the RELAP5 code has been completed. Adaptation of the interface to the TRAC-M code will be completed by the end of FY 2001. The program will become, then, a maintenance and improvement activity.

Future Directions in Thermal Hydraulics Research

It is evident that in the future more realistic thermal-hydraulics analyses rather than highly conservative, bounding analyses will be used in the safety assessment of power reactors. In order to support this evolution and the evolution to a more risk-informed regulatory process, the NRC staff needs to address two questions:

- When is the realistic estimate obtained from a code good enough?

- How uncertain is this estimate? More specifically, could the probability of exceeding some limit be unacceptable? That is, what are the quantitative risk implications in the uncertainties of the predictions obtained with the code?

At present, there are no logical quantitative approaches to answer these questions. They are, instead, answered by the use of someone's judgment. Already there have been examples of different staff members having quite different judgments on these questions and there may even be differences in the judgments of the NRC staff and its management.

The staff needs a more logical and defensible approach to answering these two questions. Answers to thermal-hydraulic questions will be acceptable when the bias in the predictions of key parameters is small enough and when the uncertainty of the combined bias and deviation of the predictions is small enough. What is enough in this regard depends on the particular regulatory decision that is to be made. Both the measures of uncertainty and the criteria for acceptance depend on the risk significance of the answer. They also depend on how closely some limit is to be approached. The smaller the margin, the smaller is the acceptable uncertainty. Therefore, the modeling uncertainty needs to be determined and must be quantitatively related to the risk predicted in probabilistic risk assessments.

The present approach to uncertainty analysis is often just the variation of some selected coefficients in the code and descriptions of how answers are changed as a result of these variations. This depends on the selection of suitable sets of coefficients for variation and suitable ranges for the variations. The process does not address the effects of the code itself, the form of the equations solved by the code or the solution method adopted in the code.

In any case, it is not enough to estimate uncertainty entirely by theoretical means. There must be some comparisons to meaningful data. Often, there is a very small set of data chosen for code assessment. For example, code assessment may involve one comparison with a single loss-of-fluid test (LOFT) facility and a comparison with a BETHSY test. The predictions are seen to be not far from the data and it is guessed that this is an adequate demonstration of code viability for predictions that may not be closely related to the tests chosen for the comparison.

The ACRS does not believe that thermal-hydraulic aspects of risk-informed regulation can proceed very far until the questions posed above can be satisfactorily addressed. It will take research to develop these answers, and the main thrusts of this research should include the following:

- Define the need for evaluating of the adequacy of the outputs of realistic thermal hydraulic codes. Specify what needs to be determined about the realism, accuracy, and other qualities of the code predictions. Develop quantitative measures of the performances of thermal-hydraulic codes.

- Develop tools for calculation of these quantitative measures for use by the NRC line organizations.

- Develop means to assess these quantitative measure from data obtained in separate effects tests and integral response tests. Find ways to reduce arbitrariness and excessive reliance on judgment in the assessment of code predictions of experimental results.

- Use the consolidated NRC thermal-hydraulics code to develop worked examples demonstrating that the above have been accomplished. This may involve the restructuring of the codes, perhaps to incorporate uncertainty evaluations into the code itself at fundamental levels rather than as responses to user inputs to the code.

- Ensure that outputs of the work is in forms best suited to the users in the NRC line organizations.

- Explicitly address model and parameter uncertainties.

Table 6. Ongoing and Planned Thermal Hydraulics Research Programs

	Task Description
W6240	Boron Mixing Experiments
W6245	TRAC-P Maintenance and Consolidation
W6698	PUMA Integral Test Facility
W6699	Oregon State University Test Facility
W6855	Rod Bundle Heat Transfer
W6995	Two-phase Flow and Heat Transfer for Computational Fluid Dynamics
W6996	Development of Models for NRC's Computational Fluid Dynamics Code
planned	Continue development and upgrading of NRC's thermal/hydraulic code
planned	Develop measures of uncertainty in code predictions
planned	Develop quantitative measures of code assessment
planned	Perform experiments in support of development of models which can significantly improve code capabilities
planned	Define the functional requirements for realistic codes in response to anticipated regulatory needs

II.7. Severe Accident Research

The risk to the public by nuclear power generation arises if accidents progress to a point that results in fuel degradation and the release of large quantities of radioactive material into a plant's environment. The NRC has invested heavily in the investigation of severe reactor accidents and has developed an integrated, systems-level computer code for the analysis of severe reactor accident progression. Research on severe reactor accidents has been curtailed substantially in recent years. The remaining severe accident research programs are listed in Table 7. Summary comments concerning these programs are provided immediately below. These summary comments are followed by a discussion of the continuing needs for severe accident research.

The following subsections summarize the ACRS comments regarding the ongoing severe accident research programs:

- MELCOR Code Development and Assessment

The NRC has done a good job consolidating the diverse results of its research into an integrated, systems level analysis tool called the MELCOR code. The continued development of this code is needed as research results from worldwide programs on severe accidents, such as the PHEBUS-FP program and the planned ARTIST program, become available. The ACRS believes that the following topical areas (among others) should be included in the continued development of this code (as further discussed below):

- degradation of high-burnup and MOX fuels with hydrided cladding, and the effects of fuel swelling on damage progression
- aerosol transport into and through steam generators during bypass accidents
- improved modeling of natural convection within pressurized reactor coolant systems during severe accidents
- modeling of loop seal clearing and reformation during severe accidents
- effects of air ingression on residual fuel behavior following vessel rupture in a severe reactor accident

There are many opportunities being made available worldwide for the testing and evaluation of the MELCOR code. These opportunities consist of International Standard Problems that involve the comparisons of code predictions to results of tests. The NRC participation in these exercises has been limited by the availability of funds. The NRC could benefit through participation in international programs as a way of validating its codes. The ACRS believes that resources should be made available for more participation in these international collaborative efforts.

- MACCS Maintenance and User Support

The MELCOR Accident Consequence Code System (MACCS) is a computer model of the dispersal of radioactive material in the environment and the health consequences of such dispersal following a severe reactor accident. This is clearly a calculational capability that the NRC needs to have. The ACRS believes that as a general rule, the maintenance and user support for a mature code like MACCS should be done by the NRC staff and not as a contracted activity. The current program may be needed as an interim measure since the MACCS code is becoming outdated relative to accident consequence models developed elsewhere in the world. Additional research to improve MACCS may, then, be needed to support the agency's move toward a risk-informed regulatory system.

- Probabilistic Consequence Model Development

This program is intended as a contribution to a collaborative European program on the use of expert elicitation to develop distributions for uncertain inputs to models of the dispersion of radioactive materials and the consequences of this dispersion. This is a crucial element in determining the uncertainty in estimates of risk associated with the use of nuclear power generation.

The ACRS believes there to be a strong justification for this research program. The research will augment the study of uncertainties in risk estimates begun with the NUREG-1150 study of five representative plants.

It will also assist in the identification of areas where the MACCS code needs major improvement in order to reduce the uncertainties in the calculations of accident consequences. This program also fits nicely into the need for NRC to maintain a level of participation in pertinent, international cooperative research programs.

- Analytical Support for CONTAIN Code Development

This program is intended to provide maintenance and user support for a mature code dealing with containment phenomena under accident conditions. Line organizations at the NRC make some use of this computer code. Again, the ACRS believes that maintenance and user support for a mature computer code should be done by the NRC staff and not as a contracted service. Furthermore, the CONTAIN code has become anachronistic since the maturation of the MELCOR code. The agency should have a strategy to wean line organizations away from the use of the CONTAIN code, encourage the use of the more modern code MELCOR, and promptly eliminate the need to provide further user support for the CONTAIN code.

