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Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program

A Report to the U. S. Nuclear Regulatory Commission

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

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TABLE OF CONTENTS

ABBREVIATIONS v

EXECUTIVE SUMMARY vii

1 INTRODUCTION 1
- 1.1 Objective 1
- 1.2 Background and Scope of RES Infrastructure Assessment Document 1
- 1.3 Scope and Structure of the ACRS Report 3

2 RES ADVANCED REACTOR RESEARCH INFRASTRUCTURE ASSESSMENT DOCUMENT 5
- 2.1 Generic Regulatory Framework Development 5
- 2.2 Reactor Safety 7
  - 2.2.1 Probabilistic Risk Assessment (PRA) 7
  - 2.2.2 Instrumentation and Control 8
  - 2.2.3 Human Factors Considerations 9
  - 2.2.4 Thermal-Hydraulic Analysis 10
  - 2.2.5 Neutronic Analysis 14
  - 2.2.6 Severe Accident and Source Term Analysis 17
- 2.3 Fuel Analysis 19
- 2.4 Materials Analysis 20
- 2.5 Structural Analysis 22
- 2.6 Consequence Analysis 25

3 PHENOMENA IDENTIFICATION AND RANKING TABLE (PIRT) PROCESS AND IMPLEMENTATION PLAN 27

4 IMPACT OF ADVANCED COMPUTER CAPABILITIES ON NRC STAFF ACTIVITIES 29

5 OVERALL CONCLUSIONS AND RECOMMENDATIONS 31

6 REFERENCES 35

APPENDIX: DOE NEAR-TERM DEPLOYMENT AND GEN IV REACTORS 37

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ABWR	Advanced Boiling Water Reactor
ACNW	Advisory Committee on Nuclear Waste
ACR-700	Advanced CANDU Reactor 700
ADS	automatic depressurization system
ALWR	advanced light water reactor
AP600	Advanced Passive Reactor 600
AP1000	Advanced Passive Reactor 1000
APEX	Advanced Plant Experiment
APWR +	Advanced Pressurized Water Reactor Plus
APR-1400	Advanced Power Reactor 1400
ASME	American Society of Mechanical Engineers
ATHEANA	A Technique for Human Event Analysis
ATWS	anticipated transient without scram
CAMP	Code Applications and Maintenance Program
CAREM	Central Argentina de Elementos Modulares
CDF	core damage frequency
CFD	computational fluid dynamics
CONTAIN	Containment Transient Analysis Program
COTS	commercial off the shelf
CSAU	code, scaling, applicability, and uncertainty (methodology)
DBA	design basis accident
DOE	Department of Energy
EPR	European Pressurized Water Reactor
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
ESBWR	European Simplified Boiling Water Reactor
ESP	early site permit
FLIRA	Future Licensing and Inspection Readiness Assessment
FLUENT	A commercial computational fluid dynamics code
GFR	Gas-Cooled Fast Reactor
GRSAC	Graphite Reactor Severe Accident Code
GT-MHR	Gas Turbine Modular High Temperature Reactor
HC-BWR	High-Conversion Boiling Water Reactor
HTGR	high-temperature gas-cooled reactor
IMR	International Modular Reactor
I&C	instrumentation and control
IRIS	International Reactor Innovative and Safe
ISFSI	independent spent fuel storage installation
LERF	large early release frequency
LFR	Lead-Cooled Fast Reactor
LWR	light water reactor

MAAP	Modular Accident Analysis Program
MACCS	MELCOR Accident Consequence Code System
MACCS2	MELCOR Accident Consequence Code System 2
MASCA	Follow-on program for RASPLAV
MELCOR	Melting of Core program
MSR	Molten Salt Reactor System
NEI	Nuclear Energy Institute
NERAC	Nuclear Energy Research Advisory Committee
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OECD	Organization for Economic Cooperation and Development
OSU	Oregon State University
PARCS	Purdue Advanced Reactor Core Simulator
PBMR	Pebble Bed Modular Reactor
PC	personal computer
PHEBUS-FP	international severe accident fission product research program
PIRT	Phenomena Identification and Ranking Table
PIUS	Process Inherent Ultimately Safe
PRA	probabilistic risk assessment
PRISM	Power Reactor Innovative Small Module
RASPLAV	OECD experimental program for severe accident research
RELAP5	Reactor Excursion and Leak Analysis Program
RES	Office of Nuclear Regulatory Research
RI-ISI	Risk-Informed Inservice inspection
SCWR	Super-critical-Water-Cooled Reactor System
SFR	Sodium-Cooled Advanced Reactor
SMART	System-Integrated Modular Advanced Reactor
SRM	staff requirements memorandum
SSHAC	Senior Seismic Hazard Analysis Committee
SWR-1000	Siedwasser Reactor-1000
THATCH	HTGR thermal neutronics modeling code
TRAC	Transient Reactor Analysis Code
TRAC-M	Transient Reactor Analysis Code Modernized
VHP	vessel head penetration
VHTR	Very High Temperature Reactor System

EXECUTIVE SUMMARY

The objective of this report by the Advisory Committee for Reactor Safeguards (ACRS) is to provide comments and advice on the Office of Nuclear Regulatory Research (RES) document "Advanced Reactor Research Infrastructure Assessment" (Reference 1), which identifies gaps in technology that need to be filled prior to the potential certification of a number of new reactor designs. These designs range from those which have evolved from the current boiling and pressurized water reactor designs, i.e., advanced light water reactor (ALWR) designs, to those using significantly different technology, e.g., high-temperature gas-cooled reactors (HTGRs). The range of designs addressed in this assessment are representative of the designs vendors have expressed an interest in submitting to NRC for certification. We note that the U.S. nuclear industry has not constructed a new reactor for over two decades and has focused on the operational and services aspects of the existing reactor fleet. As a result, there are significant regulatory, management, and technical challenges to be faced with the certification, construction, and deployment of new reactor designs, especially those based on significantly different technology.

This ACRS report concentrates on reactor safety issues. It does not cover items relating to nuclear materials and waste safety or to safeguards. Safeguards will be addressed in a separate ACRS activity. We have not commented on the several management issues that are discussed in the RES Infrastructure Assessment but recognize that these issues are important.

Our overall conclusion is that the RES document comprehensively identifies the gaps in the knowledge that need to be addressed in the review of the ALWR and HTGR designs. We recognize that the applicants are responsible for providing most of this information, but agree with the staff that the NRC should develop its own long-term capabilities to evaluate the applicants' technical arguments that relate to regulatory concerns. In general, the RES document does not specifically distinguish between information that should be developed by the applicant and information to be developed by the NRC. This will be defined upon the completion of the staff's planned Phenomena Identification and Ranking Table (PIRT) activities.

We generally agree with the identification of potential long-term RES projects that need to be considered in the areas of probabilistic risk assessment (PRA), instrumentation and control, materials analysis, structural analysis, and consequence analysis.

We have provided more specific advice as to the direction of the NRC staff work in the following areas:

Generic Regulatory Framework: A risk-informed framework should prove useful for the certification of the plants likely to be first deployed (i.e., ALWRs[1]) and will likely be mandatory for the non-LWR concepts on a longer timeframe. Thus, we recommend that this effort be continued with a focus on addressing our concerns about the current Option 3 framework, which is likely to serve as the basis for the generic regulatory framework for risk-informed certification and licensing of future plant designs.

Human Factors: We believe that the topic of control room staffing will be important for new reactors and the NRC will need to develop defensible methods for assessing the adequacy of staffing levels.

Thermal-Hydraulics: The TRAC-M code plays a central role in the plans for addressing thermal-hydraulic questions for advanced reactors. There is urgency, therefore, to complete its development and validation. Because PRA success criteria for passive designs are sensitive to uncertainties in thermal-hydraulic code predictions, these model uncertainties need to be quantified.

Neutronic Analysis: It is important that the NRC continue to be able to do independent nuclear safety assessments. This will require continued support of the improvement of its neutronic analysis capabilities. In particular, the effort to couple the thermal-hydraulics code TRAC-M with the three-dimensional neutronics code Purdue Advanced Reactor Core Simulator (PARCS) is important for the independent assessment of nuclear safety for reactors like the Advanced Passive Reactor 1000 (AP1000) and European Simplified Boiling Water Reactor (ESBWR) that rely on passive safety systems. Modification of analysis methods to account for the unique features of the Advanced CANDU Reactor (ACR-700) should be initiated now to facilitate timely addressing of certification review issues. The NRC staff should provide explicit direction as to the data and data quality that will be needed to support and validate neutronic analyses for advanced reactor designs.

Severe Accident Progression and Source Term: Issues for reactor designs that are based on LWR technology, such as the ESBWR and AP1000, can likely be addressed through modest modifications of existing codes, but issues related to higher burnup fuels will have to be addressed. Some of the new reactor designs incorporate cavity flooding, which is intended to ensure that the debris produced by a severe accident is retained in the pressure vessel. Significant effort will be needed to assess the effectiveness of this design feature. Consequently, the NRC should continue its involvement in the MASCA project.

