26

From: Thomas Bergman
To: Jacqueline Raines
Date: Mon, Apr 22, 2002 3:56 PM
Subject: Fwd: Re: Draft Advanced Reactor Research Plan

Please provide a copy to Carl, per his request.

It is 115 pages. Just so you know before you hit "Print".

Thanks.

Forwards TAB 037,
TAB 037.1 and TAB 037.1.1
~~[illegible]~~.

S-3

TAB 036

From: Stephen Koenick
To: Thomas Bergman
Date: Mon, Apr 22, 2002 3:02 PM
Subject: Fwd: Re: Draft Advanced Reactor Research Plan

Per your request

TAB 037

From: Mel Fields
To: David Skeen; Ronald Frahm; Stephanie Coffin; Stephen Koenick; Timothy Frye
Date: Mon, Mar 18, 2002 4:38 PM
Subject: Fwd: Re: Draft Advanced Reactor Research Plan

Attached is an electronic copy of the plan, including the missing pages.

>>> Timothy Frye 03/18/02 09:08AM >>>
Haven't seen a hard copy or electronic copy. Would prefer an electronic copy

>>> Ronald Frahm 03/18/02 08:23AM >>>
Good morning all,

Has anyone seen a copy of this plan yet? Better yet, does anyone have an electronic copy? - - Ron

>>> Stephen Koenick 03/15/02 09:21AM >>>
To the addressed,

Apologize for the quick turn around. Feel free to send the comments to Amy or myself. Note, this review will not serve as endorsement of the plan, only provide initial "high level" feedback. Attached is the guidance previously developed which can help focus the review.

Feel free to contact either Amy or myself if you have any questions.

Thank you,

Stephen S. Koenick
Project Manager
U.S. Nuclear Regulatory Commission
voice: 301 415-1239
email: ssk2@nrc.gov

>>> Mel Fields 03/15/02 09:05AM >>>
The schedule for providing comments directly to Steve is COBMarch 27th.

Please confirm that your divisions will be able to satisfy this date.

tnx

CC: Amy Cubbage

TAB 037.1

ADVANCED REACTOR

RESEARCH PLAN

Office of Nuclear Regulatory Research
March 2002

CONTENTS

DRAFT ADVANCED REACTOR RESEARCH PLAN

I. INTRODUCTION

On February 13, 2001, the Commission issued a Staff Requirements Memorandum (SRM) for COMJSM-00-0003, "Staff Readiness for New Nuclear Plant Construction and the Pebble Bed Modular Reactor." The SRM directed the staff to "assess its technical, licensing, and inspection capabilities and identify enhancements, if any, that would be necessary to ensure that the agency can effectively carry out its responsibilities associated with an ESP application, a license application, and the construction of a new nuclear power plant." In addition, the staff was directed to "critically assess the regulatory infrastructure supporting both Part 50 and Part 52, and other applicable regulations, and identify where enhancements, if any, are necessary." In response to this SRM, the staff prepared an information paper titled "Future Licensing and Inspection Readiness Assessment [FLIRA]," SECY-01-0188, October 12, 2001, which provided an assessment of its technical, licensing, and inspection capabilities, and enhancements necessary to support future licensing of the high temperature gas-cooled reactors (HTGRs) and the advanced light water reactors (ALWRs). In the FLIRA report, the staff also made a commitment to develop an advanced reactor research plan that would be used to develop and guide a comprehensive advanced reactor research program. It was envisaged that the research plan would help formulate, and set directions for research programs, programs to develop a regulatory framework for advanced designs, and analytical tools and experimental data to independently assess their safety capacity. To fulfill the FLIRA commitment to the Commission, the staff developed this research plan to build a research infrastructure that would be used to support independent review of advanced reactor designs. The implementation of this plan will include full participation of NRC individual staff from the Office of Nuclear Reactor Regulation (NRR), Office of Nuclear Material Safety and Safeguards (NMSS), and Office of Nuclear Regulatory Research (RES).

In developing this plan, the staff focused on critical research areas and information to technically support an advanced reactor license submittal review. At this point, the plan does not delineate what research will be conducted by the NRC verses the developers of new reactor designs, but rather on the infrastructure necessary to independently confirm the safety case of the designs. This includes information gaps and the tools, data, and expertise needed to fill the gaps.

It is recognized that the primary responsibility of an applicant or designer is to demonstrate that they have met the required level of safety or regulatory criteria. Most of NRC regulations were developed for LWRs, however, and in certain cases these regulations may not completely apply to a future licensing application. In such cases, NRC may need to develop new safety limits and data base, to assess safety margins, or issues that could extend beyond the design basis. It is expected that these activities would also be captured under this plan.

It is recognized that not all of advanced reactor research has to be done by NRC. Information can be obtained through domestic and international cooperation, as well as the developers themselves. Accordingly, resource estimates in the plan do not credit research performed by others outside the NRC, although many of these programs are recognized in the plan, and collaborations with various organizations will be pursued during the startup phase.

II. BACKGROUND

On June 19, 2001, the Commission issued an SRM that approved the staff's plan (SECY-01-0070) to proceed with pre-application review of the PBMR, and develop necessary research infrastructure to support the advanced reactor licensing process. Pre-application reviews provide for early interaction between the USNRC and the reactor designers to identify key safety and policy issues, propose paths for their resolution and establish a regulatory framework providing guidance on applicable requirements that are different from current requirements. In addition, insights from the pre-application review help prepare the staff and research infrastructure for near term licensing applications, and technical basis for review criteria. Specific research activities and infrastructure needs that are identified during the pre-application review, will enter into the Planning, Budgeting and Performance Management (PBPM) process, and assigned resources accordingly.

In addition to the PBMR, the nuclear industry has been exploring new, innovative, and revolutionary reactor design concepts and features to simultaneously attain performance and economic improvements, while preserving the defense-in-depth philosophy by employing multiple barriers to protect public health and safety. New reactor designs being pursued include the Gas Turbine–Modular Helium Reactor (GT-MHR) – another HTGR, the AP-1000 and the International Reactor Innovative and Secure (IRIS) – two ALWRs. Many of these present new challenges to the NRC. To effectively and efficiently address these challenges, modification to the existing regulatory framework may be necessary, along with a sound technical basis to support the associated regulatory decision making process.

While it is the responsibility of the applicant and designer to demonstrate the safety level of proposed new reactor designs and technologies, the NRC will conduct as necessary, supporting research to help establish the technical basis and acceptance criteria for the safety case. In this regard, the term "research" encompasses activities that aim at either applying the existing knowledge and/or tools, or creating new knowledge and/or tools. It is expected that applicants will provide the supporting arguments and documentation based on the existing knowledge and their own research results. However, this information will be independently examined by the staff to judge whether or not a safety case has been made. At a July 2001, workshop on Advanced Reactors sponsored by the Advisory Committee on Reactor Safeguards (ACRS), Committee members as well as the members of public attending the workshop, advocated the need for the NRC to conduct independent research to establish the technical basis for accepting new reactor designs.

While assessing challenges posed by the new reactor designs and technologies, the staff will need to consider in implementing the plan, what research would be conducted by the applicant(s) as part of their license application, the needs of the licensing office, and adjust accordingly. Additionally, research may be also be conducted by others with a vested interest, (e.g., generic and technology-neutral research sponsored by Department of Energy (DOE), or industry-supported organizations). Experience with AP600 certification, for example, indicates that the scope, schedule, and resources for such research programs are extensive, and that the staff could benefit from world-wide developmental research and experience. Mindful of our respective roles, and consistent with the NRC Strategic Plan, the NRC will continue to seek out opportunities to interact with and, where appropriate, initiate cooperative programs with other agencies and organizations. These include US universities and domestic organizations such as

Department of Energy (DOE), Nuclear Energy Institute (NEI), Electric Power Research Institute (EPRI), and international nuclear organizations such as Nuclear Installations Inspectorate (NII), International Atomic Energy Agency (IAEA), Nuclear Energy Agency (NEA), the European Commission (E.C.). Furthermore, in addition to off-setting costs, significant efficiencies can be gained by sharing research facilities and leveraging resources to minimize duplication. Steps to ensure that the regulatory process does not impede the use of new technology to improve safety or reduce costs is an important part of the NRC's Strategic Plan.

III. ROLE OF NRC RESEARCH

To insure that regulatory decisions are based on a sound technical basis, NRC conducts research to obtain needed information to establish the safety standards used to ensure public health and safety and protection of the environment. NRC also performs research to explore issues involving large uncertainties, and develop independent capabilities to enable the staff to review applicants submittals. The duration of this research varies between short-term effort to respond quickly to emerging issues identified by the user offices, to long-term efforts intended to develop, support, and maintain the agency's infrastructure. Long-term research is more forward-looking, and relates to evolving technologies, or issues that may become important regulatory concerns in the future. These concerns usually arises from the examination of industry trends and insights, insights that help the NRC foresee where information will be needed to respond to future regulatory issues.

In general, NRC research centers around the development of expertise, tools and methods that are needed to support the Agency's mission in understanding and resolving potential safety issues. Tools such as computer codes and experiments that generate data to validate these codes, have played an important role in that mission. For the HTGRs, experimental data will be needed to ascertain fission product chemistry, for example, in an air- and/or moisture-added environment, and address transport and plate-out of fission products in order to quantify and characterize the radiological source term. Most of the existing NRC codes, however, were developed for LWR applications. These codes have provided the NRC with the ability to perform independent analyses which are an integral part of the licensing process.

The NRC's statutory obligation demands that NRC institute a licensing process where decisions on significant safety issues are of high-quality, technically sound and are supported by robust research. In planning research activities, the focus is primarily on those areas where important gaps exist; (e.g., in technological knowledge or in understanding risk significant uncertainties or where the degree of conservatism in safety margins may not be well characterized or understood). Computer models validated by experiments are important tools used to bridge the technological gaps.

Another important facet of the planned research is materials testing, and codes and standards development which generally involves a consensus process. Such a process takes a long time to either develop new codes and/or standards, or obtain NRC endorsement of existing code cases. Consequently, as part of the planning, sufficient lead time should be allowed. As in the past, the pre-application review process is being used to identify the necessary codes and standards early in the process.

The general principle to be used to fund a specific research is that if the data are needed to support regulatory decisions on safety cases for a particular reactor design, the applicant would be responsible for the data. If NRC believes that it is important to explore issues involving uncertainties, or when it is necessary to develop independent capabilities, NRC resources would be used. When both the NRC and industry benefit from research, or if it is difficult to determine whether industry or NRC is the beneficiary, research can be jointly funded by industry (or one segment of the industry) and NRC. It is essential, however, that NRC's independence not be compromised in the process, the quality and integrity of the data be maintained, and that all legal and administrative requirements are met. The process equally

applies to relationships with other government agencies such as the Department of Energy (DOE). While research on advancing commercial reactor designs is conducted by DOE, NRC's focus is on the safety standards that these new designs must meet. This may necessitate additional NRC research beyond that conducted by DOE or the applicant of new advanced reactor designs. Research needed to establish acceptance criteria associated with a new safety standard, requirements, or to address specific issues for a particular reactor design, can be funded independently by NRC, in cooperation with DOE, or through international cooperative agreements, provided NRC's independence regarding regulatory decision-making is maintained.

It should be recognized, however, that even a well-funded, and appropriately focused program of nuclear safety research cannot transform the regulation of advanced nuclear power plants into a process in which decisions flow exclusively from scientific and technical knowledge. Defense-in-depth, safety margins and conservative decisions will need to be considered to offset limitations in state-of-the-art knowledge and understanding. Similar to the existing reactor licensing and other complex technologies, advanced reactor regulation will be a complex blend of applying technical knowledge within the context of Commission policy, and prudent regulatory decision-making. Therefore, priorities set within the program will take into consideration the relative importance of the activity to understand the real safety issues, the risk significance of the issues, and the associated cost-benefit considerations. This will be especially important as new technology is introduced or new safety issues are identified. The staff will continue to interact with applicants, vendors, and others as the technologies evolve, so that the NRC will be prepared to respond effectively.