- OECD Lower Head Failure Program

The objective of this program is to characterize the timing, size and location of the failure of the reactor vessel lower head under severe accident conditions when this head is exposed to molten core debris. The program was particularly pertinent to the NRC when examining the safety of the proposed advanced light water reactor AP600 when in-vessel retention of core debris was part of the safety strategy for this reactor. This safety strategy was abandoned and the program is now justified as supporting the assessment of severe accident management strategies related to ex-vessel fuel coolant interactions and direct containment heating.

Although this program is part of an international collaborative research agreement, the ACRS believes the research is not needed by the NRC. There are no safety strategies being proposed now that involve invessel retention of core debris. The ACRS believes that severe accident management strategies should not be dependent on the precise rupture behavior of reactor pressure vessels. The ACRS recommends that this research be brought to a prompt and orderly conclusion.

- Technical Support for Molten Fuel Coolant Interactions Research

This program provides the technical support for the development of instrumentation capable of mapping fuel-coolant mixing regions in order to characterize coarse breakup and fine fragmentation of core debris suddenly immersed in water. This breakup and fragmentation is the first step in a process that leads to explosive interactions of molten core debris and water. The current understanding of energetic interactions of molten core debris with water is not sufficient to predict the energetics of interactions expected to occur during severe reactor accidents. Research would be useful to provide a fuller predictive capability. The research needs for molten core debris interactions with water include large-scale tests with prototypical materials. Instrumentation being developed in this program would not likely be useful in such tests. Worldwide, the experimental investigation of fuel-coolant interactions has been abandoned. The need for this program should be reevaluated.

Future Directions in Severe Accident Research

With its risk-informed initiatives, the NRC is engaging in a major shift in the way that it determines the regulations and conducts its regulatory activities. This shift requires a good capability to determine risk and its associated uncertainties. Risk is dominated by severe accidents involving core degradation and the massive release of radioactive materials. It is then expected that severe accident analyses should play a prominent role in the agency's risk-informed initiatives. Currently, however, the agency relies on CDF and LERF as risk metrics. It is thought that these metrics and their uncertainties can be readily determined using available PRA technology and the current understanding of severe accidents. Because of this, there is a belief that sufficient severe accident research has been done. This belief persists because current PRAs require very little in the way of severe accident input to determine CDF and require only knowledge of those severe accident phenomena that lead to early containment failure to determine LERF. The need is to determine the containment loads due to blowdown dynamics, hydrogen production and combustion, direct containment heating, and to evaluate the mechanical response of containment to these processes. It is believed that there is sufficient severe accident knowledge to do this.

The ACRS does not share this belief regarding the current adequacy of knowledge concerning severe accident processes. First, the determination of CDF requires input on the success criteria for the emergency core cooling system to successfully terminate accidents before they can progress to extensive fuel melting. Current success criteria are based on information developed for fuel with burnups less than about 38 GWd/t. The trend in the nuclear industry is to push for much higher burnups certainly to 62 GWd/t and perhaps as high as 75 GWd/t. There is evidence that the flow blockage behavior of such high-burnup fuel under accident conditions differs significantly from the present understanding, which is founded on studies of lower-burnup fuel. This raises questions concerning the beliefs regarding what constitutes a success path during an accident. Research on the behavior of high-burnup fuel under accident conditions is needed.

With respect to LERF, the severe accident issue is not so much the determination of the risk metric but what constitutes an acceptable value of this metric. The LERF acceptance criterion discussed in Regulatory Guide 1.174 is a surrogate for the prompt fatality safety goal. This LERF surrogate value has been determined from level III PRA consequence analyses that have been based on the fission product releases in steam expected from modest-burnup fuels.

There are a number of issues associated with the LERF risk metric that should be addressed with continued severe accident research. First, the current LERF acceptance values are based on the release of fission products in steam and do not consider the changes in the source term that might accompany air ingression following rupture of the reactor vessel. Air ingression is expected in many reactor accidents and is expected to alter the fission product release from residual fuel within the reactor vessel. The alterations caused by air ingression may make the releases more hazardous to the public than what was considered in the development of the current LERF acceptance criterion.

Second, there is evidence that the release of fission products increases with increased burnup and that reactivity insertion accidents of lower energetics than currently evaluated in PRAs will make significantly higher contributions to risk. Neither of these is accounted for in the current LERF acceptance criterion.

Finally, as margins are eroded and the LERF acceptance criterion is more closely approached, it is likely that there will be a need to determine site specific values of risk acceptance criteria. Such site-specific determination will require that greatly improved atmospheric dispersion models be available.

- Countercurrent Natural Circulation in the Reactor Coolant System

Several important severe accident phenomena such as direct containment heating, steam generator tube rupture, and invessel fuel coolant interaction have been relegated to low frequency based partly on the expectation that natural convection flow will fail parts of the reactor coolant system and the system will depressurize. The thermal hydraulic calculations that are the basis for this expectation are inappropriate for two-phase natural convection flow and do not include any determination of uncertainty so that probabilities can be attached to the speculated system failure that leads to depressurization.

- Electrostatic Charging and Aerosol Physics

The consequences of severe reactor accidents are mitigated substantially by the tendency of aerosols of radioactive materials to agglomerate and deposit along flow pathways and in the reactor containment. The NRC has invested heavily in the development of technology to predict agglomeration and deposition of aerosols. The models that have been developed do not account for the real possibility that aerosols in a radiation field will be electrostatically charged. Such electrostatic charging will create interparticle forces that are far stronger than those that are now modeled and could significantly alter the tendencies for aerosols in reactor accidents to agglomerate and deposit. The effects of aerosol charging are not likely to affect estimates of the LERF risk metric. Charging could mean that fission product aerosols will remain airborne in containment much longer than is currently predicted. If so, this will impact the calculated consequences of latent fatalities and land contamination for late containment failures and may even make these the limiting consequences. Research is needed to better understand the potential for electrostatic charging of aerosols under accident conditions and the consequences of charging.

- Bypass Accident Consequences

High risks are associated with severe reactor accidents that involve bypass of the reactor containment such as accidents involving the rupture of steam generator tubes. High risks are ascribed to these accidents because of the limited source term mitigation thought to occur along flow pathways in bypass accidents such as flows through the secondary sides of steam generators in pressurized water reactors. Models that have been used to predict such low mitigation are not especially sophisticated. A cooperative research program to experimentally investigate the potential mitigation of radioactive material releases through steam generators is being proposed by the Swiss. The ACRS believes that NRC should join in this collaborative program called ARTIST. Based on the results obtained in this program, the NRC could decide if improved models of source term mitigation along containment bypass flow pathways are needed in its accident analysis codes.

Core Competencies

The ACRS believes that it is essential that the NRC maintain some core competencies in the area of severe reactor accidents. In particular, the NRC needs competencies in the following areas:

- the behavior of aerosol and the chemistry of fission products under reactor accident conditions
- reactor fuel degradation under accident conditions.

Competencies in these areas will be increasingly difficult for the agency to obtain by contract. Maintenance of the competencies will not be achieved by reliance on user needs. Continued participation in the international collaborative PHEBUS-FP program and its successors will maintain these competencies. This participation has become passive in the sense that NRC attends meetings and receives results of the work. It is not now actively using its models and codes for the analysis and planning of the tests in the PHEBUS FP program. The ACRS believes it important for the maintenance of core competencies that resources be made available for more active participation by the NRC in this program.