It appears that a substantial effort will be required to configure NRC's accident analysis models to treat the unique core and coolant configuration of the ACR-700. The NRC

[1]The term ALWRs in this report includes advanced CANDU designs that utilize a light water coolant.

staff should assess the needed modification of its codes and identify experimental data needed to validate predictions of accident progression and fission product release for the ACR-700.

The NRC staff's experience with accident analysis of gas-cooled reactors with graphite is limited. Existing NRC codes will require substantial modification to be applicable to the analysis of these types of reactors. It will take substantial time to develop adequate tools. For this reason there is a need to identify the available and needed experimental data and to begin code model development.

Fuels: Increased burnup is the primary issue to be addressed with new reactor designs (e.g. AP1000 and ESBWR) where the fuel is similar to the fuel used in currently operating reactors. This issue is being addressed for current reactors and needs to be given a high priority. With regard to the ACR-700 design, it is likely that the NRC can use existing codes and databases to evaluate the fuel using data provided by the vendors, but this needs to be confirmed. Coated-particle fuels are the focus of much of the staff's attention in its Infrastructure Assessment, since there is little NRC experience with these fuels. Because the design certification review for the Gas Turbine Modular High Temperature Reactor (GTMHR) is projected to start in early 2006, we believe that fuel modeling activities (including fission product release) that are anticipated to take substantial time to complete should be initiated soon. In terms of priority, however, due attention should be given to the fact that a fuel design has not yet been established in detail.

Computer Capabilities: Although this topic is not discussed in the Infrastructure Assessment, we recommend that the computer capabilities within RES be reassessed in the light of the increased computational speeds now available. Faster computers would increase efficiency and effectiveness in completing numerous tasks identified in the assessment and in interacting with the applicant who will likely be using these capabilities.

1 INTRODUCTION

1.1 Objective

The objective of this report is to provide Advisory Committee on Reactor Safeguards (ACRS) comments and advice to the Commission, Executive Director for Operations (EDO), and Office of Nuclear Regulatory Research (RES) staff on the "Advanced Reactor Research, Infrastructure Assessment" (Reference 1). This assessment identifies gaps in technology that need to be filled before new reactor designs concepts can be certified and deployed in the United States. Timely planning of RES activities is needed to enable the NRC to meet its obligations for certification of these new reactor designs.

1.2 Background and Scope of RES Infrastructure Assessment Document

In February 2001, the Commission issued a staff requirements memorandum (SRM) entitled "Staff Readiness for New Nuclear Plant Construction and the Pebble Bed Modular Reactor" (Reference 2), with the specific direction to assess the NRC staff's technical, licensing, and inspection capabilities and to identify enhancements, if any, that would be necessary to ensure that the agency can effectively carry out its responsibilities associated with an early site permit application (ESP), a license application, and the construction of a new nuclear power plant. In a partial response to this SRM, the staff issued an information paper in October 2001, "Future Licensing and Inspection Readiness Assessment (FLIRA)" (Reference 3), in which the staff committed to develop an advanced reactor research plan that would be used to develop and guide a comprehensive advanced reactor research program, addressing the regulatory framework and technical challenges that might be encountered with new license applications. As a first step in developing the program plan (in terms of objectives, priorities, milestones, etc.), RES prepared the Infrastructure Assessment document and its attachments to identify the gaps in knowledge. That is the assessment that the ACRS is reviewing in this report.

In 2001 and 2002, a series of actions by the nuclear industry and the Department of Energy (DOE) influenced the content and timing of the research that is discussed in the Infrastructure Assessment document.

Namely:

- DOE and the Nuclear Energy Institute (NEI) conducted a wide-ranging review of the infrastructure that would be required in order to significantly increase the Nation's electrical supply by 2020 with 30% of that supply originating from nonemissions sources. This review concluded that, if these objectives were to become reality, there would be a need for a significant increase in nuclear generating capacity that could not be achieved solely by license extension and increasing the power output and capacity factor of the existing LWR plants. Thus, there would be a need for the construction of new reactors. Subsequently, DOE developed a framework that addressed a large number of potential reactor designs, together with an assessment of the research and development required before their deployment (Reference 4). This review is summarized in the appendix of this report. The summary gives a broader context to our comments on the RES Infrastructure Assessment document.

- A request for design certification for the Advanced Pressurized Water Reactor (AP1000) has been submitted and a review is currently under way. Pre-application reviews have been requested for the Gas Turbine Modular High-Temperature Reactor (GT-MHR), Advanced CANDU Reactor (ACR-700), and European Simplified Boiling Water Reactor (ESBWR) designs, and these reviews are also currently in progress. A similar request is expected imminently for the International Reactor Innovative and Secure (IRIS) design. Although the request for the Pebble Bed Modular Reactor (PBMR) design pre-application review was withdrawn by Exelon in 2002, the Office of Nuclear Reactor Regulation (NRR) anticipates that Exelon will resubmit this request and that this review will be restarted in late 2003. All of these reactor design concepts, which are deemed deployable by the DOE in the 2010-2020 time period, are within the scope of the RES Infrastructure Assessment document. Such a deployment schedule would place a burden on RES to conduct a significant amount of research in the 2003-2006 timeframe to support the projected NRR design certification activities.

- In October 2002 the NRC published the draft Infrastructure Assessment document. The document and its supporting appendices focus on identifying gaps in knowledge that need to be addressed within the period FY2002-FY2010. The document discusses the technical challenges of the advanced light water reactor (ALWR) and high temperature gas reactor (HTGR) designs because the NRC needs to be in the position of an informed reviewer of the detailed reactor designs once they are submitted for certification. This RES document incorporates the comments in our letter to the EDO, dated July 18, 2002 (Reference 5), on the draft Advanced Reactor Research Plan (Reference 6), as well as inputs received during various industry, ACRS, DOE, and NRC workshops and interactions with European and Asian organizations with operating experience on HTGRs.

1.3 Scope and Structure of the ACRS Report

The Infrastructure Assessment document was organized to cover the three strategic arenas of reactor safety, nuclear materials safety and waste safety, and safeguards.

The scope of our report is confined to the reactor safety arena and the development of a generic regulatory framework. The nuclear materials and waste safety issues are within the purview of the Advisory Committee on Nuclear Waste (ACNW). The topic of safeguards will be addressed in another ACRS activity. Although of vital importance to the success of RES work in support of the licensing of advanced reactors, the various management issues raised in the Infrastructure Assessment document are not addressed in this report; examples of such issues are the long-term availability and sustainability of funding and experienced manpower not only within the NRC but also in the industry as a whole, since the stated RES strategy depends critically on cooperative research with outside organizations.

The Infrastructure Assessment document was divided into sections on the Generic Regulatory Framework development, Reactor Safety subsections covering the analyses of accidents, reactor systems, fuels, materials, structures, and offsite consequence. The format of our report generally follows that of the RES document. Included in our report is an additional section on the impact of advances in computer technology on computational capability. An appendix is also attached addressing GEN IV reactor concepts.

2 RES ADVANCED REACTOR RESEARCH INFRASTRUCTURE ASSESSMENT DOCUMENT

This section discusses the generic regulatory framework development and reactor safety issues (accident and reactor systems analyses), and fuel, materials, structural and consequence analyses.

2.1 Generic Regulatory Framework Development

The Infrastructure Assessment document discusses general research needs for a "Generic Regulatory Framework Development." Development of such a framework is justified because the current regulatory structure is highly biased toward current LWRs and has limited applicability to non-LWR designs, was developed without benefit of current PRA technology, and is inconsistent in some areas. The framework would provide a process and criteria that can be used to develop technology-neutral regulations and reactor-specific guidance documents.

The framework is expected to possess the following attributes: (1) a hierarchical structure with the top-level goal being to protect public health and safety, (2) a strategy that establishes a balance between the various parameters in a defense-in-depth philosophy; and (3) quantitative safety guidelines based on the Commission's safety goals.

The framework is also expected to be able to meet the objectives of the Commission's Advanced Reactor Policy Statement: (1) "...as a minimum, advanced reactors will be required to provide the same level of protection to the public that is required for current operating LWRs," and (2) "it is expected that enhanced margins of safety and simplified, inherent, passive, or other innovative means to accomplish their safety functions will be utilized."

The starting point is expected to be the risk-informed Option-3 framework, which uses a hierarchical approach that utilizes strategies that establish a balance between prevention and mitigation. This balance involves limiting initiating event frequencies, core damage probabilities, radionuclide releases, and public health effects. These are intended to be the lines of defense in the defense-in-depth philosophy and, indeed, constitute an initial working definition of defense in depth. Quantitative guidelines for these limits would not appear in the regulations but, instead, would be in design-specific guidance documents. To develop the limits on each of the four lines of defense, the prompt fatality safety goal would be utilized as the top-level criterion.

Comments and Recommendations on Regulatory Framework Needs

1. The Option 3 and proposed generic regulatory frameworks are still conceptual. We believe it is important that the development of this framework be given sufficient priority so that it is in place before certification activities for non-LWR advanced reactors are begun.

2. A technology-neutral regulatory structure must address all of the agency's regulatory objectives. The prompt fatality safety goal is just too limited to serve this purpose. There is a need to define all of the agency's regulatory objectives at all levels of frequency and consequences in a risk context and to place quantitative acceptance guidelines on these parameters.