In the course of reviewing new reactor designs, as well as results from research findings, a new set of questions may be raised. The importance of answering the new set of questions by examining their pertinence to the real safety issues being explored, poses a significant challenge to the NRC. The benefit of this research plan is that it provides a rationale for identifying the key research areas, assessing its priority, and identifies the expected end-product(s) as well as the anticipated completion date of specific research activity. In addition, routine peer reviews of the research products will be conducted to instill confidence in the scope and quality of the research, and will include frequent interactions with the ACRS to obtain feedback and guidance as well as strong involvement of NRR and NMSS to keep activities focused.

IV. OBJECTIVES

The advanced reactor research plan will be used to generate and implement a research infrastructure necessary for advanced reactors. The purpose of the plan is to identify:

- Key research areas and activities,
- Technical and safety issues and pathways to resolution,
- Methods and tools to address technical or safety issues,
- Technical staff responsibilities,
- Links between technical disciplines and flow of information,
- Key research output results and link to the regulatory process,
- Priorities used to allocate resources,
- Key milestones and resources over a 5 year period (FY 02-FY 06).

Additionally, the plan will provide a platform for communicating program objectives and goals, and receiving feedback from internal and external stakeholders. The research activities contained within the scope of the plan (which includes PBMR, GT-MHR, IRIS, and AP-1000 designs) primarily stem from the need to technically support an advanced reactor regulatory process. Two aspects were considered essential: (1) research that would be needed to establish the technical basis for regulatory decision-making, and (2) research to address uncertainties and understanding safety margins and failure points. In many ways, the first depends on the second, since building a sound technical basis will require a deep understanding of the technology, its application, and the inherent uncertainties.

In developing this plan we have maintained a technology-neutral perspective. However, as the plan evolves, it discriminates between different technologies (i.e., LWR vs. ALWR and/or LWR vs. HTGR) when design-specific safety issues are addressed, or future modifications to the existing analytical codes for specific applications are discussed. It is expected that at some point, planning would become issue-driven as the application of research tools becomes more design-specific.

The plan also integrates ongoing research initiatives in both the domestic and international arena. Budget estimates are determined in the absence of more detailed information on the role of the industry in providing some of the identified needs. As more information becomes available, we will update these resources to reflect only those activities that will require NRC funding consistent with FY2003-2005 budget projections.

It is envisioned that the plan would be maintained as a living document, and will be updated as appropriate to accommodate any new issues. Future updates will identify (1) any new information from the applicant; and (2) the NRC's research plans to independently confirm the applicant's findings. The second reduces uncertainties and increases confidence in technical and policy issue resolution. Common to both are the safety and design technical issues that need to be addressed, and the tools, methods, data, and expertise that will be required to identify pathways to resolution. Research results could be used to support a safety evaluation report, or result in guidance or review documents, such as, Regulatory Guides, Standard Review Plan, or NUREG reports.

V. PLAN STRUCTURE

V.1 INFORMATION GATHERING

The scope of the advanced reactor research plan includes two ALWRs (AP-1000 and IRIS), and two HTGRs (PBMR and GT-MHR). The existing NRC research and regulatory infrastructure, primarily supports licensing of the current generation of LWRs. Although there are several areas where research infrastructure needs to be improved to address ALWRs, such as the IRIS design, most of the needed improvements are related to the HTGRs.

In drafting this plan, the staff benefitted from the week-long DOE-sponsored HTGR training course (September–October, 2001), and various technical information gathering activities. These activities included interactions with worldwide experts on gas-cooled technology, and input from the NRC Workshop on the HTGR Safety and Research Issues and development held October 10–12, 2001. Many of the insights generated during the workshop discussions were taken into consideration. Workshop participants assigned priorities to research areas, and identified several opportunities for international cooperative research that draw upon existing domestic and international experience. Staff also participated in and capitalized on feedback from the OECD/CSNI "Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs," held February 18–23, 2002. Insights from NUREG-1802, "Role and Direction of Nuclear Regulatory Research" were utilized, and interactions with the Advisory Committee on Reactor Safeguards are planned for the second quarter CY 02.

The staff also took advantage of the DOE-sponsored Modular High-Temperature Gas-Cooled Reactor (MHTGR) pre-application review that was performed in the late 1980s and early 1990s, and the ongoing PBMR pre-application review. The MHTGR review had been supported by an integrated preliminary design document and associated probabilistic risk assessment (PRA). Insights from these documents have helped shape this plan. Technical staff also visited countries with HTGR experience, including Germany, Japan, China, South Africa, and the United Kingdom. These visits focused on technical and safety issues associated with HTGR fuel performance and qualification, nuclear-grade graphite behavior, and high-temperature materials performance. Technical exchanges and international agreements are currently being discussed in several areas, including graphite behavior, high-temperature materials research, fuel performance, and codes and standards.

V.2 STRUCTURE

To facilitate the identification of research areas important to the development of a infrastructure, a top down approach was used as shown in Figure 1. (Note that Fig 1 does not contain all the research activities, but simply used to assist in the undertaking.) To achieve the goal of protecting public health and safety and environment, the staff utilized the NRC strategic plan and aligned research programs according to the three strategic arenas: Nuclear Reactor Safety, Nuclear Materials Safety, and Nuclear Waste Safety. The forth strategic area International Nuclear Safety support is not addressed separately, but integrated throughout the

plan and planning process. Key outputs were identified and linked to key research areas and activities as described below.

V.2.1 Nuclear Reactor Safety Arena

To have reasonable assurance of adequate protection of public health and safety, a licensee must demonstrate compliance with NRC regulations. The current regulations established for LWRs using defense-in-depth principles, and conservative practices, provide a degree of margin that might not be readily applicable to PBMR or GT-MHR advanced reactor designs in all areas. To support the Nuclear Reactor Safety arena for these advanced designs, a new regulatory framework will be developed, which consists of developing the guidelines and criteria that will be used to formulate the regulation and associated regulatory guide.

Research to develop data and tools were grouped to align with the four cornerstones of reactor safety:

1. Accident Prevention
2. Accident Mitigation
3. Barrier Protection
4. Offsite Protection

Fig 1 identifies the key research areas, and some of the associated activities are listed below.

	Key Research Area	Activities
1.	Development of Regulatory Framework	Risk-informed and performance-based decision making criteria
2.	Accident Analysis	PRA, human factors, and I&C
3.	Reactor/Plant Analysis	Thermal-fluid dynamics, nuclear analysis and fission product transport
4.	Fuels Analysis	Fuel performance testing, and fuel qualification
5.	Materials Analysis	Graphite and materials performance
6.	Structural Analysis	Containment/confinement performance, external challenges
7.	Consequence Analysis	Dose calculations, environmental impact studies

In-depth discussion of these activities are provided in the plan. The key research areas and activities are generally associated with an output that either establish a technical basis for resolving specific safety issues, or supports another research area trying to do so. Once the issues become well defined, then specific analytical methods, experimental facilities, and

expertise necessary to support resolution can be defined. This information is subsequently used to identify research infrastructure needs and schedule and resource projections.

V.2.1.1 Information Transfer and Technology

Information flow between the technical groups and framework is shown in Figure 2. The process can be described in four parts:

- information in the form of data and analytic results generated by the fuels-, materials-, and structural technical groups feed the reactor/plant system analysis. In turn, reactor/plant analysis provides key information on plant operating conditions, and accident conditions, back to the fuel, materials, and structural analyses technical groups.

- Once determined, insights and data generated by the reactor/plant analysis (e.g., success criteria), together with human factors considerations, I&C, and modeling assumptions, feed the PRA and associated accident analysis activities. Accident analysis research identifies accident scenarios and frequencies for further and more detailed reactor system analysis and consequence analysis.

- Insights from the accident analysis and consequence analysis feed the regulatory framework. As the regulatory decision-making element in the process, the framework would ultimately prescribe the regulatory requirements

- Information from the framework is provided to all technical areas, from which safety related systems, structures, and components can be determined, along with the codes and standards that would need to be met.

From the above discussion it is important to note that the research plan does not generate a system of discrete and isolated technical disciplines working alone, but an integrated system that is risk-informed and technically based through the key research areas.

Identification of key accident scenarios is an important aspect of a licensing process. These events will typically drive the regulatory decision-making process not only because they impact the safety system classifications, but also because their consequences would ultimately influence the minimum safety criteria that a plant design would have to meet. Thus, accident analysis, consequence analysis, and regulatory framework have a direct link with each other. Once significant accident scenarios are identified for a plant design, reactor/plant analysis can be performed, and the results used to place performance limits on the reactor fuel, reactor internals, and other structural materials. Additionally, reactor/plant analysis and associated sensitivity studies can be used to develop PRA insights, which are crucial to a robust accident analysis. As the process is implemented, risk perspectives will support the regulatory framework decision-making activities, and research that is needed to support the framework.

Various sub-sections in Section VI of this research plan describe details of research infrastructure that is needed to support a defensible review process to ascertain safety of the new plant designs.

V.2.2 Nuclear Materials Safety and Nuclear Waste Safety Arenas

Advanced reactor research activities for the Nuclear Materials Safety and Nuclear Waste Safety arenas will focus on supporting regulatory activities at the front and back ends of the advanced reactor fuel cycles:

Front end of fuel cycle – uranium enrichment, fuel fabrication, transportation, and storage
Back end of fuel cycle – spent fuel storage, transportation, and disposal

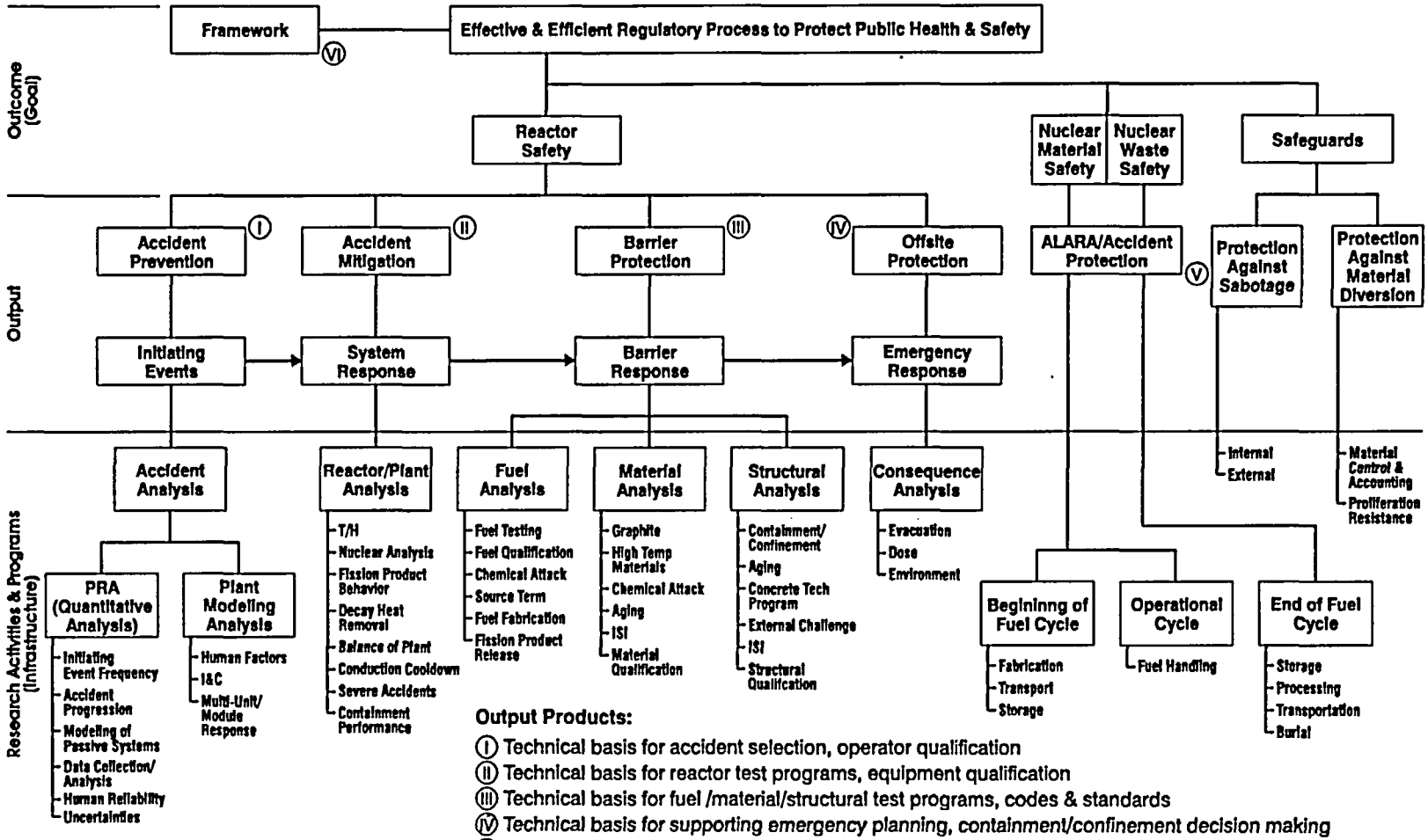
In-depth discussions of anticipated NRC research activities and infrastructure needs associated with these regulatory domains are provided in the plan.