Table 7. **Ongoing and Planned Severe Accident Research Programs**

	Task Description
K6987	Cooperative Agreement with UCSB
W6203	MELCOR Code Development and Assessment
W6231	MACCS Maintenance and User Support
W6352	Probabilistic Consequence Model Methods Development
W6758	Analytical Support for CONTAIN Code Assessment
Y6058	OECD Lower Head Failure Program
Y6073	Scientific Software and Data Distribution
Y6232	Technical Support for Molten Fuel Coolant Interactions Research
W6199	OECD RASPLAV
planned	Provide NRR with consolidated severe accidents codes
planned	Provide NRR with technical assistance in the review of the MAAP Code
planned	Provide improved severe accidents codes

II.8 Civil, Structural, and Seismic Research

The NRC has heavily invested in testing the capabilities of containment designs and understanding the seismic threats to containment and piping integrity. Much of this work has involved collaborative efforts with partners from abroad. Like severe accident research, many of the technical areas addressed by this work are not essential for the agency's current efforts to develop risk-informed regulatory practices, using CDF and LERF as metrics. There is not now an active effort to construct new plants that would require state-of-the-art capabilities to review structures and design capabilities though there are at least plans to examine the issues of the AP1000 nuclear plant containment design and more innovative containment designs that are being proposed. Over the next several years, however, it is likely that the pertinent areas of most interest for the NRC mission will involve the aging of structures, systems, and components (SSCs) of the existing fleet of power plants. There are needs to better understand the capabilities of partially degraded SSCs to withstand accident loads. Such an understanding will be important in the evaluation of licensee programs of aging management and to assess the efficacy of inspections.

Ongoing and planned civil, structural, and seismic research programs are listed in Table 8. Several programs in this table allow the NRC to achieve the following objectives:

- leverage the benefits of a much larger investment by partners in collaborative projects
- keep receiving seismic data from organizations such as the US Geological Survey, and - maintain core competencies and participate in industry activities

These considerations are significant enough to justify some level of continued investment in this area over the next few years. The programs have been grouped into categories for comment here:

- Work to assess the structural capability of containments and piping components experiencing varying degrees of age-related degradation and to validate the analytical tools to resolve related regulatory issues and concerns:

  - W6042 Capacity of Aged/Degraded Containments
  - W6043 Inspection Aged/Degraded Containment
  - W6164 Validation of Degraded Methods
  - W6775 Integrity of Nuclear Piping - Structural/Material Issues
  - W6081 Japanese Collaboration on Seismic Issues
  - W6684 Assessment of Aged & Degraded Structures & Components

This work is needed to address the immediate and future regulatory needs generated by the aging of the current fleet of nuclear power plants. In many cases the projects in this category leverage the work of international and national partners that are investing significant resources. Of particular current interest is the international effort (W6775) in response to cracks identified at Summer nuclear plant, Ringhals 4, and Biblis. The ACRS supports these research efforts, but feels a comprehensive plan is needed on research concerning the aging of structures and their abilities to sustain accident loads.

- Work needed to complete major efforts on which significant resources have already been expended:

  W6959 Degradation and Failure Characteristics of Nuclear Power Plant Containment Protective Coatings
  W6233 GV Downhole Seismic Array OP/Analysis of Data
  W6412 Garner Valley Strong Motion Study

  These programs have had value to the NRC and the work should be completed.

- Participation in programs to test the ultimate capability of pre-stressed concrete containments and many seismic projects.

Key motivations for the research in this category include:

- leveraging limited investments to garner the benefits of a more significant effort being carried out by partners
- determining if results of research conducted outside of the nuclear industry can be utilized to improve seismic related nuclear regulation
- maintaining core competencies in the seismic area

These motivations may be significant enough to justify the projects. It must be borne in mind that the uncertainties in both accident loads and seismic loads on containments and structures will be dominated by the uncertainties in the loads. We comment on the various programs in this category of work individually below:

- Y6131 Joint Containment Model Test

This work is part of a collaborative agreement with Japan. The objective is to test to failure the pre-stressed concrete containment built for this purpose by the Nuclear Power Engineering Corporation (NUPEC). The NRC's involvement is to evaluate the predictive capabilities of models used to identify failure modes and margins. It is not apparent how this work will affect the safety regulation of the current fleet of U.S. nuclear power plants.

- Y6251 Collaboration Seismic Proving Tests of Concrete Containment

This work is part of the collaborative agreement with Japan's NUPEC/Ministry of International Trade and Industry (MITI). NUPEC will provide results of a large scale seismic test of a large prestressed concrete containment and a large reinforced concrete containment with liners. The NRC will verify the accuracy of current modeling techniques in design and margins to ensure integrity of containments for seismic loadings. The NRC investment in this program leverages a much larger investment by NUPEC/MITI. It is not apparent what current NRC need(s) the research addresses. The results of this work will help improve the techniques for modeling containments and determining their seismic margins. Certainly, the results and models will not affect the current fleet of plants. Technical issues are sufficiently specific to the peculiarities of containment designs that it is not apparent that the results of the tests and comparisons to model predictions will be of great value in the analysis of containment designs proposed for future reactors.

- 6828 Risk/Performance-Based Response to External Events

This program will review previous efforts by the NRC, American Society of Mechanical Engineers (ASME), American Society of Civil Engineers (ASCE), and the Department of Energy to develop risk-informed and performance-based approaches to evaluate the performance of piping and other structures under the loads imposed by seismic events. The objective is to risk-inform regulatory practices in external event evaluations. The work could support revisions to regulatory guides and sections of NRC's Standard Review Plan. These changes are intended to lead to more uniform margins and to provide a graded approach to design, commensurate with the potential risks. The ACRS views this work as appropriate to support the NRC's move to risk-informing the reactor regulations and helps the staff stay abreast of developments made by industry.

- W6691 Displacement-based Seismic Design

Changes in seismic engineering practices outside of the nuclear industry raise questions of whether NRC seismic requirements in nuclear power plant design or modifications also warrant similar changes. For Example, are displacement criteria more appropriate than site peak ground acceleration for seismic design? The ACRS views this as valuable research that maintains NRC core compe-tencies and anticipates possible changes in design criteria that might be adopted in the future.

- W6166 Cooperative Geoscience Research

This project leverages work by the U.S. Geological Survey to improve the definition of seismic hazard in the central and eastern U.S. This has been an area of controversy and there are two sets of seismic hazard curves being used for risk analysis. Although the ACRS finds this research to be valuable, it is disappointed that this work is not part of a larger effort to resolve the differences between seismic hazard curves developed by the Lawrence Livermore National Laboratory and by the Electric Power Research Institute. Data generated in this program and work the NRC has completed on the methodology for resolving these differences could eliminate an area of continuing concern and discussion.

Future work in this program will maintain cooperation with the U.S. Geological Survey. The work is intended to reduce uncertainties associated with propagation of earthquake ground motion. The ACRS view this involvement as maintaining NRC core competencies in the seismic area and preventing isolation of the regulatory process from developments within the seismic community.

- Tadotsu Shaker Table

The modest investment by the NRC yields a wealth of information from the unique experimental facility that the Japanese have developed. One presumes that this involvement also assures access by the NRC to the facility and the expertise in the event of important regulatory decisions that can be addressed by this experimental capability.

- CSNI International Standard Problem

This program allows the NRC participation in an international standard problem exercise on soil-structure interaction modeling. The ACRS is generally supportive of this type of activity for assessing and improving calculational capabilities including capabilities in the seismic area. The ACRS would be more enthused by the participation if it were part of a bigger effort to ensure inhouse NRC capabilities in the use of seismic analysis tools.