3. The Option 3 conceptual framework definition of defense in depth, along with the strategy guidance, provides a good starting point for developing a definition that has an objective quantitative basis. It still appears, however, that this will be inadequate for certification of advanced reactors, for the following reasons:

   - The Option 3 framework only addresses the prompt-fatality safety goal as the regulatory objective (see item 2 above).

   - The Option 3 framework is still LWR biased to some extent in that it deals with "core damage probability" and "containment failure probability." There is a need for a broader definition of these concepts in terms of radionuclide releases.

   - There is a need to tie the "balance" between the various prevention and mitigation actions to the actual uncertainties associated with the assessed risk contributions at the different lines of defense. At high confidence levels, some lines of defense may not be appropriate.

   - The frequency "bins" in the Option 3 framework appear to be much too broad. There is a need for acceptance criteria that are continuous at all levels of frequency (see item 2 above).

   - Certain elements of defense in depth should be independent of the results of PRA risk assessment. These elements need to be identified and appropriate acceptance criteria developed.

   - There is likely to be a desire to develop deterministic regulations using the design basis accident concepts. There is a need to determine if these regulations are necessary and, if so, how they can be developed to be consistent with top-level risk acceptance criteria.

2.2 Reactor Safety

This section discusses probabilistic risk assessment, instrumentation and control, human factors considerations, thermal-hydraulic analysis, neutronic analysis, and severe accident and source term analysis.

2.2.1 Probabilistic Risk Assessment (PRA)

RES anticipates that future applicants will rely heavily on probabilistic arguments to support their safety analyses for advanced reactors. These reactors may have passive safety systems, new fuel designs, and primary coolants other than light water.

Various features of advanced reactors challenge the capabilities of current state-of-the-art PRAs. For example, the reliability of passive safety systems whose performance will exhibit slow evolutionary behavior over the course of postulated accidents or intermediate failure states needs to be quantified. Advanced reactors will use digital instrumentation and control (I&C) systems whose failure modes and/or responses to low-voltage direct-current spikes, fires, or loss of cabinet cooling have not been modeled in current PRAs. In addition, the operator's role and staffing levels in some advanced reactors will likely be different than in current reactors. Modeling of human performance in PRAs for advanced reactors with postulated slowly evolving or long-term accidents or with multiple reactor modules using shared control rooms will be new areas that need to be addressed. Risk metrics and success criteria used in current state-of-the-art PRAs, i.e., core damage frequency (CDF) and large early release frequency (LERF), may require modification to better represent safety goals for advanced reactors.

RES is proposing work to develop the methods, expertise, and technical basis that will be needed to support staff review of a PRA submitted as part of a design certification application for advanced reactors such as an HTGR or IRIS.

Comments and Recommendations on PRA Needs

1. The proposed research is comprehensive and appropriate.

2. Limitation of the proposed research to design certification applications for an HTGR or an IRIS reactor design seems unnecessarily narrow, given the broad range of concepts currently under consideration for possible certification.

3. The research efforts proposed in the areas of system modeling, human reliability analysis, internal flood/fire/seismic events, multiple modules, and risk metrics are of particular importance in that these efforts address new issues raised by the advanced concepts.

4. Additional work is needed to develop formal methods to quantify epistemic uncertainties in the thermal-hydraulic code predictions of success criteria.

2.2.2 Instrumentation and Control

The Infrastructure Assessment document notes some characteristics of the current state of the art and the marketplace for nuclear I&C:

- No new nuclear plant orders have been placed for over 20 years, and development of domestic engineering experience in nuclear I&C has slowed. Domestic I&C vendors have focused their development effort on the chemical and petrochemical markets.

- The extension of operating licenses for the current fleet of commercial nuclear power plants is creating a demand for replacement of aging and obsolete I&C components and systems, and will continue to do so even if no advanced reactors are ever built.

- New orders for I&C systems and components, whether for replacements in existing facilities or for the construction of advanced reactors, are likely to come from "commercial-off-the-shelf" (COTS) sources. While the safety standards for COTS equipment and designs are good, sometimes COTS components do not specifically meet nuclear standards.

- Development in I&C concepts appears to have advanced more in Organization for Economic Cooperation and Development (OECD) countries than in the United States.

- Advanced reactors are likely to incorporate levels of automation far beyond the design concepts of the current generation of reactors. The capabilities inherent to digital I&C lend themselves to such development.

The current NRC research plan for digital I&C for FY2001-FY2004 is described in SECY-01-0155 (Reference 7). Most of the work in the current plan is applicable to both digital I&C system upgrades of operating power plants and advanced reactor designs. The major areas of research identified in SECY-01-0155 are:

- Systems aspects of digital I&C systems
- Software quality assurance
- Risk assessment and digital I&C systems
- Emerging I&C technology and applications

Comments and Recommendations on to I&C

1. The analyses of I&C issues presented in the Infrastructure Assessment document and SECY- 01-0155 are comprehensive, and the staff understands the marketing and technical environment in which regulatory tasks must be accomplished. The planned research leverages on a wide range of non-NRC research projects.

2. Previous ACRS conclusions and recommendations identified in past ACRS research reports (NUREG-1635, Vol. 1 [Reference 8], and NUREG-1635, Vol. 4 [Reference 9] are still valid and should continue to be considered by the staff.

3. Regardless of the timing of the emergence of advanced reactors, the research identified by RES will need to be performed to support replacement of I&C components and systems in the existing fleet of nuclear reactors and to maintain a knowledge base within the staff.

4. Work is needed to address the implementation of defense in depth and safety margin in the digital I&C systems.

5. In addition to safety system actuation, digital I&C systems will be used for performing diagnostic evaluations, online monitoring, and performance assessment of equipment. A better understanding of the integration of these functions will be needed.

6. Lessons learned from the design and operation of recently built nuclear reactors overseas, which utilize digital I&C systems, and digital upgrades in the U.S. LWRs should be considered for incorporation into the review of evolutionary and advanced reactor designs.

2.2.3 Human Factors Considerations

Many of the advanced reactor concepts envision the operators having quite different roles than they do today. Plant operations will be highly automated and emergency operating procedures will be computer based. Furthermore, shutdowns of plants for refueling will be much less frequent and consequently opportunities for maintenance and testing of equipment within the reactor containment will be less frequent. Many of the plants will have passive designs that are intended to eliminate the need for operator intervention during upset conditions.

Regardless of these changes, the issues of human factors in the safety of advanced nuclear power plants are largely the same as for current power plants, although the emphasis among the various issues may be different. For advanced plants with passive safety systems, human errors of commission may be of more concern than are human errors of omission. As another example, reduced maintenance should reduce the opportunities for latent errors, but reduced familiarity with maintenance procedures (due to the infrequent maintenance schedule) could lead to an increased risk of introducing latent errors.

By and large, the staff feels that its current approach for the development of human factors engineering will be adequate for advanced reactors. That is, investigations of the roles of humans in advanced plants have the same intellectual foundation current NRC human factors research. Consequently, specialized human factors research programs peculiar to advanced reactors are not readily identified. Continuation of the NRC staff's current efforts to stay abreast of developments in this technical community seems to be the appropriate approach.

However, plant staffing is one human factors issue that the NRC will need to address specifically for advanced nuclear power plants. Current requirements for staffing of control rooms of nuclear power plants have been developed based on a combination of tradition and experience. Advanced reactor vendors will propose reduced control room staffing because of the passive and/or automated features of the designs. This issue will arise in connection with many advanced designs and even with some evolutionary designs. The NRC must have a firmly established technical basis NRC for judging the plant staffing requirements both during normal operations and upset conditions.

Comments and Recommendations on Human Factors Considerations

1. Continuation of the NRC staff's current efforts to stay abreast of developments in the human factors arena is appropriate.

2. Because it is likely that proposed staffing levels for some of the new advanced reactors may be significantly lower than those for the current LWR fleet, RES should specifically address the issue of plant staffing. The NRC will need to develop methods that have a defensible basis for assessing the adequacy of staffing levels.

2.2.4 Thermal-Hydraulic Analysis

The Infrastructure Assessment document provides a comprehensive discussion of the research needed for HTGRs and also identifies needed research associated with four water-cooled reactors (AP1000, IRIS, ESBWR, and ACR-700). Research that is under way or planned for these reactor designs is described.

High-Temperature Gas-Cooled Reactors (HTGRs)

In general, the single-phase gas flow of the HTGR under normal operation is simpler to analyze than the two-phase flow of the LWRs. A smaller base of data and correlations will be required and it should be straightforward to incorporate this information into the TRAC-M code. In the near-term, the existing HTGR accident analysis codes GRSAC and THATCH can be used. Existing computational fluid dynamics (CFD) codes such as FLUENT can be used for multidimensional analysis where required. The analysis of accidents involving air and/or water ingress will require multicomponent analysis capability. This capability can eventually be incorporated into TRAC-M. The staff has identified a need to perform three-dimensional analysis of flow through porous and solid structures and to develop the appropriate constitutive relationships for fuel of a particular composition and shape. Specifically, the uncertainties associated with the fluid and material properties and the correlations for the transport processes will need to be quantified. This is a significant task and the staff-identified need to assess the completeness of the available database for gas flows and transfer processes is just a start.