V.2.3 Safeguards and Security Arena

Advanced reactor research efforts for the arena of Safeguards and Security will support the regulatory offices in the assessment of proliferation potential and the evaluation of security measures and material control and accounting systems needed for preventing and detecting nuclear material diversion throughout the proposed advanced reactor fuel cycles. Discussions of anticipated research activities to support these regulatory domains are included in the plan.

In addition, RES will support other offices and agencies as requested for assessing and limiting the vulnerability of advanced reactor plants and fuel cycle activities to sabotage and outside threats. This coordinated research support will be responsive to new issues emerging from government-wide initiatives for Homeland Security.

Advanced Reactor Research Infrastructure
Key Research Areas and Areas for Examination



Output Products:

- (I) Technical basis for accident selection, operator qualification
- (II) Technical basis for reactor test programs, equipment qualification
- (III) Technical basis for fuel /material/structural test programs, codes & standards
- (IV) Technical basis for supporting emergency planning, containment/confinement decision making
- (V) Technical basis for regulating nuclear material, waste, safeguards
- (VI) Technical basis for risk-informed, performance based regulations

Fig 1. Key research areas for examination.

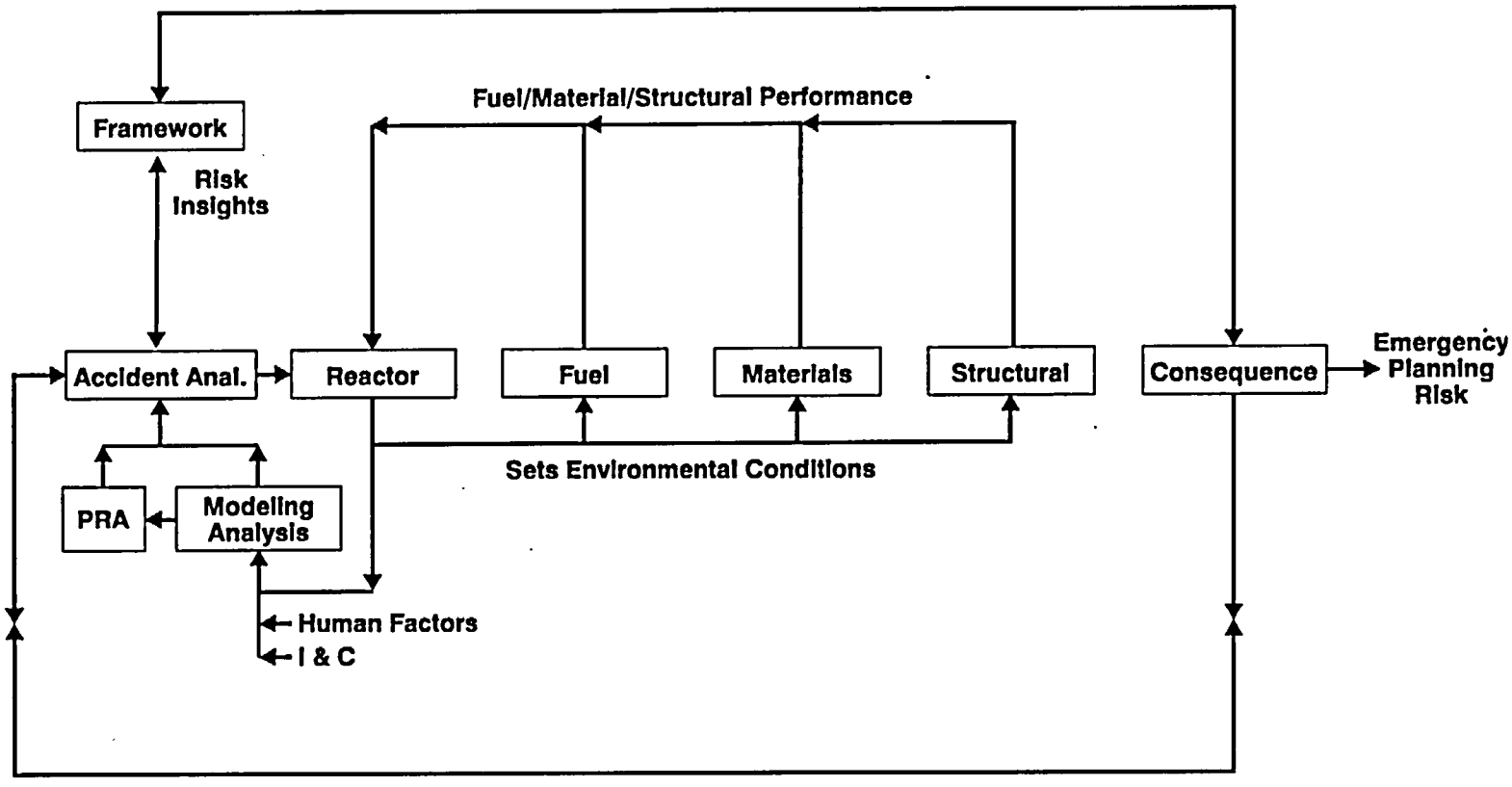


Fig 2. Information flow between technical groups.

PART 1

VI. KEY RESEARCH AREAS AND ACTIVITIES

VI.1 GENERIC REGULATORY FRAMEWORK DEVELOPMENT

V1.1.1 Description of Issues

A regulatory framework is needed that can be applied to license and regulate advanced reactors. This framework is needed because while the NRC has over 40 years of licensing and regulating nuclear power plants, this experience (e.g., regulations, regulatory guidance, policies and practices) has been focused on current light water-cooled reactors (LWRs) and has limited applicability to advanced reactors. There will be design and operational issues associated with the advanced reactors that are distinctly different from current LWR issues. However, NRC LWR experience can contribute and provide insights or "lessons learned."

The most important insight from this experience is the recognition of the need for the development of a licensing framework. This framework would help to ensure that a structured and systematic approach is used during the development of the regulations that will govern the design and operation of advanced reactors. This approach will ensure uniformity, consistency, and defensability in the development of the regulations, particularly when addressing the unique design and operational aspects of advanced reactors.

The framework for current LWRs has evolved over five decades, and the bulk of this evolution occurred without the benefit of insights from probabilistic risk assessments (PRAs) and severe accident research. It is anticipated that PRA will play a greater role in the licensing and regulation of advanced reactors and, as such, the framework needs to appropriately integrate PRA results and insights.

The proposed tasks would first develop an approach (and ultimately a framework) that would be applicable to all of the advanced reactor concepts currently under consideration. This approach, referred to as "reactor-neutral," would take full advantage of lessons learned from prior regulatory experience and assure an effective use of both deterministic and probabilistic methods in licensing and regulating advanced reactors. The approach would subsequently be used to develop reactor-specific regulations for such designs as the PBMR, GT-MHR, and IRIS.

VI.1.2 Risk Perspectives

It is expected that future applicants will rely on PRA and PRA insights as an integral part of their license applications. In addition, it is further expected that the regulations licensing these advanced reactors will be risk-informed. Both deterministic and probabilistic results and insights will be used in the development of the regulations governing these reactors. Consequently, a structured approach for a regulatory framework for advanced reactors that provides guidance about how to use PRA results and insights will help ensure the safety of these reactors by focusing the regulations on where the risk is most likely while maintaining basic principals, such as defense-in-depth and safety margin.

V1.1.3 Objectives and Planned Activities

VI.1.3.1 Plan

NRC research efforts required to systematically develop a suitable framework for advanced reactor licensing and regulation will be carefully planned. As currently envisioned, the plan will include the major tasks discussed below.

VI.1.3.2 Approach

An approach will be developed to prepare a licensing framework for advanced reactors. This approach will identify the scope and level of detail of the framework along with certain boundary conditions, ground rules, and assumptions, etc., that will be used in the development of the framework. Experience gained in NRC's Option 3 efforts to risk-inform regulatory requirements for current LWRs provides a starting point for the development of an appropriate regulatory framework for advanced reactors. The approach will include both qualitative and quantitative aspects as depicted in Figure 1. An important qualitative aspect of the approach is a hierarchal structure that supports regulatory goals including the goal of protecting public health and safety and the strategic performance goals of the NRC's Strategic Plan. It is anticipated that defense-in-depth will remain a guiding reactor safety strategy. An important quantitative aspect of the approach is the development of useful risk guidelines for advanced reactors from the Safety Goal Policy Statement. Safety Goal issues that arise in developing the quantitative guidelines will have to be resolved. In addition, guidance in the Commission's advanced reactor policy statement will be used in the development of the advanced reactor licensing framework. The advanced reactor policy statement included the expectation that, as a minimum, advanced reactors will be required to provide the same level of protection to the public that is required for current generation LWRs. It also stated the expectation that enhanced margins of safety and simplified, inherent, passive, or other innovative means to accomplish their safety functions will be utilized.

VI.1.3.3 Framework

A reactor-neutral or globally applicable licensing framework will be developed for advanced reactors that includes PBMR, GT-MHR, and IRIS. The purpose of the framework is to develop a process (i.e., guidelines) that will be used to formulate a reactor-neutral or global set of regulations for advanced reactors. Figure 2 is a general illustration of the development of the reactor-neutral framework. The process starts using safety criteria and regulatory guidelines determined to be applicable to advanced reactors, and those safety related areas identified as being important to regulating these advanced reactors. These two items are then considered together to develop a set of specific performance goals. Explicit in the performance goals will be the level of detail believed to be needed for licensing. The process is iterative, and the performance goals are revised as new information becomes available. A set of reactor-neutral regulations are then defined based on the performance goals. A key product of the framework will also be guidance regarding appropriate uses of strategies and tactics to compensate for uncertainties inherent in both deterministic and probabilistic safety analyses, including the consideration of defense-in-depth and safety margin.

VI.1.3.4 Reactor-Neutral Regulatory Requirements

A set of regulations that are reactor-neutral or globally applicable to all reactor types currently under consideration will be developed. The licensing framework will be used to identify and formulate what regulations are needed.

VI.1.3.5 Reactor-Specific Regulations/Regulatory Guides

As currently envisioned, as much reliance as possible will be placed on the use of regulatory guides rather than on formal reactor-specific regulations to supplement the reactor-neutral regulatory requirements. The reactor-specific regulatory guides will not provide the detailed guidance for implementation of specific technical requirements, but will provide the guidelines for expanding the reactor-neutral regulations to account for reactor-specific considerations. Regulatory guides can provide the designer with useful flexibility in design and operation while still satisfying formal licensing requirements. However, it is envisioned that certain reactor-specific regulatory areas may need to be addressed formally by regulations. The reactor-neutral licensing framework will be used to identify and prepare both reactor-specific regulations and regulatory guides as needed. These products will be developed for each of the advanced reactor designs under consideration.

VI.1.3.6 Oversight/Peer Review

Considering the scope of the proposed effort and its potential impact on advanced reactor licensing and regulation, appropriate oversight and peer review is deemed essential. Arrangements for such reviews will be initiated during the planning task.

VI.1.4 Related International Cooperation

The South African government has issued a Basic Licensing Guide for the Pebble Bed Modular Reactor (Document LG-1037, Rev 0).