Table 8. Ongoing and Planned Civil, Structural and Seismic Research Programs

	Task Description
W6081	Japanese Collaboration on Seismic Issues
W6251	Collaboration Seismic Proving Tests of Concrete Containment
W6691	Displacement Based Seismic Design
W6775	Integrity of Nuclear Piping - Structural/Material Issues
W6959	Degradation and Failure Characteristics of NPP Containment Protective Coatings
Y6131	Joint Containment Model Test
Y6166	Cooperative Geoscience Research
Y6233	GV Downhole Seismic Array Operation / Analysis of Data
Y6063	Update and Maintenance of Seismic Data Analysis System
W6684	Assessment of Aged & Degraded Structures & Components
J6043	Inspection Aged/Degraded Containment
Y6164	Validation of Degraded Methods
Y6167	CSNI PWG-3 Subgroup Support
W6828	Risk/Performance-Based Response to External Events
planned	Validate analytic tools to resolve concerns about containments that have experienced degradation
planned	Validate tools to resolve issues and concerns about structures that have experienced degradation
planned	Provide confirmatory data on the structural capacity and failure modes for a variety of containment designs up to and including severe accident pressures
planned	Conduct tests of degraded piping in cooperation with international partners
planned	Provide data on concrete containment and other seismic category 1 structures from the Tadotsu Shaker Table
planned	Collaborate with NUPEC on seismic structure engineering
planned	Reduce uncertainties associated with propagation of earthquake ground motion
planned	Examine risk-informed and performance bases approaches implemented in recent ASCE and DOE standards
planned	Develop the technical basis for deciding if changes in earthquake engineering practices warrant changes in NRC seismic requirements for NPP design or modifications
planned	Evaluate available soil-structure interactions analysis techniques

II.9. Research on Digital Safety Systems

The nuclear industry is being forced to change from the past practice of using analog systems for instrumentation and control of nuclear power plants to the use of digital systems simply because replacement parts for analog systems are becoming increasingly difficult to obtain. Digital systems and especially software-controlled digital systems are widely regarded as providing the ability to operate power plants more optimally and to provide enhanced safety of power plants.

Digital systems have greatly different operational and reliability characteristics than the analog systems. Whereas analog systems can process inputs and safety functions in parallel, digital systems must process these inputs and functions in a sequential fashion so much engineering attention must be paid to the timing and scheduling of algorithms used to control the digital systems. Digital systems do not fail typically by 'wearing out'. The unreliability of software-controlled digital systems is dominated by system design errors. In particular, inadequate specification of the system requirements leads to failure of systems to function properly in response to all appropriate inputs. Testing does not provide high confidence that these vulnerabilities do not exist because, it is the unexpected inputs that will not be included in the testing that could lead to the failures.

Because of the peculiarities of reliability of digital safety systems, it is not readily apparent how these systems are to be addressed in quantitative risk assessment methods that are so important for the move toward risk informed regulation, the implementation of the Maintenance Rule and the Revised Reactor Oversight Process.

The NRC staff has emphasized control of the design and development processes rather than testing of the products of the processes to ensure reliability of digital systems. Unfortunately, the standards for controlling the design and development process that are available to the nuclear industry have been developed for the larger digital electronic community and are not focused on the peculiar and usually simpler albeit higher reliability requirements of the nuclear safety community. Furthermore, the nuclear community is not so large that it can greatly influence the digital systems community with respect to hardware and software development. The nuclear industry is often forced to use hardware and software developed for other markets (so-called "Commercial Off-the-Shelf" or COTS systems) for which design and development have not been strictly controlled to the specifications of the standards. Consequently, review of digital safety systems for nuclear power plants is a time consuming process. It is for this reason NRR has asked RES to conduct research that will lead to:

- Improved methods and tools for the review of digital systems
- Risk and reliability models for digital instrumentation and control systems
- Regulatory guidance for emerging technology in the digital field

RES has produced a rather comprehensive plan for research on digital systems through FY2004. The major elements of this program fall in the following four categories (Table 9 shows more detailed breakdown of the research programs).

- Systems Aspects of Digital Technology

This program element is to address 1) environmental stressors such as tempera-ture, humidity, smoke, lightning, and electromagnetic interference to digital systems, 2) digital requirement specifications on the functions of digital instrumentation and control systems and the interfaces of these systems with other plant systems, 3) diagnostics and fault tolerance features that enable digital systems to detect and report internal problems and either avoid or handle these problems, but greatly complicate the systems, and 4) operating systems that control the basic functions of digital instrumentation and control systems.

- Software Quality Assurance

This program element is to develop objective software engineering criteria that provide a measurable acceptance level for software quality for use in the regulatory process. Criteria for software testing are to be defined as part of the software quality assessment.

- Risk Assessment of Digital Instrumentation and Control Systems

This program element involves 1) analysis of digital instrumentation and control system failure data, 2) examination of digital failure assessment methods used by the defense and aerospace industries, 3) identification of the risk importance of digital instrumentation and control systems, and 4) development or adaptation of digital reliability assessment methods.

- Emerging Instrumentation and Control Technology and Applications

This program element involves preparing the staff to consider technological improvements that the nuclear industry may adopt, including 1) online equipment monitoring systems to determine what types of maintenance is needed and when, 2) advance instrumentation for measuring flow, temperature, pressure, and neutron flux that might serve as support for power uprates, 3) "smart" transmitters that can compensate for measurement error or alter sensor functionality, 4) wireless transmission of data from sensors to operators, and 5) "firewalls" to bar accessing computer systems and corrupting or degrading the performance of computer systems.

RES is to be congratulated for preparing such a detailed plan of its activities over the next several years. The plan makes transparent the precise products and uses that are to come from the research. The ACRS offers the following comments on the proposed research programs:

- Overall, the planned work appears to be complete with regard to the digital safety systems issues of concern to the regulatory system.

- The ACRS also supports those aspects of the plan intended to provide improvements to the efficiency and effectiveness of digital safety system reviews by line organizations. If, indeed, the prognostication of improved safety from digital systems are to be had, it is imperative that the regulatory process and the reviews it involves not constitute a significant barrier to the adoption of these systems.

- The examinations of environmental stressors on digital electronic systems focuses on the issues of electromagnetic interferences and gives scant attention to other stressors cited in the plan as possibly affecting the performance of digital systems. Most notably, the plan does not address the issue of deposits of smoke from fires on digital system performance especially long after the fire has occurred.

- The ACRS endorses the staff's questioning of the efficacy of fault tolerant methods and self diagnostics (Program Element 3.2.5) that can be incorporated into digital safety systems. The ACRS is concerned that complexity may make digital systems more susceptible to errors caused by rare events. The increased complexity brought on by the proliferation of features in digital systems is usually done: 1) to improve the utility's plant operations, and 2) to continuously assess the validity of signals being sent to safety systems. The staff should find means to assure that such performance enhancing systems are isolated from digital safety systems to eliminate interactions that might degrade safety.

- The ACRS is pleased by the attention given in the plan to the possibility of removing conservatisms that inhibit the effective use of COTS systems in digital safety systems.

- It is apparent from the plan that the staff is attempting to garner advantage by examining broadly data on digital systems failures experienced within and outside the nuclear community.

- The plan expresses a willingness to develop techniques for qualitative rather than quantitative risk assessment of digital systems and their controlling software. The ACRS believes that to be most useful to the regulatory process the risk analyses must be quantitative. Techniques the staff develops or adapts must be able to be incorporated into the existing PRA technology. It is evident that the staff is struggling in this area. They are certainly searching for existing methods that can be applied to the particular issues confronting NRC. This might be an area where the NRC should make a broader solicitation of ideas, even to the extent of modestly funding multiple institutions and researchers for the competitive development of ideas and approaches, rather than using the approach toward contracting research.