The staff also plans to develop a code with two different working fluids to support the analysis of moisture-ingression accidents. Such a task will require major resources because there are insufficient analytical capabilities and databases to evaluate moisture penetration into fuel and graphite, the subsequent chemical reaction and heat transfer effects, and the consequent material degradation effects. Existing codes, however, appear likely to be adequate for predicting the cooling of intact fuel. The staff should evaluate whether these codes are sufficient to predict behavior should the fuel fragment or undergo significant changes in geometrical properties.

Advanced Light Water Reactors (ALWRs)

- AP1000

  A crucial safety strategy for the AP1000 is depressurization through the automatic depressurization system (ADS) valves and lines. Entrainment of liquid into the ADS-4 line under stratified flow in the hot leg has a significant influence on the AP1000 transient response during depressurization. Although this topic is being studied, we are concerned that important scaling issues have not been properly considered and that the basic mechanism of entrainment may be misunderstood. This is a particular concern if the major source of entrainment is an oscillating slug of water, the frequency of which is dependent on the hot leg geometry and scale.

Upper plenum entrainment in the AP1000 has so far been approached theoretically using correlations that are not specifically validated for AP1000 conditions. The Advanced Plant Experiment (APEX) facility at Oregon State University is being modified to better model the AP1000 and the data from testing will be used to assess the prediction capability of the TRAC-M code.

Low-pressure choked flow tests are being conducted at Purdue University using the PUMA facility in order to extend the empirical database to cover the regimes of operation of the ALWRs. The results of these experiments will be used to improve and validate the choked flow models of TRAC-M.

The current RES approach is to perform experiments and to develop corresponding models for use in TRAC-M at a later date. It would be more effective to develop models that are compatible with the TRAC-M framework as part of the work scope for these experimental studies.

- IRIS

  Most post-accident phenomena in IRIS should be amenable to prediction using existing codes such as TRAC-M, MELCOR, and CONTAIN. The staff has identified four phenomena that may require experimental verification. Plans for such tests should be prepared and justified after code runs and assessments of uncertainty show the need for specific tests.

- ESBWR

  The staff plans to develop the TRAC-M and CONTAIN codes to analyze the ESBWR. The staff has identified several key phenomena that need to be modeled, particularly those involving natural circulation, and anticipates that three-dimensional modeling capability will be required. This is a significant upgrade that will require suitable assessment, perhaps against new data that may be provided by the integral tests planned for the PUMA facility. The staff is planning to use a CDF code such as FLUENT to analyze boron mixing following an anticipated transient without scram (ATWS) event. The staff should assess the need to validate these predictions against a broader database than is currently available.

- ACR-700

  The staff plans to assess the still fledgling TRAC-M code to determine its applicability to ACR-700. The horizontal orientation and geometrical arrangement of tubes in this reactor are quite different than in current U.S. LWRs. Thus the large database that has been developed for assessing current LWR designs is mostly inapplicable. The staff needs to carefully assess the existing Canadian database and decide how well it will support the certification process for the ACR-700 design. The staff has identified several key phenomena that require modeling, but has yet to determine the extent of the need for new code development or experiments.

TRAC-M Development and Uncertainty Issues

Several of the current RES research efforts are likely to have applicability to advanced reactors as well as to current LWRs. For example, the research program at Penn State University on reflood heat transfer, in response to needs identified by RES, has already produced important data. These research results need to be analyzed and used to develop analytical methods to be incorporated into the TRAC-M code (not a trivial matter). An assessment should then be performed of the adequacy of this database for evaluating similar phenomena in ALWRs.

Thermal-hydraulic codes play a central role in the analysis of transients and accidents in all present reactors and will continue to do so for future reactors. The RES plan for addressing future thermal-hydraulic issues of a system nature is based primarily on the development of the thermal-hydraulic system code TRAC-M. This code will combine the features needed for all current generation LWRs as well as any features specifically identified as needed for ALWRs and HTGRs. The development and implementation of TRAC-M needs to be accelerated to the point where it is the current tool of use. Having the code available is important, but this is just the first requirement; the agency needs extensive experience with its qualification. Much needs to be learned about how much effort it will take to adapt the code to future designs. While there is general agreement that this can be done, it will be uncertain how much effort is needed until the specific qualification work is undertaken.

The current trend toward the reduction of excessive conservatism and the use of "best estimate" codes requires explicit consideration of the uncertainties associated with the predictions of the code. The need to assess uncertainties associated with thermal-hydraulic calculations has motivated the development of various methods (such as nonparametric probabilistic evaluation of epistemic and aleatory uncertainty, and response surface methods) to evaluate the relative importance of the various sources of uncertainty. Use of these evaluation methods will require that the parameters in the code and the accuracy of the thermal-hydraulic models in the code be characterized by uncertainty ranges and associated frequency distributions. These methods should be included as integral parts of the codes so that Monte Carlo methods for assessing the effect of the uncertainties can be applied routinely.

Although the nonparametric probabilistic and response surface methods can address uncertainty of the modeled phenomena, there is also a need to address the overarching uncertainty associated with phenomena that are not modeled or are incorrectly modeled. This may require evaluation and identification of levels of conservatism needed in order to achieve the desired margin of safety. The code scaling applicability, and uncertainty (CSAU) evaluation methodology can identify the importance of these areas of uncertainty, but stops short of identifying methods for evaluating the uncertainties.

The assessment of the performance of passive safety systems may be particularly sensitive to uncertainties in the codes because the driving forces in natural circulation are much smaller than in a forced circulation system. Therefore, more accurate predictions of the balance between the competing effects of natural and forced circulation are needed. For instance, the outcomes of PRA analyses of emergency cooling systems for current reactors are largely determined by the availability and reliability of active components such as pumps. In the new designs utilizing passive safety features, the successful prediction of emergency cooling systems will be assessed by codes the results of which are sensitive to uncertainties in predicting flows and heat transfer. The first priority is, therefore, to quantify the uncertainties in these codes and evaluate the probability of success. Formal methods to incorporate this model uncertainty into the PRA need to be developed.

Comments and Recommendations on Thermal-Hydraulic Analysis

1. The TRAC-M code is viewed as the major tool to be used by the agency for predicting the thermal-hydraulic behavior of advanced reactors. Data are being developed that need to be analyzed and used to develop analytical methods that will be incorporated into the TRAC-M code. There is a need to speed up the code's development and range of applicability. The staff also needs extensive experience with the use of the TRAC-M code before the code can be relied upon for making regulatory decisions.

2. Development of improved models of thermal-hydraulic phenomena for incorporation into the TRAC-M code should be an integral part of the RES programs related to the investigation of the phenomena.

3. Formal methods need to be developed to quantify epistemic uncertainties in the code predictions of success criteria for use in PRAs.

2.2.5 Neutronic Analysis

The neutronic analysis of the reactor core is fundamental to the safety of nuclear power plants. Power reactor cores must be stable, reliable, and predictable. The safety analysis of the core requires detailed databases and computer codes for analyzing both local details of neutronic behavior and core-wide behavior.

The NRC staff has done a good job of maintaining its neutronic-analysis capabilities at or near the state of the art. It is essential that the staff continue to maintain these capabilities at a high technical level. There must be no doubts in the public's collective mind about the nuclear safety of power reactor cores.

Existing reactors have been designed conservatively based on neutronic analyses that are quite simplified relative to the capabilities that now exist. They have also been licensed using simpler analytic methods that entail substantial conservative margins.

Even within the operating reactor community there is a growing desire and need to take advantage of the technical developments in neutronic analyses that have taken place over the last two decades to remove unnecessarily conservative margins in core design and operation. As an example, this need has become manifest as plants use reactor fuel to higher levels of burnup. Experiments have shown that high-burnup fuel cannot sustain as much energy from reactivity insertion events once thought. But, improved neutronic analyses have shown that energy inputs from the more probable reactivity insertion events are much smaller than had been thought based on older analyses. There is, then, pressure from the industry on the staff to accept operation of cores with higher burnup fuel based on the predictions of the improved analysis tools.

This pressure on the staff to adopt more advanced neutronic analysis capabilities in the regulatory process will continue to grow as advanced reactors with cores designed using the more sophisticated analysis tools are proposed for certification. The licensing process needs to evolve in a way that takes advantage of the technical advances and abandons unnecessarily conservative approaches of the past to neutronic safety.

Though the NRC staff has done a remarkable job of maintaining its databases and computational resources, advanced reactors under consideration today do pose some challenges. For purposes of discussion, three groups of advanced reactors are considered. The first group consists of advanced reactors such as the AP1000 and the ESBWR that involve modest evolutions from the current reactor technologies and that depend on passive safety systems. Safety analyses of these reactors demand close coupling between models of neutronics and thermal-hydraulics within the reactor coolant system. Steps being taken by the staff to couple the three-dimensional neutronics code PARCS to the TRAC-M code appear adequate to meet the regulatory safety analysis needs for these reactors, though the challenges of implementation of the analyses should not be underestimated. Thus, we recommend that the development of the combined TRAC-M/PARCS computer code should continue to be supported.