VI.1.5 Schedule

The schedule for the proposed tasks is summarized below. The overall duration of the effort is four years to support having the licensing framework in place in time to support design certification application for the PBMR, GT-MHR and IRIS. The overall approach and initial framework, which are closely related, come first. Draft products for these tasks will be developed in the first year. In the next two years, a draft set of reactor-neutral regulations and reactor-specific regulatory requirements and guidelines will be developed. The approach, and framework will be finalized early in the fourth year based on "lessons learned" from the draft regulations. Reactor-neutral and reactor-specific regulations and guidelines will be finalized in the fourth year. Oversight/peer review meetings will be scheduled at key milestones throughout the process to integrate the work as it is performed. This direct interface between the peer review group and the developers of the regulations will make the development process more efficient and effective. As noted by the schedule, this work will be performed in an "iterative" manner, such that as knowledge is gained, it is continually fed back into the process.

VI.1.6 Priority

This work is considered high priority since its outcome will have a large influence on the types of data, analyses, and methods that applicants and the NRC will need for future plant licensing.

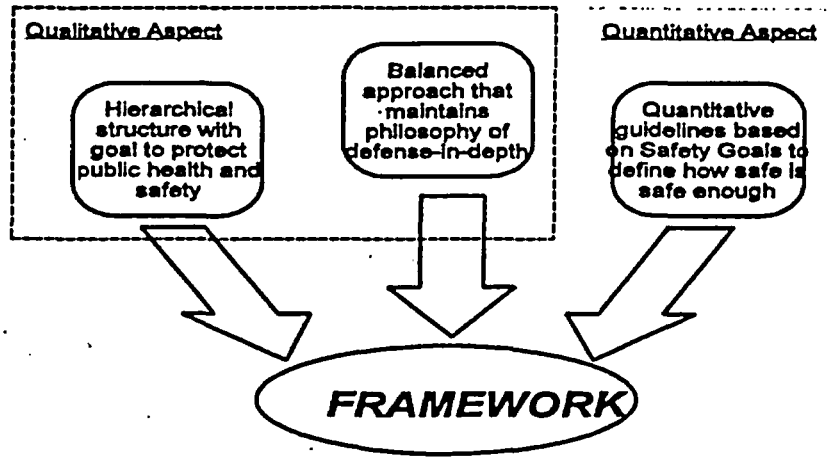


Figure 1 - Aspects of the Framework

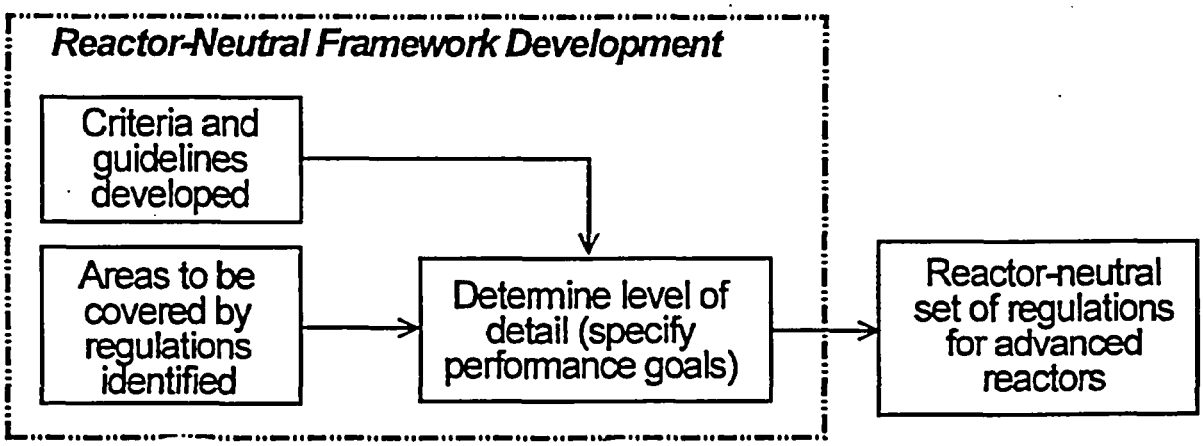


Figure 2 - Development of a Reactor-Neutral Licensing Framework

VI.2 REACTOR SAFETY

VI.2.1 Accident Analysis

VI.2.1.1 Probabilistic Risk Assessment

VI.2.1.1.1 Background

Future licensees have indicated that PRAs will be an integral part of their application and will play a crucial role in the licensing process for new reactor designs. Therefore, the NRC should be prepared with the tools and expertise to perform an independent review of the PRAs submitted as part of the licensing applications for either HTGRs, such as the PBMR and GT-MHR, or ALWRs such as AP 1000 and IRIS.

For the past 30 years, the NRC has performed several PRAs, and has promoted its use as a means of developing nuclear power plant risk perspectives and identifying improvements. As a result, the NRC has developed the capability to use PRAs in regulatory decision-making for current generation reactors. This capability is founded on the staff's in-depth understanding of the techniques and data employed in a PRA, the design and physical characteristics of the reactors modeled, and how the design and characteristics are modeled in a PRA in terms of underlying hypotheses and data.

However, advanced reactors (especially PBMR, GT-MHR, and IRIS) are new designs and, therefore, the current PRA experience will need to be expanded to capture the new technology. The limitations of current PRA experience applies to system modeling approaches and associated underlying hypotheses (e.g., treatment of passive systems); to risk metrics used (e.g., core damage frequency or large early release may not be the best figure of merit to some proposed advanced reactor designs); to failure data; and most importantly, to the design, materials, systems, and safety approach. These limitations will be addressed as part this work. Extensive use will be made of the NRC-reviewed existing HTGR PRA. The tools, expertise, and data (including information related to uncertainties) will be developed to enable the staff to evaluate advanced reactor PRAs.

This work will interface with virtually every other area of this plan. Given that PRA is an iterative process, knowledge of reactor systems, fuels, materials, human performance, and instrumentation and controls (I&C) will be used for postulating accident initiators, modeling of systems, and quantifying accident sequences. The results will indicate what issues are important from a probabilistic perspective and what areas need investigation as part of this research plan.

VI.2.1.1.2 Purpose

The purpose of this work is to develop the methods, expertise, and technical basis needed for an independent staff review of a PRA submitted as part of an advanced reactor (HTGR or IRIS) licensing application and to provide support to the staff in the decision-making process of licensing advanced reactors. This work does not include review of any applicant's PRA.

In the past, the selection of licensing basis events was done on the basis of engineering judgement, and, where uncertainties associated with a proposed design were not well understood, the approach to licensing was to increase the safety margins and the defense-in-depth. However, experience has shown that this conservative approach to licensing can lead to unnecessary regulatory burden and may not have identified and addressed all of the potential safety concerns. Experience also has shown that PRA supplements the conservative approach and provides a tool to identify weaknesses in both design and operations, especially when used in an iterative manner.

During the development of the methods, expertise, and technical expertise, areas where there is insufficient information (e.g., due to insufficient operating experience) will be identified. These areas will be the subject of expert judgement or sensitivity studies to gain an understanding of the uncertainties associated with these areas.

PRA will be used in the licensing of advanced reactors, which is an application much more challenging than the use of PRA in risk-informed regulation of current LWRs. Applicants will provide technical arguments for the acceptability of their proposed advanced reactor design, on the basis of PRA results. While safety margins and defense-in-depth will be maintained to protect the health and safety of the public, PRA results and insights may be used to enhance the traditional approach and to reduce the conservatism traditionally provided. Therefore, developing the PRA tools, methods, and expertise is important for the review and licensing of these reactors. Having this capability will enable the staff to do comparisons with submitted analyses and results, thus, gaining an independent and more complete understanding of the safety issues associated with the proposed designs. These tools, methods, and expertise are also needed to direct other areas in this plan, (e.g., identification of the most probable accident scenarios for accident modeling and source term identification with MELCOR and consequence assessment with MACCS2).

VI.2.1.1.3 Objectives and Planned Activities

The objective of the advanced reactor PRA work is to develop:

- the necessary data for the PRA,
- an understanding of the uncertainties,
- the methods necessary to understand the PRA aspects of advanced reactor designs,
- the expertise to evaluate advanced reactor PRAs,
- an understanding of regulations needed as part of the licensing process, and
- identification of additional research needed.

The end product of this work will be the guidance for NRC reviewers, explaining how the results can be used by to independently review advanced reactor PRAs.

This plan is comprised of three tasks. The first task is to develop the methods, data, and tools needed for evaluating the different design and operational characteristics of advanced reactors from those of current reactors. The second task is to use the results of the first task to: (1) gain expertise, (2) provide guidance for assessments in other areas of this plan, and (3) evaluate advanced reactor designs. The third task is

documentation which not only documents what has been done, but will provide guidance for the review of applicant's advanced reactor PRAs.

VI.2.1.1.3.1 Task 1. PRA Development for Advanced Reactors
There are fundamental tasks that need to be performed to support either performing an independent PRA or reviewing a submitted PRA for advanced reactors. The information from the tasks described below, some of which would be developed in other areas of RES, is needed for this work.

VI.2.1.1.3.1.1 Initiating event identification and quantification. The events that challenge the current generation of LWRs may not be applicable to advanced reactors. It is necessary to correctly and comprehensively identify those events that have the potential to initiate an accident. Therefore, understanding what events can occur (as a result of design characteristics, equipment failures, and human errors) that challenge the plant operation comprise the first step in assessing the risk associated with a given reactor design. Extensive use of existing HTGR information and PRA will be used, as appropriate. This quantification will provide the necessary initial data on initiating event frequencies for use in the PRA.

VI.2.1.1.3.1.2 Accident progression and containment performance (including source term). The likely accident progression phenomena need to be determined based on ongoing research, previous experiments, experience in other industries, and expert judgment. Success criteria, accident progression, and source terms for advanced reactors may be different from those for LWRs. A combined deterministic/probabilistic approach, with elicitation methods, similar to those used for the liner melt through and direct containment heating issues in some LWRs, may be possible. The accident progression for different advanced reactor designs needs to be understood. For example, although hydrogen should not be generated in HTGRs during the course of an accident, the potential generation of other combustible gases needs to be examined. In addition, the loss of helium and the effects of air (and potentially water) ingress on the accident progression needs to be considered. Assessment of potential combustible gas generation, for example, will be performed as part of thermal hydraulics and severe accident work of this plan and will be feed into the PRA as part of the necessary data to evaluate advanced reactors.

A probabilistic containment analysis (Level 2 PRA) is needed to assess the benefits of having a reactor containment or confinement with a filtered venting system to provide adequate protection against release of fission products. (The confinement concept has been successfully modeled in past PRAs, although not for commercial reactor designs.) While the technical assessment of the performance of containment versus confinement will be performed as part of thermal hydraulics and severe accident work of this plan, those results are needed as input to the PRA model of advanced reactors. The benefit of complete underground siting, instead of the partial underground siting now proposed for some HTGR designs, needs to be evaluated. These analyses would be applicable to safeguard and security considerations for review of license applications.

Source term work will be performed as part of thermal hydraulics and severe accident work of this plan. The knowledge of fuel performance is a prerequisite to performing an

independent review of the PRA. We need to understand how the core behaves in accidents such as overheating or immersion in media other than helium (in air or, if possible, in water). This behavior should be understood not only for fresh fuel but also for end of life fuel to evaluate the impact, if any, of burn-up.

VI.2.1.1.3.1.3 System modeling. The probabilities and failure modes of passive systems (used extensively in advanced reactors) and the digital I&C systems in advanced reactor designs needs to be determined for incorporation into the PRA. Passive systems have been treated in PRAs as either initiators (e.g., LOCAs) or complete failures. As a result, current PRAs model only the performance of active systems using a binary logic which is suitable for such purposes. It is not clear that this approach would be suitable for modeling passive systems exhibiting slow evolutionary behavior during accidents. Therefore, the modeling approach needs to be reconsidered for potential modifications for advanced reactors.

Digital systems typically have not been considered in the past PRAs. In advanced reactors, however, I&C systems will normally be digital. The reliability of digital systems is being addressed in another part of this plan. PRA modeling needs to address the issues concerning digital system performance. Digital I&C may have failure modes that have not been considered previously. Methods should be developed for incorporating digital system failure in the PRA logic.

The uncertainties associated with the development of modeling the failures of passive and digital systems will be addressed to the extent practical.