- The ACRS agrees that it is of some importance for the research staff to stay abreast of technological developments in the field of digital instrumentation and control that are likely to be applied to aspects of nuclear power systems that are regulated by the NRC. The staff cannot be expected to stay abreast of all aspects of so dynamic a field as digital systems. This activity should be kept specific to advances that are known to be pertinent to the nuclear community. The topics listed in the plan, predictive maintenance (3.5.2), advance instrumentation (3.5.3), smart transmitters (3.5.4), and wireless communications (3.5.5) do appear to be such pertinent technologies that the NRC research office should stay aware of.

- The plan needs to specify in greater detail what unique technical information the staff will derive from its participation in the Halden project and how this information contributes to the goals of its research and the missions of the NRC.

- During its review of Generic Safety Issue (GSI)-168, the ACRS was shown several low-voltage cables that had been artificially aged to the equivalent of 40 years of operation and then subjected to an environment expected to exist inside containment following a LOCA. The deterioration of the insulation on some cables was quite serious whereas on other cables the deterioration was only modest. Some experts attribute the differences in cable deterioration to differences in the insulation. Others question the use of Arrhenius activation energy as the basis for artificial aging of the cables. Since the behavior of electrical cables in the age range of 40 to 60 years is an important consideration in the license renewal process, better understanding of the issues of cable qualification would seem important. The ACRS is pleased to hear that the NRC is participating in an Interagency working group on cable insulation. The Department of Energy, in cooperation with the Electric Power Research Institute, has undertaken a research program dealing with condition monitoring of both low- and medium-voltage cables. The NRC should establish a connection with this research. Furthermore, the issues of cable aging are of sufficient interest throughout the worldwide nuclear power reactor community that it may be possible to establish international cooperative research activities in this area.

Table 9. Ongoing and Planned Research on Digital Systems*

- Systems Aspects of Digital Technology

  3.2.1 Verify EMI/RFI Qualification Levels
  3.2.2 Complete Environmental Qualification Guidelines
  3.2.3 Lightning Protection Guidelines
  3.2.4 Investigate Requirements Specification Assessment Methods
  3.2.5 Diagnostics and Fault Tolerant Techniques
  3.2.6 Operating Systems

- Software Quality Assurance

  3.3.1 Investigate Software Engineering Practices and Measures
  3.3.2 Investigate Criteria for Software Testing

- Risk Assessment of Digital Instrumentation and Control Systems

  3.4.1 Perform Data Analysis on Digital Instrumentation and Control Failures
  3.4.2 Investigate for Digital Failure Assessment Methods
  3.4.3 Identify the Risk Importance of Digital Instrumentation and Control Systems
  3.4.4 Investigate Digital Reliability Assessment Methods

- Emerging Instrumentation and Control Technology and Applications

  3.5.1 Review of Future Applications for Digital Instrumentation and Control and Research Infrastructure
  3.5.2 Predictive Maintenance/Online Monitoring
  3.5.3 Advanced Instrumentation
  3.5.4 Smart Transmitters
  3.5.5 Wireless Communications
  3.5.6 Firewalls

*The project numbering in this table comes from the excellent plan developed by the staff for the research entitled "NRC Research plan - Digital Instrumentation and Control".

II.10. Spent Fuel Storage and Decommissioning Research

It is evident that it will be some time before a permanent repository for spent nuclear fuel is available to relieve licensees of the continuing burden of storing spent fuel on individual reactor sites. The NRC and the industry have been actively addressing the limited capacity of spent fuel storage pools and have developed the capabilities to store fuel on site in dry casks. These dry cask storage systems have been designed and are regulated based on the best available data. The available database, however, is not extensive. The NRC staff has examined the issues of hydrogen generation, corrosion, and fuel decrepitation, but further work is still needed. The prolonged duration of spent fuel storage on individual reactor sites that can now be anticipated means that these spent fuel systems can pose safety issues for a very long time.

Spent fuel storage pools themselves have become issues especially for plants that are undergoing decommissioning. Although there is now great enthusiasm within the nuclear community over the successful license extension process that the NRC has developed, there is potential that, in the near future, several plants may elect to undergo decommissioning and, of course, all plants eventually will have to be decommissioned. Episodically, there have been recommendations that the NRC's regulations for decommissioning be reexamined and made more coherent. Today, such a reexamination should be done with an eye toward making the regulations more risk informed. Until the spent reactor fuel can be removed from the site, the dominant risk during plant decommissioning comes from accidents involving the spent fuel storage pool. The exact magnitude of this risk is being assessed by the NRC.

Clearly, there are areas within the general topics of dry cask storage of spent fuel and the decommissioning of reactors where research would be of significant aid to the regulatory process. The planned and ongoing research programs in these areas are listed in Table 10. Comments about the individual programs are presented below:

- Spent Fuel Storage Fees

This is a legacy requirement imposed on the NRC involving the continued storage of fuel used in reactor safety research tests conducted at the Idaho National Engineering and Environmental Laboratory. There is, of course, no current options available for relief from this continuing charge against research resources.

- Seismic Behavior of Spent Fuel Storage Cask Systems
- Seismic Criteria for independent spent fuel storage installation

The objective of this program is to provide the technical basis for the evaluation of seismic safety of a dry cask storage system. It is a cooperative program with the Electric Power Research Institute and there is the potential for collaboration with Taiwan and Japan.

The seismic issues of spent fuel casks ought to be fairly easily resolved with existing analytical techniques. Experimental verification of the analyses will be more challenging. The ACRS would be more enthusiastic about this program if it were a part of an overall effort to establish the risk associated with dry cask fuel storage on reactor sites. Such an overall effort would require not only the understanding of the seismic stability of the fuel cask systems, but also the understanding of the consequences of rupture of the fuel casks. Such ruptures could well involve the release of radionuclides in manners different from those made familiar by the analysis of reactor accidents. Of particular concern would be the ignition of cladding as a result of air interactions with zirconium hydrides in clad that had been broken and fragmented in a seismic event. There is great similarity between these concerns and the risk concerns associated with transportation accidents involving high-burnup fuel. There are clear opportunities for coordination of research efforts, as further discussed below.

- Dry Cask License Renewal Criteria
- Characterize condition and material behavior of spent fuel and safety related components in dry cask storage systems
- Support for resolution of fuel integrity issues with dry cask storage systems
- Demonstrate the performance of a cask loaded with spent, high burnup fuel
- Examinations of CASTOR-V/21 Fuel

These seem to be important programs to establish data on the performance of spent fuel storage systems. In the tasks, the experience and the material behavior of spent fuel storage systems at Department of Energy sites will be gathered and evaluated for its implications concerning spent fuel storage systems at commercial nuclear power plant sites. Also, in a joint program with the Electric Power Research Institute, and the Department of Energy, the NRC will examine specimens of fuel that have been housed in the CASTOR-V/21 fuel for about 15 years. Information derived from these examinations will provide technical foundations for license renewal of dry cask storage systems. One of the most exciting elements of these programs is the instrumentation and monitoring of an actual cask loaded with high-burnup fuel. This could be exceptionally useful source of real data on the behavior of the system. The staff faces two challenges in conducting this activity. The first is defining the instrumentation that will be applied to detect phenomena and processes that may not be readily anticipated today. The second is, of course, maintaining the level of interest in the effort for the many years that it will have to be continued to gather the kind of data needed for the regulatory process. The ACRS is supportive of these initiatives that the staff has undertaken to establish a good technical foundation for licensing dry cask storage systems:

- Performance of spent fuel transport casks during severe accidents
- Confirmatory testing for spent fuel transport casks under severe accident conditions
- Evaluate source terms for risk assessment of dry cask storage and transportation accidents

The NRC methods for evaluating risks associated with transportation accidents involving fuel casks are now rather old. They were developed at a time that quantitative assessment of risk was not a pervasive philosophy at the NRC. The assessments of fuel damage during an accident were for fuel that had not experienced the levels of burnup that are now becoming common so that cladding on the fuel was not as embrittled as it is now. Source terms were estimated in quite conservative and bounding fashions. Studies that the NRC has completed have indicated that there are substantial opportunities to improve these risk assessments substantially. The ACRS finds that there is merit in these proposed research activities, but cautions that substantial resources will be required to make improvements in area where the risks to the public health and safety are probably not at all large. The cost-benefit tradeoff of these studies should be examined with some care before undertaking what could be a substantial effort.