The ACR-700 is a second type of advanced reactor for consideration. This reactor is similar to the CANDU reactors which have positive void coefficients that affect the neutronics in ways that are usually considered undesirable. Modest enrichments of the fuel currently proposed for the ACR-700 and the use of light water as the coolant are intended to ameliorate the tendency for positive void coefficients. Still, the interplay between neutronics and the unfamiliar thermal- hydraulics of a core configured horizontally will challenge the analytic tools available to the staff for performing safety assessments. Modifications of these tools to account for the core composition, core configuration, and reactor coolant system thermal-hydraulics of the ACR-700 are long-lead-time activities that need to be initiated now if the regulatory process of certification is to proceed in a timely manner.

The gas-cooled high-temperature reactors make up the third group of advanced reactors for discussion. Gas-cooled reactors involve substantial departures from today's LWR technology. These reactors received much of the attention in the NRC staff's Infrastructure Assessment because the reactors present challenges that are at or even beyond the current state of the art in neutronic analysis. The moving annular core of the PBMR with its stochastic fuel densities and unpredicted temperature distributions is just one example of the analysis challenges that may be presented to the NRC staff by gas-cooled advanced reactors.

The staff has presented an appropriate research program to confront these challenges. The staff does have some philosophical issues to confront as advanced reactors based on novel technologies are proposed. Historically, the NRC staff has done almost completely independent analyses of reactor neutronics, even to the point of using independently developed analytic tools. The question arises, therefore, should this practice continue, or should the NRC staff confine itself to a review of analyses done by the applicant? We believe that independent analyses should continue to be the practice. Independent analyses may become more important as technically advanced tools are increasingly used to study novel concepts. Neutronic analysis is simply too central to nuclear safety to take a lesser approach.

An overarching issue is the availability of data to validate the predictions of neutronic analyses. In the past, there have been abundant data for such validations and it is likely that data can still be obtained for code validation using lead test assemblies for reactors and fuel that are modest departures from existing technologies. For the more novel designs, neither the NRC staff nor the reactor designers have access to experimental facilities to provide validation data that are prototypic. Indeed, the staff has already identified challenges in obtaining suitably prototypic data for the ACR-700. The staff must insist that advanced reactor designs be supported by prototypic experimental data. Though modem tools of neutronic analysis are very sophisticated, they are neither transparent nor immune to error. Therefore, we recommend that the staff provide explicit direction to the applicant on the experimental data and data quality that it thinks are essential to support and validate neutronic analyses of reactor designs submitted for certification.

Comments and Recommendations on Neutronic Analysis

1. The maintenance and improvement of neutronic analysis capabilities at the NRC should continue to be supported so that the NRC can continue to perform independent assessments of reactor neutronics.

2. The effort to couple the thermal-hydraulics code TRAC-M with the three-dimensional neutronics code PARCS will be essential for certification of reactors such as AP1000 that depend on passive safety systems.

3. Modifications to analysis methods to account for the different features of the ACR-700 should be initiated now to facilitate anticipated certification review.

4. NRC staff should provide explicit direction to the applicant as to the data and data quality that will be needed to support and validate advanced reactor designs submitted for certification.

2.2.6 Severe Accident and Source Term Analysis

Risk associated with any nuclear power plant is dominated by the frequencies and consequences of severe accidents that involve substantial releases of radioactivity from the fuel. The issues of accident progression, fuel degradation, fission product release and transport within the reactor coolant system and fission product behavior within the reactor containment or confinement are discussed in this section.

As with neutronic analysis, it is convenient to consider three groups of advanced reactors for the discussions of accident progression and accident source terms. The first of these groups consists of reactors whose designs evolved from current reactors. This group includes the AP1000 and the ESBWR. Accident analysis and fission product behavior codes developed by the NRC for the current generation of power reactors appear applicable with modest modification to this group of reactors.

For instance, two issues have emerged for these reactors that necessitate additional experimental data and specialized analyses. One of these is the degradation and fission product release from fuel taken to high levels of burnup (>60 GWd/t). The NRC is considering an extension or follow-on to the current PHEBUS-FP program to obtain data on degradation and fission product release from high-burnup fuel. We certainly encourage these efforts and recommend that a program to provide the needed data be defined.

A second issue has to do with the arguments by some advanced reactor designers that core debris produced by a severe accident can be retained within the reactor pressure vessel. The necessary heat removal is to be achieved by flooding the reactor cavity. This safety strategy was proposed for the AP600 design. The strategy was not credited in the staff's safety case because of concerns over the heat transfer model and chemical interactions within the core debris.

The NRC has been an instigator of the international cooperative research programs RASPLAV and MASCA. These programs have provided important information on the convective heat transfer from core debris in the lower plenum of a reactor vessel. These programs have also shown that chemical interactions within core debris and between core debris and reactor vessel materials are more complicated and more aggressive than envisaged by the advanced reactor designers. The NRC needs to develop a quantitative understanding of these chemical processes for review of proposed reactor designs. The MASCA experimental studies should be continued and appropriate models incorporated into the NRC's reactor accident models.

The ACR-700 is a second category of advanced reactor for discussion of severe accident concerns. It appears that substantial effort will be required to modify the NRC's accident analysis models to treat this reactor, which has a unique horizontal configuration of the core and a complicated reactor coolant system. The NRC does not have a body of data on the degradation of such a reactor core design under severe accident conditions comparable to the data available for fuel configurations found in current U.S. power reactors. It is not evident what accident progression data will be available from the reactor designers. Early examinations suggest that accident progression could be rapid when significant pressure tube deformation occurs. Furthermore, issues of energetic interactions, of molten core debris with coolant or moderator may be more problematic for the ACR-700 than these processes are thought to be for current U.S. reactors. The NRC needs to investigate modification of its accident analysis codes to address the ACR-700 and identify experimental data needed to validate predictions of accident progression, fuel-coolant interactions, and fission product release. It should be the responsibility of the applicant to provide the data identified in this effort.

The third class of advanced reactors consists of HTGRs with graphite moderator. It is evident that some severe accidents in such reactors will involve the ingression of air. The NRC staff's experience with accident progression in such reactors is limited. Similarly, the NRC staff has little experience with either the release of fission products from fuels or the transport of these fission products within the coolant system and containment for such reactors.

Existing NRC accident analysis tools and fission product release and transport models would require very substantial modification to be applicable to these gas-cooled reactors. It will take substantial time to develop adequate tools for the staff's analysis of accident progression and source term predictions. Consequently, there is a need to initiate these efforts. It is not readily apparent, however, what data are available from well-scaled tests for validation of the modified models. The staff will need to address this issue as part of the effort to update the tools.

Comments and Recommendations on Severe Accidents and Source Terms

1. Accident progression and source term analyses for new reactors such as the ESBWR and AP1000 can be addressed through modest modifications of existing codes. Issues associated with the use of high-burnup fuels will have to be addressed.

2. Some of the new reactor designs incorporate cavity flooding, which the designers argue will retain the debris produced by a severe accident in the pressure vessel. Research will be needed to assess these arguments. The NRC should continue its involvement in the MASCA work.

3. A substantial effort will be required to configure the NRC staff's accident analysis models to treat the unique core and coolant configuration of the ACR-700. The NRC staff needs to investigate what modifications are needed for its codes and to identify the experimental data that should be supplied by the applicant to validate predictions of accident progression and fission product release.

4. The NRC has limited experience with severe accident analysis and fission product release for gas-cooled reactors with graphite moderators and structures. Existing NRC codes will require substantial modification to be applicable to the analysis of these types of reactors. It will take substantial time to develop adequate tools and to identify the available and needed new experimental data before code model development can begin.

2.3 Fuel Analysis

Advanced reactor designs under active consideration today use fuels that can be grouped, as in discussions of neutronics and severe accident analyses, into three classes. The first of these classes comprises somewhat prosaic fuels little different than those used in the current LWRs, though possibly with improved cladding alloys. The primary issue associated with this class of fuel will be increased levels of burnup. The NRC currently has a confirmatory research program to ensure the safety of fuel to burnups of 62 GWd/t. It is imperative that this research continue to confirm the adequacy of regulatory decisions made to date concerning high-burnup fuel. As stated in the previous section, it is also important that a companion research program on degradation of high-burnup fuels under severe accident conditions be continued.

Licensees and designers are interested in using fuels to burnups in excess of 62 GWd/t. The NRC has decided that licensees will have the responsibility to provide technical justification for using fuels to the higher levels of burnup. Regardless, RES will still have to provide the agency with tools to review and independently evaluate these technical justifications since currently available fuel behavior models cannot be reliably extrapolated to such high burnup levels. A crucial issue for the staff to consider is the need for experimental data on fuel behavior under upset conditions when fuel has been used to burnups on the order of 75 GWd/t. In light of the change in fuel physics that has been observed as burnups exceed ~50 GWd/t, it seems likely that experimental data would be needed to justify use of fuel to burnups in excess of 62 GWd/t.