VI.2.1.1.3.1.4 Data collection and analyses. Advanced reactors introduce different systems and components and hence, LWR data may not be applicable. The use of appropriate data is crucial in the assessment of the risk associated with a given reactor type. Therefore, collecting and analyzing data applicable to advanced reactors is essential. The existing NRC-reviewed HTGR PRA will be used, as applicable.

This task includes addressing the data uncertainties. Understanding the uncertainties is a very important aspect for any PRA; it is much more crucial for these types of reactors given the limited or lack of operating experience and the expected significant use of the PRA in the licensing process.

VI.2.1.1.3.1.5 Human reliability analysis. The operators' role in the new reactors is not well understood. The advanced reactors are proposed to be built on the premise that they will be human-error free and that, if an event occurs, human intervention will not be necessary for an extended period of time. Issues related to the need for reliable operator performance (e.g., staffing and training) are part of a different activity of this plan. Human reliability methods, such as ATHEANA, were developed to assess the impact of human performance on plant safety. When dealing with long-term and slowly evolving accidents, such as those expected to be dominant in graphite-moderated reactor accident sequences, revision to those probabilities may be needed. This task will determine if (and what) modifications are warranted to appropriately incorporate the impact of human performance in advanced reactors. Operator performance may be

affected by having multiple modules that share the same control room. The likelihood of errors of commission or omission need to be understood under these conditions.

VI.2.1.1.3.1.6 Other events (internal flood, fire and seismic). As with any design that uses digital I&C, failure possibilities of electronics need to be addressed. Specifically, the response of digital electronics in a fire or flood is expected to be quite different than that of electromechanical components. The differences may not be just in probability but also in the kinds of failures that could potentially occur. Furthermore, current plants have shown that the core damage frequency from external events may be similar to that from internal events. Therefore, external events needs to be considered for advanced reactors from a scoping perspective to identify, if possible, unique vulnerabilities.

VI.2.1.1.3.1.7 Quantification. The SAPHIRE code would be used in the performance of an independent PRA. The code needs modifications for a full scope PRA (external and internal events, full and low power). A full scope PRA will generate many more "cut sets" than can reasonably be handled now. In addition, the rationale developed for other designs for pruning the results may not be appropriate for advanced reactor designs. Source terms and consequences (Level 3) which will be performed as part of severe accident and consequence work of this plan, needs to be incorporated into the PRA tool. SAPHIRE needs modifications to integrate Level 1 (CDF) analyses with Level 2 and Level 3 analyses, and dynamic modeling.

VI.2.1.1.3.1.8 Uncertainties. Identification of uncertainties will help the decision-making process for either reducing the uncertainties by more research or strengthening the regulatory requirements and oversight, e.g., defense-in-depth and safety margins. A PRA provides a structured approach for identifying the uncertainties associated with modeling and estimating risk. There are three types of uncertainty: modeling, data, and completeness. Data uncertainty is addressed in Section 1.4. Processes will be developed to identify and understand the significance of the modeling and completeness uncertainties.

VI.2.1.1.3.1.9 Unique operating characteristics of advanced reactors, operating in other than full power mode, needs to be examined in order to be correctly accounted for in the PRA.

VI.2.1.1.3.1.10 Multiple modules. Current PRAs are usually performed for a single unit or sometimes for two sister units. Advanced reactors (e.g., PBMR) will operate up to 10 modular units together at a site with a centralized control room. The PRA tool needs to address potential interactions among the multiple units. The potential effects of smaller operator staffs in a common control room under potential common cause initiators (such as seismic events) needs to be considered.

VI.2.1.1.3.1.11 Risk metrics. The concepts of core damage frequency (CDF) and large early release frequency (LERF) may not be the best figure of merit for some advanced reactor designs (e.g., PBMR). However, Level 3 PRA results (offsite consequences which will be performed as part of severe accident and consequence work of this plan) need to be considered for advanced reactors and incorporated into the

PRA, including SAPHIRE. Therefore, the subsidiary figures of merit for advanced reactors, consistent with NRC top level safety goals, needs to be verified or identify more appropriate figures of merit. The appropriate figure(s) of merit are needed for risk evaluations as well as for developing regulatory criteria and guidance documents for design review and acceptance.

VI.2.1.1.3.1.12 Safeguards and security. As mentioned above in Section 1.2, there are some portions of this work where explicit information can be generated regarding the safeguards and security for the design. We need to explore how this can be accomplished in the most efficient manner and what other areas of the PRA studies can assist in this endeavor.

VI.2.1.1.3.2 Task 2. Use of PRA

The results developed in Task 1 will be used to: (1) gain expertise, (2) provide guidance for assessments in other areas of this plan, and (3) develop an independent capability to evaluate advanced reactor PRAs. The level of detail in these analyses will be determined by the PRA information needed for supporting the licensing process. The results will provide a basis for performing comparisons with advanced reactor PRAs submitted as part of their license application.

VI.2.1.1.3.3 Task 3. Documentation

The documentation will not only document what has been done, but will provide review guidance for advanced reactor PRAs. A wealth of information will be generated by performing Tasks 1 and 2. The PRA and review guidance is needed to be capable of determining the probabilistic implications of different design configurations and operation conditions. This is needed to develop probabilistic perspectives to support NRC risk-informed decision-making throughout an advanced reactor licensing process. However, using this information appropriately is not an easy task. Users should be able to understand both the results of the PRA work as well as the underlying hypotheses driving the results. Therefore, guidance will be developed to:

- review applicants' PRAs for advanced reactors,
- help identify research needs, and
- develop regulatory guides and SRP sections.

VI.2.1.1.4 Application of Research Results

The application of this work will be to:

- provide staff guidance explaining how the results of this work can be used to independently review an advanced reactor PRA,
- interface and interact with the work performed in other areas of the RES plan to feed its results back to help identify where there is inadequate information, and, thus, support staff decision-making for research, and
- provide input to potential modification to the regulations and the development of regulatory guides and SRP sections.

VI.2.1.2 Instrumentation and Controls

VI.2.1.2.1 Background

The new generation of advance reactor concepts, both for HTGRs and ALWRs will be the first opportunity for vendors to build new reactor control rooms in this country. The advances that have been made in the development of many of the current generation of operating reactors in other parts of the world will be used in the design and construction of new plants. These new plants are expected to have fully integrated digital control rooms, at least as modern as the N4 reactors in France or the advanced boiling-water reactors in Japan. In addition the desire for much smaller control room staffs will also push the designs of the plants in the direction of a much higher degree of automation. The use of multiple modular plants may also require more complex control of both the primary I&C systems and for all of the support systems including the switch yard.

I&C systems play an important role both in reactor control and in providing information on the balance of the plant. Research of the advanced (digital) I&C is needed in these areas to ensure that the NRC is capable of reviewing these new designs.

The NRC Research Plan for Digital Instrumentation and Control (SECY-01-155) outlines current and future research into several areas of emerging I&C technology and applications that will be used in the HTGRs and ALWRs. These include smart transmitters, wireless communications, advanced predictive maintenance, on-line monitoring methods, and enhanced cyber security issues. The NRC has recently started new research programs in the areas of wireless communications and on-line monitoring. This research will support the development of review guidance for NRR for these new and improve technologies that will be applicable to both current reactor retro fits and advanced reactors. In addition to this research, the programs described in this section are needed to the develop of the knowledge and tools needed to support the review of these new reactor technologies.

The national and international research community has been involved with research and development of advance control and monitoring systems for nuclear power plants for many years. The international community, particularly in Europe, Japan, and Korea, have developed integrated advanced control rooms and performed more research in the areas of automation of plant operations and advance plant monitoring and diagnosis than in the US. Therefore, there will be significant opportunities for international cooperation in this area.

General Atomics is doing detailed control systems design studies using plant simulators to help optimize control system designs. PBMR Corporation is also looking into advanced control systems. This research and development is being performed by both the vendors and through joint efforts with other organizations, such as the universities and U.S. national laboratories, including Oak Ridge National Laboratory (ORNL) and Idaho National Engineering and Environmental Laboratory (INEEL). There may be an opportunity to collaborate on some of these research programs, particularly in the areas of advanced control algorithms and control of multiple plant modules.

One of the major areas of research outlined in the Department of Energy (DOE) Long-term Nuclear Technology Research and Development Plan is the I&C area. Several of the research topics proposed in this plan are of particular interest to HTGRs, such as robust communications and wireless sensors, smart instrumentation, and condition monitoring. Also of interest is research into distributed computing, condition monitoring, advanced control algorithms, and on-line monitoring. As part of the implementation of this long-term research plan, DOE has developed six Nuclear Energy Research Initiative programs in this area. These include research in the areas of automatic generation of control architectures, self diagnostic monitoring systems, smart sensors, and advance instrumentation to support HTGRs.

VI.2.1.2.2 Purpose

The advanced reactor plants will be designed for autonomous operation with a minimum of supervision by plant operators for long periods of time. This may include automated startups, shutdown, and changes of operating modes. There will be fewer operators compared with current generation nuclear power plants, as few as three operators for ten modules. This will require that not only normal operations but off normal operations and recovery be more highly automated. To make modular reactor concepts effective the plant should function like a single larger plant. This will require a level of automation and coordination that is heretofore unheard of in the nuclear power industry. The NRC needs to enhance both our basic understanding of how plant control and safety systems will be designed cope with partial failures of interconnected systems, particularly at the switch yard and the control room, and our review guidance and tools.

Because of the longer fuel cycles and much longer time between maintenance outages, the plants will require more extensive use of on-line monitoring, diagnostics, and predictive maintenance. Instrumentation will be needed to support this increased automated surveillance. How these systems will integrate with the control systems needs be understood. Because some of the systems in this new generation of ALWRs and HTGRs will be operating in new temperature ranges, it is expected that several new kinds of sensors will be developed. The limitations of these new sensors will need to be investigated. There may be temperature, pressure, flow, and neutron detectors used that will require changes in the methods for performing design and safety calculations (drift, calibration, response time, etc). Current regulatory guidance and tools will need to be reviewed and enhanced to support the review of these systems.

The application of highly automated control rooms in other industries has used modern control theory controllers to increase plant availability and decrease workload on operators. It is likely that these new HTGRs will use some of these advance modern control methods. These could include simple feed forward controllers, non-linear controllers, neural-fuzzy controllers or even more exotic methods. How these control algorithms will affect the operational modes of the plants need to investigated. Additionally, review guidance and tools will need to be developed to analysis these methods.

To adequately understand the more complicated digital I&C systems within a risk-informed licensing framework, additional risk modeling will be needed. This will also be

needed to support the research of the operator and control interface. Because of the lack of adequate models and data to support risk analysis, the uncertainties in this area are relatively high and can only be reduced by significant new research in this area.

VI.2.1.2.3 Objectives and Planned Activities

To develop the regulatory infrastructure (review methods and tools) to support the review of applications in this area, the NRC will need to conduct research into the following areas :

VI.2.1.2.3.1 Review of Current Practices and Lessons Learned from ABWR and N4 Control System Development and Regulatory Review

This is an effort that has to be performed for each type of reactor design for which sufficient information is available. The review of both operational experience and design lessons learned will be the first priority. Additionally the review will focus on the regulatory analysis methods and tools that have been used by foreign regulators. The effort will also have to be continued over time as new information becomes available.

VI.2.1.2.3.2 New risk models for I&C systems in advanced reactors

This effort will complement the work that is at the University of Maryland and the University on Virginia but will focus of the development risk models for advanced reactor I&C systems (for review of the possible safety issues of the systems and for integration into advance reactor risk models.

VI.2.1.2.3.3 Analysis of the Requirements and Potential Issues Involved with HTGR Reactor Instruments

This effort will include review of the requirements for and the development of new instruments to support design, construction and operation of the HTGR. This work will include developing a better understanding of how the requirements were developed and what review methods are the most appropriate. These will include new neutron detectors, particularly for PBMR, temperature sensors, etc. This effort will also support the review of needed prototype plant instruments.

VI.2.1.2.3.4 Development of models of autonomous control

This effort will include the development of information and models to review and examine advanced autonomous control methods that will be used in advanced reactors. The effort will review both current methods used in other areas, such as natural gas power plants and methods that have been proposed by the vendors. The product will be revisions to current review guidance or new tools.