- Data and models for assessment of public exposure to environmental releases of radioactive materials from site decommissioning. Develop tools for estimates of radionuclide transport for decommissioning scenarios

The ACRS' enthusiasm for these research programs is tempered by the fact that the programs do not seem to be part of a larger effort to define the risks associated with decommissioning and the fact that with the possible exception of accident scenarios involving the spent fuel pool, source terms from decommissioning events will be small and are not likely to pose significant risks to the public. By themselves, the programs do not seem to hold the promise of benefits commensurate with their costs. If the programs could be viewed within the context of a larger effort to determine the risks of decommissioning and make the NRC regulations on decommissioning risk informed, they might be more readily justified. In any event, these activities would seem to have some features in common with ongoing efforts to address uncertainties and improve the MACCS code (see the discussion of Severe Accident Research in Section II-7).

Table 10. Ongoing and Planned Research Programs in Spent Fuel Storage and Plant Decommissioning

	Task Description
A6893	Spent Fuel Storage Fees
W6829	Seismic Behavior of Spent Fuel Storage Cask Systems
Y6038	Dry Cask License Renewal Criteria
Y6248	Examinations of CASTOR-V/21 Fuel
Y6301	Human Reliability Analysis of Dry Cask Storage Activities
planned	Data and models for assessment of public exposure to environmental releases of radioactive materials from site decommissioning
planned	Characterize condition and material behavior of spent fuel and safety related components in Dry Cask Storage Systems
planned	Demonstrate performance of a cask loaded with spent, high burnup fuel
planned	Evaluate source terms for risk assessment of dry cask storage and transportation accidents
planned	Develop tools to support license termination decisions
planned	Develop tools for estimates of radionuclide transport for decommissioning scenarios
planned	Seismic criteria for independent spent fuel storage installation
planned	Support for resolution of fuel integrity issues with dry cask storage systems
planned	Performance of spent fuel transport casks during severe accidents
planned	Confirmatory testing for spent fuel transport casks under severe accident conditions

II.11. Research in Criticality Safety

Criticality safety is the name given to the arcane aspect of nuclear safety concerned with criticality operation with nuclear materials outside the configuration of a reactor assembly. Activities in refining, enriching, blending, and fabricating fuels for reactors; shipping and storing these fuels prior to exposure in a reactor and the handling; storing, and disposing of spent fuels after use all fall in the realm of criticality safety. In assessing the criticality safety of these operations, heavy reliance is placed on standards issued under the aegis of the American National Standards Institute (ANSI). These standards rely on safe limits on fissile material concentration, total material inventory, and container configuration to guard against criticality. Supplemental use of neutron absorbers and restriction on the availability of moderating materials also plays a role in defining a criticality safe material configuration. The use of these standards along with the so-called double contingency principle (the requirement that an acceptable design is judged to be safe base on independently satisfying two of the safe limits) has proved to be effective.

There is an important point here. Fatal exposures to radiation from fissile materials have occurred in operation with nuclear materials in the USA. These took place at Department of Energy facilities and involved operation with higher enrichment uranium or plutonium. An important qualifier may be placed on these events after careful examination of the circumstances under which they took place. In every case where a fatality occurred, an unusual activity outside the scope of the design of the facility was being attempted. The most common activity of this type was the recovery of high value scrap from fabrication processes. The ad hoc use of equipment to conduct operation without design review has been forbidden form many years now, and there have been no further incidences involving fatalities. Furthermore, a review of more recent criticality events in other countries supports this assessment. When fatalities have occurred, the ANSI Standards were not followed and the requirement for independent safe assessment of new operation was not met.

It must be recognized that for all of the success in using the standards-based approach to criticality safety, the resulting products (designs, procedures, etc.) are quite conservative, and this has meant that they are expensive. Process equipment, storage facilities, and criticality alarm systems designs have all had substantial margin. Shipping rules have also been conservative.

The current criticality safety assessment methods do not have the precision necessary to allow a defensible comparative evaluation of risk to be made to support a choice between competing alternative approaches to safety. They also do not meet the needs of a risk-informed regulatory system. It may well be that the approach to criticality safety should remain deterministic. Unfortunately, there are occasions where a risk-informed assessment could be useful. An example is in the resolving a dispute between NRC requirements and requirements of the International Atomic Energy Agency (IAEA) on spent fuel burn-up credit. The NRC regulations impose significantly less burden on the licensees than do the IAEA requirements. Resolution of this dispute could be a worthwhile goal of NRC research efforts.

A development that will affect the NRC involvement in criticality safety is the interest in using weapons-grade plutonium blended with uranium to make a MOX fuel for use in light-water reactors. Initial efforts are now under way to acquire a batch of the weapons grade plutonium from the former Soviet Union for demonstration of the concept using a light-water reactor operated by Duke Power. A review of the division of safety responsibilities between the NRC and the Department of Energy in conducting this demonstration has led to the decision to have material blending and fabrication done under license from the NRC and compliance with NRC regulations. The blending operation can pose a range of criticality problems, as a batch of plutonium is fissionable with fast neutrons without a moderator present. Indeed, a subcritical mass of unmoderated plutonium can become highly supercritical if it is moderated by accident. The sensitivity of a mixture of uranium and plutonium to the energies of the neutrons will vary as blending progresses. A competent criticality safety assessment of the blending process is needed. The NRC must assess this analysis and may require some experimental verifications with intermediate mixtures that arise in the blending process, While there is no doubt that pressures will be imposed on the staff to review this assessment expeditiously, it is imperative that the review be done correctly. NRC experience with plutonium is limited.

With this background, comments are presented below on the specific tasks in the NRC's criticality research program (See Table 11).

- Development of Criticality Safety Data for Licensing Review

As the availability of experimental facilities for criticality studies is reduced due to concerns about operating costs, it is prudent to assess the available data for completeness in addressing the parameter ranges of current and anticipated interests. It is also useful to test the abilities of computer codes to predict experimental results. Some latitude in defining the experimental data sets that might be needed for these assessments must be accepted.

Modernization of the codes to use the most recent ENDFB cross section files is an important part of this task. This is Version 6 of these cross-sections, the same set that the NRC has found to be most reliable in predicting neutron attenuation of pressure vessel walls. It is reassuring to see the emergence of demonstrated generality in the application of a cross section set to a range of different problem areas.

The task includes an assessment of U.S. data and analysis methods in the resolution of the differences in the NRC and IAEA regulations on allowable burn-up credit. This effort is central to maintaining a capability to address power reactor related concerns in criticality safety. This effort also provides NRC access to the capabilities at Los Alamos National Laboratory to address plutonium criticality safety assessment that will be so important to the resolution of fuel blending concerns discussed above.

The ACRS is supportive of this proposed activity. Some greater detail in specifying the individual tasks and the need for the expected results of the activities would be appropriate. A strategy by which maintenance of criticality codes becomes an in-house activity of NRC should be considered. The value of these activities in future years rests heavily on decisions concerning use of standards-based regulation versus risk-informed regulation in the area of criticality safety.