A second class of fuel for advanced reactors is the fuel to be used in the ACR-700 design. This fuel is a slightly enriched UO_2 fuel which is horizontally configured in the reactor. The NRC staff has minimal experience with this fuel and fuel configuration. It does appear, however, that the fuel is not a radical departure from designs that the NRC staff has considered in the past. It may well be possible for NRC to extrapolate its existing fuel behavior codes and databases to evaluate the ACR-700 fuels. The confidence with which this can be done will depend on experimental data available from the developers of the ACR-700 concerning fuel behavior, especially under upset conditions and conditions of reactivity insertion events.

The third class of advanced reactor fuel is the coated-particle fuel to be used in gas-cooled reactors, such as the PBMR and the GT-MHR. Because little experience is available within the NRC for the review of coated-particle fuel, it may be necessary now to initiate long-term efforts to develop these capabilities. Development of these capabilities will have to begin with analysis methods using data for fuel prepared and tested in the past, largely overseas. We caution that analysis methods now available in the literature are for highly idealized circumstances and may not be adequate for the safety analysis of coated-particle fuels during normal operations and upset conditions.

Comments and Recommendations on Fuel Analysis

1. As with currently licensed reactors the primary issue to be addressed with ALWR fuels is increased burnup. This issue is being addressed for current reactors and needs to be given a high priority.

2. It may be possible for the NRC to extrapolate the use of existing codes to evaluate ACR-700 fuel. The confidence with which this can be done will depend on the experimental data provided by the developers of the ACR-700.

3. Coated-particle fuels were the focus of much of the staff's attention in its Advanced Reactor Research Infrastructure Assessment since there is little NRC experience with these fuels. Because of the long-term nature of the associated research, efforts to develop the NRC staff's capability to address the review of these types of fuels should be initiated.

2.4 Materials Analysis

The Infrastructure Assessment document identifies critical gaps in the materials behavior knowledge needed for transitioning from the relatively well-understood LWR systems to HTGR systems are identified in. The document focuses on ensuring an adequate understanding of the behavior of metallic and graphite HTGR component which have critical structural, barrier, or retention functions under normal and off-normal conditions. Potential degradation mechanisms in high-temperature helium for these components are thoroughly discussed and include, for instance, graphite swelling, the effect of coolant impurities, and creep-fatigue interactions in metallic components. There is an existing experience base in many of these areas, and the designers are expected to develop additional data. However, because the degradation phenomena impact the codes and standards for the design and fabrication of the reactor as well as the safety margins under normal and accident conditions, we agree with the staff's argument that the NRC must have an independent research capability in the most critical of these areas. Previous experience has shown that characterization of these material degradation phenomena requires a long-term effort. Thus, it is necessary to initiate such efforts relatively soon. We agree that prioritization should be given initially to (a) evaluating the completeness of the existing databases needed to formulate design codes, (b) initiating the construction of the experimental facilities needed to investigate the potential degradation of proposed materials in HTGR environments, and (c) finalizing effective collaborative agreements with overseas organizations. These tasks will lead to a definition of additional data needed for regulatory decisions, and will ensure that the experimental infrastructure is in place to provide timely information.

The ACR-700 design is sufficiently different from the current LWRs that research is needed in several areas. The staff identified the following concerns: (a) the effect of the ACR-700 environment on component fatigue and creep life, (b) irradiation and embrittlement of the pressure tube material, (c) the performance of the large number of dissimilar metal welds in the header system, and (d) component material behavior under severe accident conditions. We agree with these concerns and believe that their resolution should be given priority since the ACR-700 is currently undergoing pre-application review by the staff and relevant data will be required before certification.

For ALWRs like the AP1000 and ESBWR, the materials issues are similar to those associated with the current LWRs and hence are being addressed by the RES programs dealing with the current operating plants. We note that RES has ongoing programs on technical basis for reevaluation of the pressurized thermal shock screening criteria, steam generator tube degradation, and vessel head penetration cracking and degradation. This work should continue and is of direct relevance to most of the ALWR design concepts. The development of risk-informed regulations for both the current LWR fleet and the future reactors will require knowledge of the probabilities of material "failure" and an assessment of associated uncertainties. There is a considerable amount of international interest in the following reports of this topic:

- The relevance and quality of environmentally assisted cracking and corrosion databases upon which structural integrity decisions are made

- Databases to support the factors of improvement for "cracking-resistant" alloys such as 690, 52, and 316L

- The development of prediction models which lead to proactive life management capabilities

Although the development of databases and proaction is primarily the applicant's responsibility, we recommend that RES continue to participate in these activities so as to understand the assumptions inherent to the applicant's assertions of materials integrity. Materials degradation issues have been at the core of many of the current LWR fleet problems, and in order to ensure adequate regulation and control in the future it is important that active participation in the research activities for current and advanced water-cooled reactors be continued.

Finally, the Infrastructure Assessment document correctly identifies a need for an improved inservice inspection program for both ALWRs and HTGRs. This need is prompted by the integral nature of many of the advanced reactor designs (leading to diminished accessibility for inspection) and the anticipated longer operating cycles between scheduled, short-duration refueling outages when inspections would normally occur to quantify damage and the impact on safety. This task is primarily the responsibility of the applicant, but RES should conduct sufficient research to remain aware of the technical issues.

Comments and Recommendations on Materials Analysis

1. RES should undertake research activities to identify potential materials degradation processes in HTGRs. Tasks in the near-term (i.e., FY2002-FY2004) include (a) evaluating the completeness of the existing databases needed to formulate design codes, (b) initiating the construction of the experimental facilities needed to investigate the potential degradation of proposed materials in HTGR environments, (c) initiating research on environmental effects on cracking (fatigue, stress corrosion) and creep, and, (d) finalizing effective collaborative agreements with overseas organizations. This prioritization is appropriate given that these lead tasks are long-term items that prepare the foundation for future work in support of projected design certification.

2. The proposed work outlined for the ACR-700, which involves somewhat different materials and environments than current PWRs, is appropriate and will provide a technical basis for NRC assessment of industry proposals and supporting data.

3. We concur with the assessment that the ongoing studies to address materials issues for operating reactors should continue. RES should remain involved in international collaborations to develop and analyze materials degradation databases in order to assess the probability of failure of "degradation-resistant" materials.

2.5 Structural Analysis

The Infrastructure Assessment document section on structural analysis primarily addresses the analysis, aging, and inspection of containment/confinement structures and the response of those structures and the primary system to external events. The document also recognizes that while acceptance requirements for current LWRs are primarily deterministic, there is a strong interest in performance-based and risk-informed acceptance criteria.

The areas in which the Infrastructure Assessment document indicates that research will be needed include (1) seismic hazard assessment, (2) nonlinear seismic analysis of reactor vessel and core support structures, (3) the analysis of the soil-structure interaction of deeply embedded or buried structures during seismic events, (4) the effects of high temperature on properties of concrete, (5) the use of modular construction techniques for safety-related structures, and (6) risk-informed inservice inspection methodologies for containment and associated structures.

Current NRC guidance for determination of seismic hazards states that both the Lawrence Livermore National Laboratory and the EPRI probabilistic seismic hazard methodologies are acceptable for determination of the safe shutdown earthquake for nuclear power plants. For some sites, the estimates from the two methodologies can be significantly different, which can lead to difficulties in cases where quantitative acceptance criteria must be met. RES proposes to use a set of guidelines developed jointly by the NRC, DOE, and the EPRI Senior Seismic Hazard Analysis Committee (SSHAC) to update the two currently accepted seismic hazard assessment methodologies for the Central and Eastern U.S. to obtain more consistent seismic hazard assessment.

Research is planned to develop an independent NRC staff capability to evaluate the seismic integrity of advanced reactors. The applicability of existing finite element analysis codes to analyze nonlinear configurations such as the nonductile graphite core reflectors and supports in HTGRs will be assessed. In designs with long fuel tubes like AP1000 and IRIS, the seismic margin may be controlled by the fuel design, and nonlinear static and dynamic structural analyses may be needed to assess seismic margins.

For some new reactor designs, the entire reactor building and a significant portion of the steam generator building will be partially or completely embedded below grade. Soil-structure interactions will have a significant effect on the seismic response of such structures, and models of soil-structure interaction in existing computer codes (which have generally been developed for structures with relatively shallow embedments) will be reviewed to determine their applicability. Shake table tests will also be performed to provide experimental verification of analytical results.

The Infrastructure Assessment document notes that concrete structures in HGTR designs will be subjected to sustained high temperatures. Substantial information is available from the literature and previous work at the Sandia and Oak Ridge National Laboratories. The available information will be collected and reviewed to identify gaps in the data.

Although used in Japan, modular construction has not been used in the U.S. for nuclear power plants. The modules of interest are concrete-filled steel plate wall and foundation modules. The designers of the GT-MHR, AP1000, and the IRIS reactors have proposed using modular construction techniques in order to shorten the construction schedule, reduce costs, and improve the quality of construction. Technical issues that must be addressed to accept use of modular construction include assessment of the strength and ductility of the module and of the joints and connections needed to join the modules, and determination of appropriate seismic damping values for the resulting structures. The proposed research will be used to develop evaluation criteria that will facilitate review of reactors that use modular construction.