VI.2.1.2.3.5 Analysis of control systems used to integrate the control of multiple module plants

The amount and the way in which systems will be integrated in advanced reactors using multiple modules will be investigated. At what points control and safety systems are integrated and the amount of automated actions will also be invistigated. New review guidance will be developed.

VI.2.1.2.3.6 Analysis of on-line monitoring systems and methods and advanced diagnostic methods needed to support HTGR

This effort will include the review both current methods and investigate the required development of instruments and techniques to support the current availability and maintenance schedules. How the limits and new capabilities of these systems will affect other issues, such as in service inspection intervals will also be evaluated.

VI.2.1.2.3.7 Review of advanced control algorithms for application to advanced reactors

The effort will develop information on the current methods likely to be used in advanced reactors and investigate the potential issues with these algorithms when used in a reactor setting.

VI.2.1.2.3.8 Analysis of the requirements and potential issues involved with advanced light water cooled reactor instruments

This effort will review the requirements for and the development of new instruments to support design, construction and operation of advanced light water cooled reactors. These will include new neutron detectors needed to support ultra long life cores. How review guidance will need to be modified to support these instruments will be investigated.

VI.2.1.2.3.9 Analysis of on-line monitoring systems and methods and advanced diagnostic methods needed to support ALWRs

This effort will review both current methods and investigate the required development of instruments and techniques to support this the current availability and maintenance schedules. How the limits and new capabilities of these systems will affect other issues, such as in service inspection intervals will also be evaluated.

VI.2.1.2.4 Application of Research Results

The results from the first effort will provide insights which will help identify those I&C systems and technologies that have been used in other reactors such as the ABWR and N4, and any issues that may be related the to operation of these systems. The remaining efforts will provide both independent tools and methods to assist in assessing new technology that will be an integral part of these reactors. The existing tools are not sufficient to complete these assessments. These programs will provide information for revisions to Chapter 7 of the Standard Review Plan and in the supporting Regulatory Guides.

VI.2.1.3 Human Factors Considerations

VI.2.1.3.1 Background

Nuclear power plant (NPP) personnel play a vital role in the productive, efficient, and safe generation of electric power, whether for conventional LWRs or for advanced reactors. Operators monitor and control plant systems and components to ensure their proper functioning. Test and maintenance personnel help ensure that plant equipment is functioning properly and restore components when malfunctions occur.

It is widely recognized that human actions that depart from or fail to achieve what should be done, can be important contributors to the risk associated with the operation of nuclear power plants. Studies of operating experience demonstrate that human performance contributes to a large percentage of events and has a significant impact on the risk from nuclear power generation. Studies of PRA results found human error to be a significant contributor to core damage frequency (CDF), that by improving human performance, licensees can substantially reduce their overall CDF, that a significant human contribution to risk is in failure to respond appropriately to accidents, and that human performance is important to the mitigation and recovery from failures.

VI.2.1.3.2 Purpose

Advanced reactors are expected to present a concept of operations and maintenance to the staff that is different from what is currently the case at conventional reactors. Operators will be expected to control multiple modules at one time and those modules may be in different operating states. Operators will be required to monitor online refueling, while other modules are in a normal operating state, while another could be facing a transient. The control rooms will be fully digitized using glass cockpit concepts. Procedures will be computerized and control actions may be taken directly from the procedure display or automated, with the operator only in the position to bypass the automation. Different training and qualification may be required of the plant staff to maintain digital systems and to focus decision making on monitoring and bypassing automatics rather than the active control that operators now take. Higher level knowledge may be needed to respond to situations where automatic systems fail. Any of these changes can pose new and challenging situations for operators and maintainers. RES needs to provide regulatory staff with tools, developed from the best available technical bases, necessary to accomplish their licensing and monitoring tasks. This will ensure that advanced reactor personnel have the tools, knowledge, information, capability, work processes and working environment (physical and organizational) to safely and efficiently perform their tasks. The ultimate goal is to minimize the human error contribution to the risk associated with the design, construction, operation, testing and maintenance of these facilities.

In accordance with 10 CFR 50 and 52, the staff of the NRC reviews the human factors engineering (HFE) programs of applicants for construction permits, operating licenses, standard design certifications, combined operating licenses, and for licence amendments. The purpose of these reviews is to help ensure safety by verifying that accepted HFE practices and guidelines are incorporated into the applicant's HFE

program. The review methodology (NUREG-0711, "Human factors Engineering Program Review Model" and SRP Chapters 13 and 18) are the basis for performing reviews. The reviews address 12 elements of an HFE program: HFE Program Management; Operating Experience Review; Functional Requirements Analysis and Allocation; Task Analysis; Staffing; Human Reliability Analysis; Human-System Interface Design; Procedure Development; Training Program Development; Human Factors Verification and Validation; Design Implementation; and Human Performance Monitoring.

Current regulations and guidance (for example: 10 CFR 26, 10 CFR 50, 10 CFR 52, and 10 CFR 55, Regulatory Guides 1.8, 1.134, 1.149, NUREG-0700, NUREG-0899, NUREG-1220) that address human performance issues were developed for review of LWRs and ALWRs. Though many of these may be applicable to new concept advanced reactors with little or no adaptation, as newer reactor and control technology is developed and introduced into advanced reactors, new regulations and guidance may need to be developed to address the new concept of operations. A sound technical basis needs to be developed as part of the guidance development process. The HFE aspects of advanced reactors should be developed, designed, and evaluated on the basis of a structured systems analysis using accepted HFE principles at the same time as other systems are being designed. The role of the human needs to be considered as a part of the system from the initial concept development stage so that the role is appropriate to the function eventually assigned, as specified in IEEE 1023.

To assure that human factors activities are risk-informed, there needs to be a close synergism with the HRA aspects of this plan. To perform in-depth PRA/HRA analyses for advanced reactors, new sources of data and information will be needed. Human factors research can help to develop the data base necessary to adapt the HRA techniques to advanced reactors. HRA in turn can help prioritize the human factors efforts. However, until HRA models can be accurately developed for these new designs to define and prioritize human factors issues, conventional human factors methods may need to be applied.

Currently there is no facility in the United States for performing human factors research for advanced reactors. Such a facility could be used to independently confirm applicant proposals in the areas of human factors and digital I&C. It could also be used to develop data for HRA. The French have reactor simulators that they operate or are in development for the N-4 reactor and for concepts they are considering. There are also research simulators in Japan and Korea. The OECD Halden Reactor Project operates three reconfigurable research simulators (PWR, BWR and VVER) at their facility in Norway. These simulators can all be controlled through a common advanced design control room. They do not have a simulation of any of the advanced plants (e.g., PBMR), but they have the capability to develop a simulator when sufficient system and thermodynamic information is available. Virtual Reality techniques that can simulate virtual control stations can conceivably be used to perform this type of confirmatory research.

VI.2.1.3.3 Objectives and Planned Activities

VI.2.1.3.3.1 Develop Insights Report on the Impact of Human Performance on Advanced Reactors

Currently little is known about the planned role of humans in the operation and maintenance of advanced reactors, because the concept of operations has not yet been fully developed by vendors or potential applicants. What little is known would lead one to believe that there may be a change in the human's role from LWRs. Therefore, to develop a detailed human factors plan the NRC should first determine from best available information what human performance issues need to be addressed, what research facilities might be needed, what regulatory guidance may be needed, and what confirmatory research the NRC should be prepared to perform. As issues are identified they can be integrated into the overall plan. The elements of the plan that follow are those that are common to human factors programs found throughout the government and the human factors profession. This initial effort will be accomplished by:

VI.2.1.3.3.1.1 Examining concept of operations and the role of automation. Prototype advanced reactors have been operated in the past. A review of operating experience at these prototypes would be the starting point for this effort. Further, there are many advanced automated systems in transportation, aerospace, and the petrochemical industries that may have operational similarities to advance reactors. Research and experience related to such systems would also be a source of information since advanced reactor control rooms are anticipated to be highly automated. The nature and level of automation are important aspects for the operator because it affects their situation awareness and workload. Operators will be facing a new concept of operations. Many questions need to be answered to have a good understanding of the role of the human in advanced reactors. Will the design be based on the concept of human centered automation? Will they deal with the automation and potential failure of automation? How will they be expected to operate multiple modular modules? What will their role be in maintenance and on-line refueling? What other roles might the operator have? What role will the operator have in configuration management? What limits will be placed on their activities during periods of work underload? What information will the operators need and how should it be presented? Should procedures be automated or should intervention be required? What will be the consequences of bypassing or overriding automated systems? Who will make operational decisions during emergencies and what should their qualifications be? What is the role of plant staff other than operators? This review would result in the identification of human performance issues for the various reactor types that require the development of new review tools and guidance to assist the regulatory staff in reviewing applicant submittals and to develop a knowledge base for performing those reviews. The tools and guidance should have a sound technical basis derived from research and/or information that can be adapted to NRC guidance without the need for further research.

VI.2.1.3.3.1.2. Reviewing existing requirements. Once the concept of operations is better understood, the next aspect of the review would be to systematically review the existing licensing criteria to determine their applicability to proposed advanced reactors. This would include rules, regulatory guides, NUREGs, the standard review plan (SRP) and consensus standards from IEEE, ANS/ANSI, or from industry organizations

(e.g., NEI, EPRI) for topics such as staffing, procedures, training, human-system interface, fitness-for-duty. As part of this effort, it would be necessary to understand the proposed concept of operations, control station concepts, control room environment, expected working conditions, activities in the balance of plant, etc.

VI.2.1.3.3.1.3 Review existing human performance research facilities. It is important to understand the operator's role in the operation of advanced reactors, since it is anticipated that it will be significantly different from that role for conventional reactors. Since each of the existing conventional reactors is unique, each plant has a plant specific-simulator. As advanced reactors are developed it is anticipated that they will be more standardized and thus generic simulators will be more practical. Such simulators would be the means for conducting procedure and design V&V called for by Chapter 18 of NUREG-0800, and possibly for operator licensing examinations required by 10 CFR 55. To meet these requirements, it would be to the advantage of the industry to develop such simulators. In addition these generic simulators (especially, if reconfigurable) could also be used as a test bed for human factors, digital I&C and HRA studies. Since there are currently no existing human performance research simulation facilities in the nuclear power sector in the US, and the facilities that do exist in Europe are not for advanced reactors. The NRC may want to consider sharing in the development of such a simulation facility to perform confirmatory studies of applicant submittals relative to issues such as staffing, control station design, procedures, etc.

A feasibility study to determine the availability of facilities that could be used to perform confirmatory human performance studies will need to be performed. This would include review of the facilities in Europe and Asia to determine their applicability or adaptability to advance reactor issues, as well as facilities that are currently used for other applications that are based on advanced systems (e.g., transportation, aerospace, chemical processes, maritime). Alternatively, the feasibility of establishing such a research facility, perhaps in cooperation with the industry, will be explored. The use of the facility to support I&C research or for use in collecting data for HRA quantification will also be considered. Depending on the outcome of the feasibility study, additional resources may be needed to acquire simulator time or to develop a facility.

VI.2.1.3.3.2 Function and Task Analysis

Since the HFE Program Review Model described in the Purpose section is dependent on function and task analysis, tools and techniques to perform and review such analyses during the design stage are important to the rest of the elements of the model. Such analytical approaches for evaluating HFE requirements for complex systems have been evolving over the past few decades. Human behavioral modeling techniques, such as task network modeling and discrete event simulation, have been developed and tested by the United States Army and Navy for a decade and some of these techniques have been accredited by the U.S. Department of Defense for use in HFE analyses during system design and engineering. These human behavioral modeling techniques and tools need to be developed or adapted for use by the regulatory staff to for use in the licensing of advanced reactors. The use of such analytical models could enhance the efficiency and effectiveness of the licensing reviews and would be used in a manner similar to thermal-hydraulic, fuel and accident analysis codes and models. Data from

human performance studies would be used to populate and maintain the code and would be used to assess applicant submittals.