- Preservation of Criticality Benchmark Data

This task is to collect and archive the documentation of criticality experiments conducted at a number of national laboratories. The data are to be stored at the Criticality Safety Information Resource Center at Los Alamos National Laboratory. Many of the experimenters are now retired and there would be great value in upgrading some of the less expansive entries in the old log books. The ACRS is supportive of the modest contribution NRC makes to this overall effort.

- REBUS Experimental Program

This program is an international effort involving reactivity tests of selected light water fuel bundles for direct evaluation of the burn-up credit. The program thus directly addresses the issues concerning burn-up credit between the NRC and the IAEA. The ACRS supports this program.

Table 11. Research Programs in Criticality Safety

	Task Description
W6479	Development of Criticality Safety Data for Licensing Review
Y6035	Preservation of Criticality Benchmark Data
Y6225	REBUS Experimental Program
planned	Conduct studies on materials criticality safety to preserve criticality benchmark data and provide a methodology for use of codes and update codes

II.12. Radiation Health Effects

Ongoing and planned research activities in the area of health effects of exposure to radiation are listed in Table 12. The first of these, the BEIR VII program, is a continuation of the National Research Council's examination of the stochastic health effects of low doses of ionizing radiation. For some time now, there have been scientists who argued against the linear, no threshold (LNT) hypothesis for extrapolating the stochastic effects of radiation from higher doses where the effects are measurable, but still stochastic, to low doses where there are not reliable data. They have based these arguments on epidemiological studies and analogies to the toxic effects of chemicals. Unfortunately, the epidemiological studies lack statistical power. C. Lamb of the National Cancer Institute has presented persuasive statistical arguments that the sample sizes for meaningful epidemiological studies to resolve controversies concerning the LNT hypothesis are prohibitively large.

Phase 11 of the BEIR VII study will reexamine the database available to support the LNT hypothesis for low linear energy transfer (LET) radiation (gamma and beta rays) and, perhaps, refine our understanding of this hypothesis. (Data supporting the LNT hypothesis for high LET radiation and neutrons is more persuasive.) It will not end the controversy.

The technical basis of the LNT hypothesis is a biological model concerning the reliability of repair mechanisms for damage caused by radiation. Resolution of the debate concerning the LNT hypothesis will be possible only after those opposed to the LNT hypothesis formulate their concepts in terms of a biological model that can be tested with cells and animals. The considerable advances in instrumentation being sponsored by the Department of Energy offer the hope that definitive testing of alternative models will be possible in the coming years.

Several of the programs deal with the collection of occupational exposure data. It seems to be an obligation of NRC to stay abreast of these data.

A most interesting program (Y6112) is the adaption of the RESRAD and RESRAD-BUILD codes by the NRC for site-specific probabilistic dose assessments. RESRAD and RESRAD-BUILD are widely respected computer codes developed by the Department of Energy. The NRC use of these codes seems to be a meaningful step toward the greater use of risk information in dose assessments. Results and insights obtained from the work with these codes may well have some importance to the further development of NRC's MACCS code for accident consequence analysis (See Section II.5).

One of the programs (G6251) provides support to the work of the International Committee on Radiation Protection (ICRP). The ACRS is surprised that there are not efforts focused on determining if NRC should upgrade its radiation protection standards from ICRP30 to ICRP60. The differences between these standards could well be a point of contention in the licensing of a facility for the fabrication of mixed plutonium oxide - uranium dioxide fuel (MOX) assemblies for disposal of some of the world's excess supply of weapons-grade plutonium dioxide. The dose estimates for Pu^{239} in ICRP30 are more conservative than the estimates that come from the use of methods described in ICRP60. Indeed, ACRS expects that the entire accident analysis for the MOX fuel fabrication facility will require the research delineated in Table 12 to put consequence analyses for inhalation of Pu^{239} on a firmer technical foundation especially when the Pu^{239} is mixed with uranium dioxide.

Table 12. Ongoing and Planned Research on Radiation Health Effects

	Task Description
G6071	BEIR-VII
G6143	NEA Information System on Occupational Exposure
W6801	JCCRER Project 2.3 - Deterministic Effects of Occupational Exposure to Radiation
Y6112	Default Parameters for RESRAD and RESRAD-BUILT Computer Codes
Y6133	Collection and Analysis of Occupational Radiation Exposure Data
G6251	Radiation Protection Standards Development
planned	Develop technical basis to support rulemaking in the area of occupational radiation protection
planned	Validate current health effects models
planned	Develop and maintain analytic tools for health effects applications
planned	Develop dose assessment for clearance of materials
planned	Develop radiation protection standards and guidance for use of byproduct materials

II.13. Analysis and Evaluation of Operational Data

From the start of operation of licensed power reactors, value was ascribed to the review of accumulated experience and assessing the lessons to be learned over the broad range of designs and operations, including human performance. These assessments are conducted by the NRC and the industry, and have proven to be very useful both in the design of the regulatory system and the evaluation of its effectiveness.

The Data Collection and Analysis staff reviews the experience from nuclear power plant operations as an essential feature of the effort to protect the health and safety of the public. Plants from the entire fleet of operating plants in the country are monitored to identify vulnerabilities as they develop with time. Using consistent statistical criteria, potentially significant events can be detected rapidly and common problems identified. The assessment of the root causes of events can identify plant system performance, human performance, and externally induced events that warrant special attention by the regulators. Indeed, operating events sometimes disclose unanticipated interactions, phenomena, and behavior. The objective is to develop strategies to avoid event initiation and to mitigate event consequences. An important activity that has impressed the ACRS, recently, has been the effort to evaluate the effectiveness of rules such as the Station Blackout rule and the rule concerning anticipated transients without scram.

Until recently, the activities in this area were the responsibility of the NRC's Office for Analysis and Evaluation of Operational Data (AEOD). The Office's responsibilities were reassigned to the RES. At the time of this reassignment, the ACRS was concerned for the long-term commitment to continue these activities, and the possibility that those doing the data evaluation may lose some of the independence that they enjoyed in AEOD. The ACRS concerns have not been realized, and the work in this area continues at the high standards that it had in the past.

Activities in this program area are carried out in two branches. The Operating Experience Risk Analysis Branch (OERAB) conducts activities involving the assessment of operational data, analysis of data on an industry-wide basis, plant -specific event analyses, and identification and evaluation of risk-based performance indicators that may eventually replace those now in use in the NRC's Revised Reactor Oversight Process. The elements of this range of activities are shown in Figure 1. The Branch has planning that extends well beyond the current budgeting cycle that reflects a comprehensive view of its mission.

The Regulatory Effectiveness and Human Factors Branch examines accumulated experience to address specific questions on plant behavior. Some of the recent and ongoing activities of this branch are listed in Table 13. The Branch is now considering reviews of human performance during steam generator tube rupture events and a review of unnecessary paperwork oriented rules and requirements. The later of these has the potential of providing the basis for ameliorating substantially burdens on both licensees and the NRC staff. The ACRS regularly reviews the products of the two branches involved in the analysis and evaluation of operational data. The ACRS is enthusiastic about the programs now under way in these branches and their plans for the future.