The Infrastructure Assessment document notes that inservice inspection requirements for new reactors will be largely based on risk-informed approaches. Indeed, the ASME has formed a task group to develop methodologies for risk-informed inservice inspection of containments. The staff plans to actively participate in this ASME Code activity while independently developing methodologies for risk-informed inservice inspection of containments. The work the staff intends to do to support this effort includes compiling data on degradation mechanisms for structures, developing appropriate inspections strategies for these degradation mechanisms, and defining risk categories based on the potential consequences of failure. We address this proposal in the following comments and recommendations.

Comments and Recommendations on Structural Analysis

1. The Infrastructure Assessment document appears to address the potential areas of concern for the structural analysis of the advanced reactor design bases and seismic events.

2. We support the effort to implement the SSHAC guidance and associated methodology to obtain new probabilistic seismic hazard estimates to provide a sounder technical basis for assessing seismic risk. This is important not only for advanced reactors but also for current reactors.

3. The seismic design and analysis of advanced reactors will involve extensions of current technology. Although the primary responsibility for the development of this technology lies with the nuclear industry, a modest program in this area seems warranted to help ensure that the staff maintains an awareness of the state of the art and the limitations of the available technologies.

4. The potential degradation of containment or confinement structures is an important area of concern for the NRC. Enough is known about the design of the proposed HTGRs to realize that the temperatures will be high enough to require an extension of current design codes and a reassessment of the aging of concrete structures. The RES Infrastructure Assessment document properly addresses this as a potential area of concern.

5. Risk-informed inservice inspection (RI-ISI) clearly has provided a more rational approach to the inservice inspection of piping, and it is reasonable to expect that it can be beneficial in developing inspection programs for containment. RES played an active role in the initial development of RI-ISI for piping systems. This was appropriate since it was a new approach and neither the NRC nor the industry had prior experience. Although the extension to containment systems will involve new concepts and approaches, we believe that it is not necessary for the staff to

independently develop this methodology. It seems sufficient that the staff participate actively in the ASME Code activity. The staff's other research efforts in this area should be sufficient to maintain an awareness of the necessary expertise and technology.

2.6 Consequence Analysis

Licensing of advanced reactors will require an evaluation of the offsite consequences of severe accidents for such reactors. Specifically, a technical justification for a recommendation to the Commission on the size of the emergency planning zone (EPZ) will be required. The supporting calculations will need to be consistent with those currently utilized for choosing a 10-mile EPZ for the present LWRs, which follow the criteria established in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," and NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants."

The RES Infrastructure Assessment document foresees that the staff will perform offsite consequence analysis of advanced reactors using NRC's Level 3 evaluation code MACCS2. The mix of radionuclides and the chemical forms in the releases from severe accidents in advanced reactors may be different than in the releases during accidents in LWRs. A Level 2 analysis will provide estimates of the radionuclides produced. RES plans to evaluate MACCS2 to ensure that design-specific biologically important radionuclides, dose conversion factors, and factors that account for their chemical form are appropriately modeled in MACCS2.

Comment on Consequence Analysis

The RES plan will enable the staff to perform consequence analysis of advanced reactors and support developing recommendations to the Commission on the size of the EPZ.

3 PHENOMENA IDENTIFICATION AND RANKING TABLE (PIRT) PROCESS AND IMPLEMENTATION PLAN

A PIRT process, similar to that used by RES in the review of advanced reactor designs in the early 1990s (MHTR, PIUS, PRISM, CANDU-3), is outlined in the RES document. This will be used to identify and prioritize the additional research needed for the development of a risk-informed regulatory framework for advanced reactors. The implementation of the advanced reactor research activities will reflect the outcome of these task prioritization activities and budget and timing constraints. For instance, long-lead time tasks such as rulemaking, codes, and standard development activities should be completed in advance of the start of formal design certification.

Numerous management questions are raised in the RES document, including, the need for collaborative projects with international and domestic partners, the risks associated with noncompletion of task objectives, longevity of funding resources and related issues. All of these implementation questions are valid and pertinent and need to be addressed.

Comments and Recommendations on PIRT activities and Plan Implementation

1. The PIRT process that RES has outlined is appropriate for defining the prioritization of tasks necessary to formulate a risk-informed regulatory framework for advanced reactors.

2. RES has identified valid research plan implementation issues, which need to be addressed.

4 IMPACT OF ADVANCED COMPUTER CAPABILITIES ON NRC STAFF ACTIVITIES

Computing power is expanding at extraordinary rates. Modern personal computers (PCs) perform at speeds > 4 billion calculations per second (4-gigaflops) that exceed the capability of supercomputers just a few years ago. However, these speeds pale compared to those available in modern multiprocessor systems. Currently, there are 47 systems in the world with speeds that exceed 1000 gigaflops (1-teraflop). Six months ago, there were only 23 such systems. IBM is now developing a 360-teraflop computer for DOE. Although such powerful systems are currently available primarily at Government laboratories and university supercomputer centers, Linux PC clusters with speeds of 200-400 gigaflops now cost less than $100K and are becoming widely available. Indeed, gigaflop machines are becoming standard tools for engineering analysis by commercial firms.

The availability of such computing power makes possible calculations of far greater complexity and opens new possibilities for visualizing information. We expect that new reactor designs will take advantage of virtual reality tools to optimize the layout of the plants. Clearly, enormous increases will be possible in the scale of problems that can be analyzed by computational fluid dynamics models; truncation, approximation, and recalculation of PRAs need no longer be a significant consideration; core designers will find ever more ingenious ways to eke out more power from existing reactors; and enormous increases in the sophistication of sensitivity and uncertainty analyses will be possible.

We see little evidence that the NRC staff is preparing for the potential impact of these changes. The staff should assess the possibilities that quantum increases in computing power can have on the way it does analyses and develop a plan to use the new tools to increase the efficiency and effectiveness of the agency. The potential challenge the agency faces is that new reactors will be designed using software that runs only on these advanced computing resources. Without similar calculational tools (both software and hardware), the NRC will be hard pressed to carry out its safety assurance mandate without imposing unnecessary and even anachronistic burdens on applicants and licensees.

Recommendation on Computer Capabilities

The NRC staff needs to consider the impact the enormous increase in computer capabilities that are occurring will have on the way engineering analyses (thermal-hydraulic, fire, core neutronics, and PRA analyses, etc.) are done and assess how this capability can be used to increase the efficiency and effectiveness of the Agency's work.

5 OVERALL CONCLUSIONS AND RECOMMENDATIONS

The Advanced Reactor Research Infrastructure Assessment document is comprehensive and reflects a good understanding of the issues, existing state of the art, and past and ongoing research results and activities as they pertain to future reactor designs. It identifies the increments in technology necessary for the development of a risk-informed and performance-based regulatory structure for some specific ALWR and HTGR design concepts.

We generally agree with the RES assessment of development needs and planned activities in several of the technology areas (i.e. probabilistic risk assessment, instrumentation and control, materials analysis, structural analysis, and consequence analysis) and offer supportive comments on details. In other technology areas, we have more specific comments on the focus that RES proposes. These later comments are expanded upon in Section 2 of this report, and include the following:

1. The research needs for a technology-neutral "Generic Regulatory Framework" are discussed in the general terms of a hierarchical structure based on the preliminary framework under development for "Option 3." This framework should prove useful for the certification of the more likely plants to be first deployed (i.e., ALWRs which will be probably licensed under the current 10 CFR Part 52 rule), and will be mandatory for the non-LWR concepts on a longer timeframe. This development will take some time. Our concerns with the current risk-informed Option 3 concept should be addressed.

2. Proponents of advanced reactor designs are proposing fewer numbers of control room staff than are used in today's LWR fleet. In the area of "Human Factors," we believe that the nuclear industry will be challenging existing requirements for control room staffing. It is important, therefore, that the NRC staff undertake activities to ensure that its positions on control room staffing have firm and defensible basis.

3. In the area of "Thermal-Hydraulics", much reliance is placed on the TRAC-M code to meet the licensing challenges posed by features of the various advanced water reactor designs. It is recommended, therefore, that priority be given to quickly code's developing this code and extending its range of applicability and, just as importantly, putting it to use. Because PRA success criteria for passive designs are sensitive to uncertainties in thermal-hydraulic code predictions, these model uncertainties need to be quantified.

4. The maintenance and improvement of neutronic analysis capabilities at NRC should continue to be supported so that the NRC can continue to do fully independent nuclear safety assessments. Completion of the effort to couple the thermal-hydraulics code TRAC-M with the three-dimensional neutronics code PARCS will be essential for certification of reactors such as AP1000 that represent only modest departures from current reactor technologies. Modifications to analysis methods to

account for the different features of the ACR-700 should be initiated now to facilitate timely completion of the anticipated certification review.

Work to develop neutronic analysis of gas-cooled reactors will be challenging given the quite limited experience at the NRC in this area and the significant challenges associated with, for example, stochastic fuel pellet movements and unpredictable temperature distributions. The NRC staff should provide explicit direction to the applicant as to the data and data quality that will be needed to support and validate advanced reactor designs submitted for certification.