VI.2.1.3.3.3 Staffing

Exelon has already indicated that they plan to ask for a waiver from 10 CFR 50.54(m), the staffing rule for LWRs, to allow for fewer licensed operators at the PBMR. Central to the safety of any manned-system is the balance between the demands of the work and the available time of the staff. Not only does the humans' workload capacity have to be sufficient to fulfill their requirements during periods of normal operation, human capacity should also be sufficient to handle the periods of high task demands associated with other-than-normal operations. In fact, it is during these periods of off-normal activity that sufficient human capacity to understand the situation, make the appropriate diagnosis, and select the correct action is most critical. It is expected that operators will have longer to respond to unusual situations at advanced reactors than at LWRs; however it will still be necessary to determine the number and qualifications of individuals needed to safely operate and maintain these new reactors. An analytical or modeling approach as described in item 2. above could be used to develop and review staffing using a performance based approach, rather than developing prescriptive requirements. Such an approach would be consistent with the finding in NUREG/IA-0137. This could result in a change to 10 CFR 50.54.

VI.2.1.3.3.4 Training and Qualifications

Training for LWRs is controlled under 10 CFR 50.120 and accredited by the National Academy of Nuclear Training to be consistent with the Systems Approach to Training. NUREG-1220 is used by staff and inspection modules are used by staff in the event a for-cause training review is needed. The current training review methods should be evaluated and updated as necessary to account for possible changes, (e.g., use of cognitive task analyses, in addition to traditional task analyses, for development of learning objectives). Further, innovative training concepts, such as embedded training and the use of virtual reality may be proposed, so the NRC would need tools to evaluate such possible enhancements to training. Qualifications are generally based not only on training but also education and experience. Questions that need to be considered include: From where will the operators and other staff familiar with advanced systems and digital interfaces come? Will past power plant or Navy experience be effective? How will operator licensing need to be changed? What will the requirements be for simulation? Can training and simulation be embedded into the operational setting? This could result in the need to revise 10 CFR 55, 10 CFR 50.120, Regulatory Guide 1.8, Regulatory Guide 1.149, and NUREG-1220.

VI.2.1.3.3.5 Procedures

Currently the NRC has human factors review guidance only for paper-based emergency operating procedures and the operating plants use only paper-based procedures. Limited guidance for the review of computerized procedures has been developed but needs to be assessed against advanced reactor systems, since advanced reactors will have computer based or glass cockpit control rooms the procedures are likely to be computerized. Guidance for the review of these system will have to be developed to modify NUREG-0899 and SRP Chapter 13.

VI.2.1.3.3.6 Human-system Interface (HSI)

The recent revision to NUREG-0700 is expected to be applicable to much of the human-system interface, however there are certain issues not covered in NUREG-0700 for which guidance may need to be developed. These issues were not included in NUREG-0700, Rev. 2, because there were no validated criteria available and there was not sufficient technical basis on which to develop the criteria. These issues include: guidance for high-level displays that are based on processed information, with different types of processing, (e.g., functional decomposition; new display types such as flat panels and large screens). This work could result in changes to or new review guidance.

VI.2.1.3.4 Application of Research Results

The result of the first effort listed will be an Insights Report which will identify human performance issues that may be related to the operation and maintenance of advanced reactors. The report will be used to identify human performance issues that require further research or information that can be adapted to NRC guidance without the need for further research. The need for any changes to regulations, regulatory guidance or review guidance will be identified.

The effort on function and task analysis will focus on the development of guidance or an analytical tool or model to assess the quality of the function and task analysis performed by applicants. Such guidance is needed since the function and task analysis is basic to staffing, training, HSI, procedures, and work practices. The use of an analytical tool or computer-based model would enhance regulatory efficiency.

The efforts on staffing, training and qualifications, procedures, and human-systems interaction will result in possible changes to the regulations, regulatory guidance, or review guidance and methods for each issue as identified above. In many cases, a detailed technical basis would be developed before developing the regulatory tool.

The results of any field or simulator research could also be used to support HRA quantification, through the identification and quantification of performance shaping factors (PSFs) or error forcing contexts.

VI.2.2 Reactor Systems Analysis

As stated previously, the primary goal of the advanced reactor research program is to establish an appropriate database and develop the analysis tools to help the staff make sound decisions on key technical and regulatory issues concerning the safety of advanced reactors. To address these infrastructure needs for staff capabilities in reactor and plant analysis, RES will develop data, tools and methods that will allow the staff to independently assess advanced reactor safety margin, and to evaluate reactor safety analyses submitted by applicants in support of future advanced reactor license applications. This research effort is also designed to provide analytical support for the development of a regulatory framework for advanced reactor licensing and establish the technical basis for related policy decisions.

This section will discuss research activities needed in the area of reactor systems analysis, which includes thermal hydraulic analysis, nuclear analysis, and severe accident and source term analysis. For the thermal hydraulic analysis of helium-cooled, graphite-moderated reactor systems (HTGRs), the discussion will describe a planned approach to providing the data and modeling tools needed for predicting HTGR-specific heat transfer and fluid flow phenomena, including "multi-phase (helium with air and/or water ingress)" fluid flow with convective, conductive and radiative heat transfer in irregular and complex geometries. For analyzing reactor designs cooled and moderated by water, the need to investigate two-phase flows under new ranges of conditions will be reviewed. Research in the area of nuclear analysis will start with the development of modern, general-purpose nuclear data libraries that will support all nuclear analysis activities throughout the arenas of reactor safety, materials safety, waste safety, and safeguards. Nuclear analysis research for reactor systems analysis will include the development and testing of (a) reactor physics codes and methods for modeling reactor control and feedback and for predicting the in-reactor heat sources from fission chain reactions and fission-product decay, and (b) neutron transport and shielding models as needed in analyzing reactor material activation and damage fluence. In the area of severe accident and source term analysis, the discussion will address the data and analysis tools needed for (a) evaluating the progression of credible severe accident scenarios involving core damage phenomena such as fuel melting or high-temperature chemical attack and (b) modeling any resulting releases and transport of radioactive fission products within and outside the reactor system boundaries.

The research outlined in this section will produce specific information that will be incorporated into a suite of reactor system analysis tools (i.e., computer codes and methods) and thereby give NRC staff the necessary independent capabilities to reliably predict system responses. The development of a suite of reactor system analysis tools and the data to support and validate them will permit the NRC staff to (i) conduct confirmatory analyses in the review of applicants' reactor safety analyses, (ii) support development of the regulatory framework by assisting, for example, in the identification of safety-significant design basis and licensing basis events, and (iii) conduct exploratory analyses to better understand the technical issues, uncertainties and safety margins associated with these new designs. The reactor systems analysis research discussed in this section of the plan will also provide needed information to many other parts of the research program. This will include providing fluences and temperatures, pressures and mechanical loads for use in work described in the sections on Materials Analysis and Fuel Analysis as well as information on damage sequences for PRAs.

VI.2.2.1 Thermal Hydraulic Analysis

VI.2.2.1.1 Background

Power reactors are licensed by showing compliance with specified safety limits. Some limits are easily identified and predicted while others require complicated modeling for proper evaluation. When modeling is required, applicants apply what are typically complicated mathematical representations of the system. Many of these "models" are typically combined into a computer code that represents the significant phenomena in the system under consideration. Due to their complexity, these "codes" need detailed assessment to demonstrate that they are appropriate for the proposed application. Additionally, thermal hydraulic analysis is used to determine the best estimate of system states to support analysis of the probability and mechanism for systems failures.

VI.2.2.1.2 Purpose

Thermal-hydraulic analyses are typically used to assess what safety limits are needed and whether limit and margins such as fuel design limits are met, and to predict transient effects on system components and materials, and to develop information for PRA. Understanding the effects of these features on local and system-wide thermal-hydraulics is necessary in order to confirm and quantify the expected safety margin of the proposed plants as well as to audit the applicant calculations.

VI.2.2.1.2.1 High-Temperature Gas-Cooled Reactors

In order to independently review an applicant's HTGR safety analysis, the NRC may need an independent thermal-hydraulic assessment capability. The staff has completed a preliminary review of the analysis capabilities needed to model fluid flow and heat transfer in HTGRs. The findings of this review can be summarized as follows. Given the nature of HTGR transients, a code will need to reliably and efficiently predict transients that evolve over time scales of days, not hours as we have become accustomed to in LWR analyses. Furthermore, some design basis transients are driven by radiative and conductive heat transfer through porous and solid structures, not convection, and this capability, although it currently exists in all codes, will have to be extended to three-dimensions and a spherical fuel element model will have to be added for analyzing PBMR transients. The NRC analysis tools should be able to model all of the turbo-machinery and passive decay heat removal systems, which implies that we need to accurately model gases (helium and air) in natural circulation. These systems are important for long term heat removal and recovery as well as determining initial steady state operating parameters and conditions. Turbo-machinery will likely be simulated using existing pump models, but this capability will have to assessed and modified as needed. For pebble bed designs, the staff needs the capability to model flow and heat transfer in a packed bed configuration. The code will need to model two different working fluids at once to model component cooling water systems. Finally, the capability to model graphite as a solid structure will have to be added.

Two types of codes will be used to fulfill this need for HTGRs. These are the traditional reactor systems analysis codes, such as TRAC-M, and general-purpose computational fluid dynamics codes, such as FLUENT. The reactor system analysis code for HTGR

applications will be built upon our existing TRAC-M code. Also, as discussed elsewhere in this plan (see section on Severe Accident Codes and Source Term Analysis), the MELCOR code will be used in conjuction with TRAC-M and/or FLUENT for analyzing events that cause core damage (e.g., air ingress with significant graphite oxidation).

Where appropriate, the development of new capabilities in TRAC-M will utilize or build upon corresponding features in the two earlier HTGR accident analysis codes, GRSAC and THATCH. The forerunners of GRSAC, called ORECA and MORECA, were developed in the 1975 to 1993 time frame at Oak Ridge National Laboratory (ORNL), largely under NRC sponsorship, to support the staff's licensing safety evaluation for Fort Saint Vrain and the pre-application review for the DOE MHTGR. After 1994, MORECA became GRSAC and, through non-NRC funding sources (mainly the Defense Nuclear Agency), was further developed to model past accidents and postulated events in various non-HTGR gas-cooled reactors, such as Windscale, Magnox, and AGRs. ORNL is now adding pebble-bed and Brayton cycle code models to GRSAC for their near-term use in support of an NRC interagency agreement with DOE on assessment of generic HTGR safety analysis code requirements and the staff's pre-application review activities for the PBMR. The THATCH code was developed at Brookhaven National Laboratory, likewise through NRC sponsorship in the 1975 to 1993 time frame, and was likewise used to support the staff's review activities for Fort Saint Vrain and the MHTGR. Unlike GRSAC, however, the THATCH code was not maintained after the NRC's MHTGR review activities were terminated in 1994, although THATCH code documentation is still available.

Over the longer term, incorporating the necessary HTGR code features from GRSAC into TRAC-M will be the best use of agency resources, as TRAC-M already possesses many of the features discussed above, the staff owns and controls the TRAC-M source code, and, given the code's modular structure, new capabilities can be added with relative ease. For example, TRAC-M already can model helium as a working fluid and the necessary material properties for helium are already in the code. These models will simply have to be assessed for accuracy. Where specific capabilities are not currently in TRAC-M (for example, modeling helium turbines), adding this capability can be readily achieved by changing one or more of the TRAC-M functional modules. SNAP (the graphical uses interface for TRAC-M) will also need to be updated to allow analysts to model HTGR designs.

FLUENT will be used because it gives us the ability to more reliably predict parts of the fluid system when we need to assess the capability of our reactor system code against some assumed known reference standard or when we need to assess a particular phenomenon in more detail.

One area that needs special consideration is test data. Data is needed to evaluate the accuracy of codes and assess margins of safety. Test data can be obtained from facilities ranging in size and complexity from small scaled component tests to scaled representations of the entire system. HTGR research has been conducted for a number of years at such facilities as the AVR, THTR in Germany and the HTTR in Japan. These and other programs have develop significant thermal hydraulic and other data. However, additional data is needed to investigate issues including pebble bed hot spots inferred from the melt wire test results at AVR, incomplete mixing of reactor outlet

helium and thermal stratification, natural circulation under loss of forced circulation accidents, air and moisture ingress accidents, and reactor cavity cooling. As part of this research, the staff will initiate cooperative efforts with the international community to identify data needs and develop experimental facilities to provide data where little or no data exist. The staff will also evaluate data available from previous U.S. efforts related to HTGRs and assess their applicability to current designs.