Table 13. Past, Ongoing and Planned Activities in the Regulatory Effectiveness and Human Factors Branch

Routine Events Assessment Activities:

Diablo Canyon electrical bus fire and Loss of Onsite Power event
Arkansas Nuclear RHR pump common mode bearing failure
Hatch scram and failure to open of safety relief valves
Indian Point steam generator tube rupture

Long-Term Studies:

Evaluation of Air-operated Valves
Scram occurring during surveillance and maintenance activities
Regulatory effectiveness of the ATWS rule
Regulatory effectiveness of the station blackout rule
Causes and significance of design basis errors

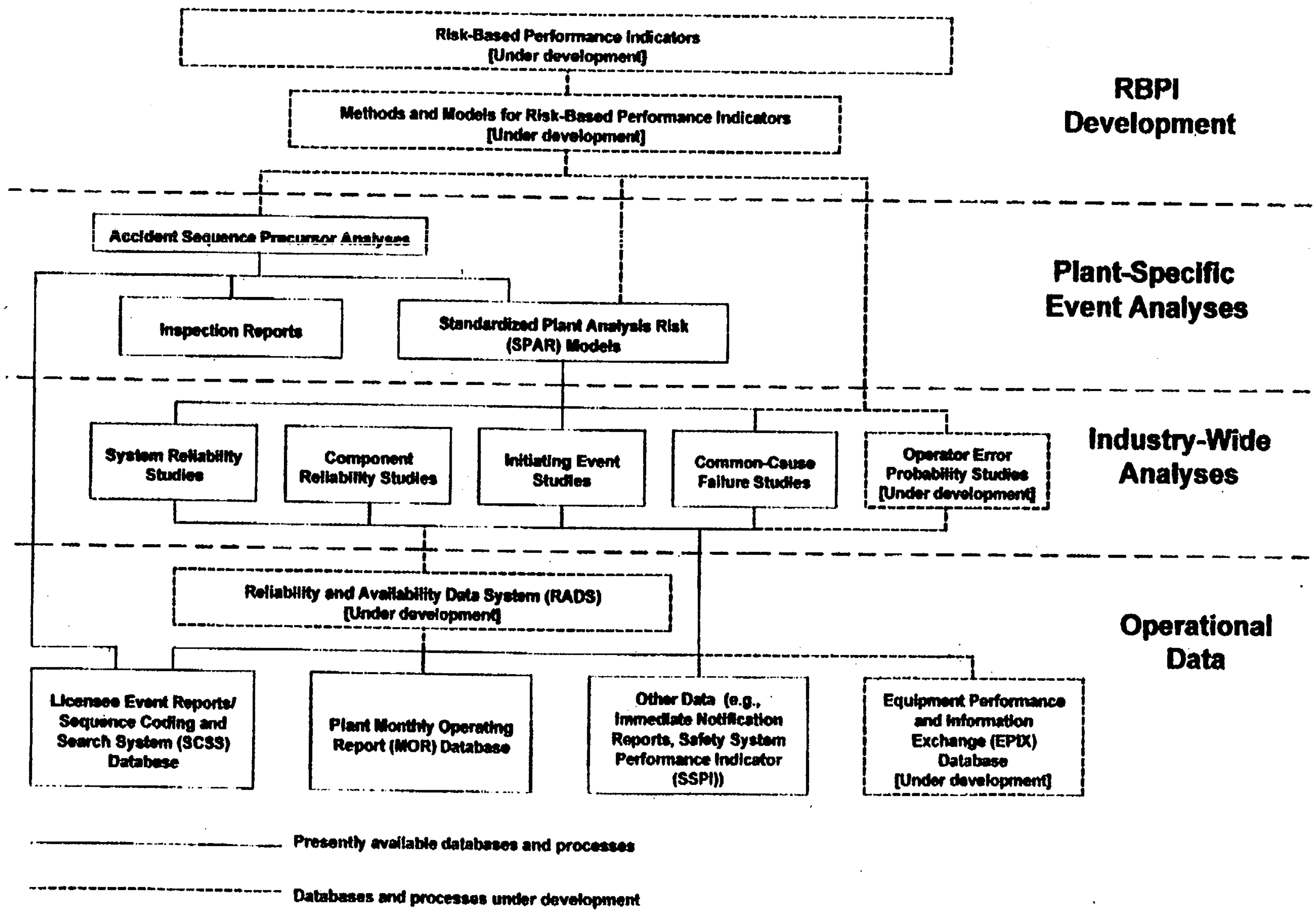
Ongoing and Planned Studies:

Audit of the plant specific backfit process
Regulatory effectiveness of Appendix J — Containment Leakage
Electrical system performance following a reactor trip
Latent failures in human performance

Topics Under Consideration:

Human performance during steam generator tube rupture events
Unnecessary paperwork oriented rules and requirements

FIGURE 2 - OERAB PROGRAMS AND ACTIVITIES



REFERENCES

1. U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, NUREG-1635, Vol. 1, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, A Report to the U.S. Nuclear Regulatory Commission," June 1998.

2. U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, NUREG-1635, Vol. 2, Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, A Report to the U.S. Nuclear Regulatory Commission," June 1999.

3. U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, NUREG-1635, Vol. 3, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, A Report to the U.S. Nuclear Regulatory Commission," March 2000.

4. U. Schmocker, M. Khatib-Rahbar, E.G. Cazzoli, and A. Kuritzky, "An assessment of the risk impact of reactor power upgrade for a BWR-6 Mark -III plant," in: *Probabilistic Safety Assessment and Management*, P.C. Cacciabue and I.A. Papazoglou, Eds., pp. 1842-1847, Springer, London, 1996, (Proceedings of ESREL '96 and PSAM III).

5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

6. Report from D. A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: "Importance Measures Derived from Probabilistic Risk Assessments," February 11, 2000.

7. Report from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Human Performance Program Plan, February 13, 1997.

8. Report from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Final Draft of the NRC's Human Performance Plan, June 12, 1998.

9. Letter from D. A. Powers, Chairman, ACRS, to William D. Travers, EDO, NRC, Subject: SECY-98-244, NRC Human Performance Plan, February 19, 1999.

10. "Summary of INEEL Findings on Human Performance During Operating Events," Report No. CCN 00-5421, transmitter by letter dated February 29, 2000.

11. Memorandum dated March 28, 1997, from John C. Hoyle, Secretary of the Commission, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements-COMSECY-96-066 Research (Direction Setting Issue 22).

12. Staff Requirements Memorandum dated April 15, 1997, from Annette L. Vietti-Cook, Acting Secretary of NRC, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: COMSECY-96-061 - Risk-informed, Performance-Based Regulation, April 15, 1997.

13. National Fire Protection Association (NFPA) 805, "PerformanceBased Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition.

14. U.S. Nuclear Regulatory Commission, NUREG-1150, Vol. 1,"Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," dated December 1990.

15. Committee on Nuclear Safety Research, Revitalizing Nuclear Safety Research," National Academy Press, Washington, DC, 1986.

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1635, Volume 4

2. TITLE AND SUBTITLE

Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program

A Report to the U. S. Nuclear Regulatory Commission

3. DATE REPORT PUBLISHED

MONTH	YEAR
May	2001

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Annual

7. PERIOD COVERED *(Inclusive Dates)*

Fiscal Year 2001

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report to the U.S. Nuclear Regulatory Commission (NRC) presents the findings and recommendations that emerged from an assessment conducted from March 2000 through March 2001 by the NRC's Advisory Committee on Reactor Safeguards (ACRS).

In conducting this assessment, the ACRS sought to fulfill the following objectives:

Assess the ongoing and planned reactor safety research being carried out by the Office of Nuclear Regulatory Research (RES).

Recommend additional research that will be needed to promote the efficient and effective execution of the agency's mission in the future.

The attentions of the report focuses on the research addressing issues of power reactor safety. Research dealing with the permanent disposal of radioactive waste in geological repositories will be discussed separately in a report by the NRC's Advisory Committee on Nuclear Waste (ACNW).

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

Nuclear Reactors
Nuclear Reactor Safety
Reactor Operations
Safety Engineering
Safety Research

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)
Unclassified

(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001