5. The needs in the area of "Severe Accident and Source Term Analysis" for new reactors such as the ESBWR and AP1000 can be addressed through modifications of existing codes. Issues related to higher burnup fuels will have to be addressed.

   Some of the new reactor designs have features, such as cavity flooding, that the designers argue will result in severe accident debris being retained in the pressure vessel. Research will be needed for the NRC staff to evaluate these arguments and become knowledgeable on this issue. Consequently, the NRC staff should continue its involvement in the MASCA work.

   It appears that a substantial effort will be required to configure NRC's accident analysis models to treat the unique core and coolant configuration of the ACR-700. The NRC staff needs to investigate what modifications to its codes are needed and to identify experimental data that may be needed to validate predictions of accident progression and fission product release.

   The NRC staff's experience with severe accident analysis for gas-cooled reactors with graphite structure and moderator is quite limited. Existing NRC codes will require substantial modification to be applicable to the analysis of this type of reactor. It will take substantial time to develop adequate tools and there is a need to identify the available and needed new experimental data and to begin code model development.

6. In the area of "Fuels Analysis," the advanced reactors under active consideration utilize fuel designs that can be grouped into three classes: (1) fuel little different than used in currently operating reactors, (2) the fuel that will be used in the ACR-700, and (3) coated-particle fuels to be used in gas-cooled reactors such as the PBMR and GT-MHR.

   The primary issue to be addressed with the first class of fuel is the potential impacts of increased burnup. This issue is being addressed for current reactors and needs to be given a high priority. It may be possible for the NRC to use existing codes and databases to evaluate ACR-700 fuel using data provided by the ACR-700 developers, but this needs to be confirmed.

Coated-particle fuels were the focus of much of the staff's attention in the Advanced Reactor Research Infrastructure Assessment since there is little NRC experience with these fuels. Given that design certification activities for the GT-MHR are projected to start in early 2006, it is necessary that efforts be initiated to develop NRC's capability to address the review of these types of fuels.

7. The NRC staff needs to consider the impact that the enormous increase in computer capabilities that are occurring may have on the way engineering analyses in the areas of thermal-hydraulics, fire, core-neutronics modeling, PRA etc. are done, and to assess how this capability can be used to increase the efficiency and effectiveness of the Agency.

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APPENDIX

DOE NEAR-TERM DEPLOYMENT AND GEN IV REACTORS

The RES Infrastructure Assessment document addresses some of the DOE advanced reactor designs (AP1000, ESBWR, PBMR, ACR-700, GT-MHR, IRIS) and these are discussed in previous sections. However, the RES document does not discuss the wider range of design concepts that DOE considers for near-term deployment (i.e. by 2015) or the "Generation IV" designs which, in DOE's best case assessment, may be deployable by 2020-2025.

In 2000 DOE's Office of Nuclear Energy, Science, and Technology initiated a review of the electrical generation deployment needed to provide increased capacity and to continue to provide 30% of the electrical supply from emissions-free sources. The role of nuclear power obviously would play a significant role in achieving the latter goal. The initial challenge was to identify the reactor designs that would satisfy the estimated additional 50,000 MWe nuclear capacity required on the grid by 2020, with the first reactors being deployed around 2010. Consequently, DOE's Nuclear Energy Research Advisory Committee (NERAC) conducted a study to identify the actions needed by government and industry to overcome the technical and regulatory barriers to new plant construction. The participants in this assessment of potential reactor designs included international representatives from reactor vendors, utilities, national laboratories and academia. The results of this study were documented in an October 2001 report titled "A Roadmap to Deploy New Nuclear Power Plants in the U.S. by 2010" (Reference 10). Six designs were found to be at least possibly deployable in the U.S. by 2010, provided new plant orders were placed by 2003. These U.S. Near-Term Deployment plants are as follows:

Advanced Boiling Water Reactor (ABWR)
Advanced Pressurized Water Reactor 1000 (AP1000)
European Simplified Boiling Water Reactor (ESBWR)
Gas Turbine Modular High Temperature Reactor (GT-MHR)
Pebble Bed Modular Reactor (PBMR)
Siedewasser Reactor-1000 (SWR-1000)

DOE identified other designs which had significant international support, and were deployable by 2015 or earlier. These International Near-Term Deployment plants are as follows:

<u>Advanced Boiling Water Reactors</u>
Advanced Boiling Water Reactor II (ABWR II)
European Simplified Boiling Water Reactor[2] (ESBWR)
High-Conversion Boiling Water Reactor (HC-BWR)
Siedewasser Reactor-1000[2] (SWR-1000)

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[2]These reactors are also identified as U.S. Near-Term Deployment plants

Advanced Pressure Tube Reactors
Advanced Pressurized Water Reactor 600 (AP600)
Advanced Pressurized Water Reactor 1000[3] (AP1000)
Advanced CANDU Reactor 700 (ACR-700)

Advanced Pressurized Water Reactors
Advanced Power Reactor 1400 (APR-1400)
Advanced Pressurized Water Reactor Plus (APWR+)
European Pressurized Water Reactor (EPR)

Integral Primary System Reactors
Central Argentina de Elementos Modulares (CAREM)
International Modular Reactor (IMR)
International Reactor Innovative and Secure (IRIS)
System-Integrated Modular Advanced Reactor (SMART)

Modular High-Temperature Gas-Cooled Reactors
Gas Turbine Modular High-Temperature Reactor[3] (GT-MHR)
Pebble Bed Modular Reactor[3] (PBMR)

It is noteworthy that the ABWR design and the AP600 design (which is the precursor to the AP1000) have been certified under 10 CFR Part 52. The other DOE U.S. Near-Term Deployment plants and the ACR-700 and IRIS reactor designs are addressed within the RES Infrastructure Assessment document. (Note: the pre-application review for the SWR-1000 is currently expected to begin in mid-2004.)

In addition to these Near-Term Deployment reactor designs, DOE considered a number of more advanced design concepts (GEN IV designs) which addressed the entire fuel cycle from ore extraction to final waste disposal (e.g., actinide management) and also address the potential interactions between higher temperature nuclear reactor designs and, for instance, other energy conversion systems such as hydrogen production. Nearly 300 potential reactor designs water-cooled, gas-cooled, or liquid metal-cooled cores were evaluated by DOE against specific sustainability, safety, and reliability, and economics goals.

The resulting "Generation IV Roadmap" (Reference 4) issued in 2002 defined the necessary research and development to support a generation of innovative GEN IV reactor designs believed to be deployable by either 2020 or 2025:

Deployable by 2020
Sodium-Cooled Fast Reactor (SFR)
Very High Temperature Reactor (VHTR)

[3]Ibid.

Deployable by 2025
Gas-Cooled Fast Reactor (GFR)
Lead-Cooled Fast Reactor (LFR)
Molten Salt Reactor (MSR)
Supercritical-Water-Cooled Reactor (SCWR)

These GEN IV reactor design concepts are not addressed in the RES Infrastructure Assessment document because they are in a preliminary stage of development and there are some significant areas of research and development still to be addressed. There are currently no vendor requests for NRC reviews of these designs.

Given the extensive set of potentially deployable reactors over the next two decades, RES has a daunting task to chart a workable regulatory research plan that covers all eventualities. Thus, the plan is expected to be a living document and will need to be updated to reflect the listed domestic and international concepts in some coherent way. The objective of the plan, given the significant uncertainties as to the course and timing of domestic deployment efforts by industry, should be to position the agency to review applications as they are received. RES will need to proactively acquire an early indication of concepts that are moving towards the regulatory interface, and to put in place mechanisms to revise plans and redirect funding in an appropriate and responsive manner. Technologically neutral plans may be useful, but are likely to fall short of the specificity needed by NRR to efficiently support licensing actions.

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11. ABSTRACT *(200 words or less)*

The objective of this report by the Advisory Committee for Reactor Safeguards (ACRS) is to provide comments and advice on the Office of Nuclear Regulatory Research (RES) document "Advanced Reactor Research Infrastructure Assessment" (Reference 1), which identifies gaps in technology that need to be filled prior to the potential certification of a number of new reactor designs. These designs range from those which have evolved from the current boiling and pressurized water reactor designs, i.e., advanced light water reactor (ALWR) designs, to those using significantly different technology, e.g., high-temperature gas-cooled reactors (HTGRs). The range of designs addressed in this assessment are representative of the designs vendors have expressed an interest in submitting to NRC for certification. We note that the U.S. nuclear industry has not constructed a new reactor for over two decades and has focused on the operational and services aspects of the existing reactor fleet. As a result, there are significant regulatory, management, and technical challenges to be faced with the certification, construction, and deployment of new reactor designs, especially those based on significantly different technology.

This ACRS report concentrates on reactor safety issues. It does not cover items relating to nuclear materials and waste safety or to safeguards. Safeguards will be addressed in a separate ACRS activity. We have not commented on the several management issues that are discussed in the RES Infrastructure Assessment but recognize that these issues are important.

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