Several issues will need to be addressed by the proposed research:

- Confirm and modify as needed the capability to model flow and heat transfer in packed beds. The solver in TRAC-M is based on a porous medium assumption which should be directly applicable to packed bed analyses if given appropriate inputs. Appropriate constituitive relationships will have to added. Three-dimensional conduction and a spherical conduction model will have to be added. An improved radiation model is also needed. These capabilities will have to be assessed.

- Confirm and modify as needed the capability to model HTGR turbo-machinery. At a minimum we will need to change the turbine model to remove some restrictions related to LWR applications. Appropriate data will also be needed for input preparation.

- Confirm and modify as needed the capability to model natural circulation of gases.

- Add the capability to simultaneously model two different working fluids, to support, helium, water and air in the reactor as a result of air and moisture ingression accidents. Along with this, the ability to track multiple non-condensible gas sources will need to be added.

- Assess speed of the code and improve as necessary to allow for efficient simulation of transients on the order of days. This may require extensive modification of the code to support the much longer analysis times, however, before this is undertaken, other means will be looked at to partitioning the analysis into time periods where similar phenomena will be taking place in an effort to maximize the computational efficiency.

- Add graphite as a structural material including graphite oxidation.

- Update the GUI to work with HTGR designs.

- Using a PIRT process and the information developed as part of previous HTGR programs and the IAEA review of data to develop data needs for code development and assessment.

- Based on the conclusions of the above, initiate efforts to develop necessary data. Every effort will be made to develop data collaboratively with the international community.

- Perform an assessment of the code using the PIRT and the available data. This effort might identify a need to modify the code in areas not mentioned above.

VI.2.2.1.2.2 Advanced Light Water-Cooled Reactors
The thermal-hydraulics of advanced light water reactors (ALWR) is relatively well understood because of the experimental and analytical efforts made to investigate the performance of conventional light water reactor systems. Advanced reactors however still pose significant challenges to engineering analysis due to several unique design features. Understanding the effects of these features on local and system-wide thermal-hydraulics is necessary in order to confirm and quantify the expected safety margin of the proposed ALWRs. This section discusses those features and the thermal-hydraulic issues for advanced light water reactors.

Specifically, two advanced light water reactor systems are discussed; the AP 1000, and IRIS (International Reactor Innovative and Secure). Both designs rely on passive safety systems to insure adequate core cooling and prevent core uncovery. Preliminary assessments show that for each of these designs, the passive systems adequately remove decay heat for a wide spectrum of pipe ruptures. Confirmation of this safety margin depends on assessing the performance of these passive systems, and quantifying uncertainties associated with the thermal-hydraulic processes which they utilize.

The AP 1000 relys on passive safety systems for decay heat removal. Pipe breaks throughout the primary system will need to be considered as part of the design basis, as they are in conventional PWRs. The most critical accident scenarios in AP 1000 have been defined through past work on AP 600 Design Certification. The test programs conducted in support of the AP600 remain valid for many of the thermal-hydraulic processes that are important to the AP 1000. There are some thermal-hydraulic phenomena that are not well represented by previous tests for conditions expected during a hypothetical accident in an AP 1000 however. The major thermal-hydraulic issues for AP 1000 are primarily those thermal-hydraulic processes that are strongly dependent on the higher core steam production rate expected during an accident.

The major thermal-hydraulic issues for the AP 1000 include:

VI.2.2.1.2.2.1 Entrainment from horizontal stratified flow. Higher core steam production increases steam velocities in the hot leg and automatic depressurization system (ADS) during later phases of a small break loss of coolant accident. Sufficiently high steam velocities can entrain water from the hot leg and carry droplets into the ADS. This increases the pressure drop between the core and containment, and delays injection from the IRWST. New experimental data and models to predict this process are being generated. Currently, the staff is sponsoring a separate effects test program at Oregon State University to investigate phase separation at pipeline tees that will help satisfy this need. Integral tests in the Oregon State University APEX facility planned by DOE will also provide data useful in evaluating this process.

VI.2.2.1.2.2.2 Upper plenum pool entrainment and de-entrainment. High core steam production may entrain a significant amount of water from the pool in the upper

plenum during a small break LOCA. This may result in core uncovery for accident scenarios where the two-phase level drops below the bottom of the hot legs. Experimental data for prototypical upper plenum geometry is needed, and analytical models to account for entrainment and de-entrainment in the upper plenum are needed. The integral tests in the Oregon State University APEX facility by DOE will provide useful data on total vessel carry-over. Separate effects tests however may be needed and more effective in developing a database suitable for correlation and model development.

VI.2.2.1.2.2.3 Low pressure critical flow. Transition from high pressure phases of a small break accident to the IRWST injection period occurs while steam is vented through the ADS fourth stage. Because of the rapid depressurization, the flow remains critical with an upstream pressure that is much lower than pressures maintained in previous experiments used to examine critical flow. A lack of applicable data, and uncertainty in existing predictive tools is partly responsible for requirements in the AP600 Safety Evaluation Report (SER) for fourth stage ADS testing prior to operation. Currently, the staff is sponsoring experimental work at Purdue University using the PUMA facility to obtain these confirmatory data.

VI.2.2.1.2.2.4 Direct vessel injection. Flows from the core makeup tank (CMT) and IRWST are injected directly into the downcomer in the AP1000. This design feature is intended to reduce ECC bypass during a large break LOCA. Validation of models to predict bypass flows is made difficult because of the lack of experimental data for this injection geometry. Satisfactory resolution of ECC bypass for direct vessel injection may require new experimental data, and additional code validation. This need is being addresses internationally in support of the Korean advanced (conventional) reactor, which makes use of direct vessel injection.

The IRIS is a modular light water reactor with a power of up to 335 MWe. It makes use of passive safety systems to insure adequate core cooling, but because of the system design, the possibility for many of the conventional design basis accidents is eliminated. The steam generator, pressurizer, and coolant pumps are all internal to the reactor pressure vessel (RPV), which is contained within a relatively small containment shell. A loss of coolant from the RPV is expected to cause a rapid increase in containment pressure, which will subsequently reduce the rate of vessel inventory loss.

Because of the unique vessel design and intimate coupling between the vessel and a small containment, risk significant accident scenarios are not well defined. Few evaluations have been performed to identify the worst break location and failure conditions or to explore system response to a wide range of accident conditions.

The major thermal-hydraulic issues for IRIS include:

- Two-phase flow and heat transfer in helical tubes: The in-vessel steam generators for IRIS are of a modular helical coil design. The coils are located in the annular space between the core barrel and the vessel wall. Each of coil has an outer diameter of approximately 1.6 m. During loss of coolant accidents, heat transfer by the steam generators are an important mode of heat removal. Flow

conditions may vary significantly on the outside of the tubes, as the conditions change from forced flow to natural circulation during an accident. Prototypical experimental data will be need to determine internal, external and overall heat transfer coefficients for accident conditions. These data will be necessary to develop analytical models for computer codes to predict system response.

- Two-phase natural circulation: The IRIS design operates with a high level of natural circulation, with more than 40% of the total core flow caused by natural convection. During a LOCA, natural circulation through the core and within the vessel will be responsible for decay heat removal. Experimental data is needed to benchmark and verify computer codes to predict IRIS behavior during accident conditions.

- Containment – RCS interaction: A major difference between IRIS and conventional PWRs is the strong coupling between its small, passively cooled containment, and the primary system. Rapid pressurization and flooding of the containment are important processes in mitigation of a LOCA. The rapid change in pressure differential across the break will pose unique problems to code capability. New experimental data for critical break flow, and to evaluate system response due to rapidly changing containment backpressure will be needed. Modeling the vessel - containment interaction will use thermal-hydraulic codes for system response and containment response. Experimental data is needed to validate the codes used for the thermal-hydraulic simulation of the IRIS primary and containment.

- Parallel channel flow instabilities: Because the IRIS has an open lattice core, the core is essentially composed of many parallel channels with boiling taking place in the upper part of the core. As such, the system may be prone to two-phase flow instabilities. A confirmatory experimental investigation of conditions that might lead to instabilities in IRIS is warranted.

VI.2.2.1.3 Objectives and Planned Activities

VI.2.2.1.3.1 Related NRC Research

As mentioned above, work is underway at ORNL to modify the GRSAC code for its near-term use to support RES scoping and sensitivity studies for postulated accident sequences in pebble-bed and prismatic modular HTGRs. GRSAC will also be used to support TRAC-M development and assessment efforts. An effort to modify TRAC-M to add the currently identified capabilities is being initiated at Los Alamos National Laboratory.

VI.2.2.1.3.2 Related International Research

The IAEA sponsored an international standard problem modeling the conduction cooldown of a HTGR. Specifically, this effort was directed at modeling passive heat removal systems. This effort highlighted the importance of accurate modeling of heat sources and difficulties with modeling these passive systems. The results of this study are documented in IAEA TECDOC-1163.

As of this time, the information that has been identified in previous research and as a part of the IAEA work will be used. Additional data will be identified as part of a PIRT process that will focus the review of previous HTGR programs and the IAEA review of data to develop data needs for code development and assessment, and will include collaborative efforts with the international community.

The NRC has maintained an active, confirmatory thermal-hydraulics research program to better understand phenomena that are important to advanced passive plants such as the AP 1000. Central to this effort has been the experimental program conducted at Oregon State University using the APEX facility. APEX is a scaled integral effects facility, which has been used to simulate a wide range of accident scenarios applicable to the AP 1000. The facility is currently being upgraded to operate at higher power levels.

The NRC has also maintained an active experimental program using the PUMA facility. This facility is a scaled representation of an SBWR and has most recently been used to obtain experimental data for low pressure critical flow.

Separate effects test facilities have been established at Penn State University to investigate rod bundle heat transfer, and at Oregon Sate University to investigate entrainment from the hot leg to branch lines. Both of these facilities are expected yield experimental data important in predicting advanced plant behavior.

In addition to the experimental programs, the NRC is actively developing the TRAC-M thermal-hydraulics code for application to advanced passive plants. This code is applicable to the AP 1000, and has nearly all of the features necessary to model and simulate IRIS.

VI.2.2.1.3.3 Planned NRC Research Activities

NRC needs an independent capability for HTGR thermal-hydraulic analyses that has been thoroughly assessed and peer reviewed. The effort will be focused on adding the necessary capability for HTGR analysis to TRAC-M. This is the first priority. The staff will use a PIRT process to identify further development and experimental data needs. The results of the analysis could lead the staff into further code development activities and experimental data collection. At a minimum, the analysis will identify and rank relevant phenomena and assessment needs. The staff will assess the code according to the rankings of the analysis. An uncertainty analysis will be performed to assess the effect of code modeling relative to an as yet undetermined figure of merit. Finally, the staff code will need to be peer reviewed and validated.

VI.2.2.1.3.3.1 High Temperature Gas Cooled Reactors.

- TRAC-M Development: Confirm and modify as needed the capability to model flow and heat transfer in packed beds. Modify the porous medium solver and develop appropriate inputs for modeling of PBMR. Develop three-dimensional conduction and a spherical conduction models. Improved radiation model is also needed. Confirm and modify as needed the capability to model HTGR turbo-machinery. Confirm and modify as needed to capability the model natural circulation of gases. Add the capability to simultaneously model two different

working fluids. Along with this, the ability to track multiple non-condensible gas sources will need to be added (helium and air). Assess speed of the code and improve as necessary to allow for efficient simulation of transients on the order of days. Add graphite as a structural material. Update the GUI to work with HTGR designs. The deliverables will be the modified code with associated SQA documentation for HTGR analysis.

- PIRT analysis: Conduct analysis using PIRT methodology on thermal hydraulic issues for the HTGR's. The analysis will inlcude issues and sequence raised in early analysis and for the workshop. The deliverables will be ranking of NRC thermal-hydraulics issues for HTGR's.

- Develop Database: Development of need data, based on the analysis of the HTGR's designs and analysis methods, including development of test facilities to collect information needed to complete code validations. Appropriate data will also be collected for input deck preparation. The deliverables will be reports describing the facilities and the relevant data.

The NRC research objectives for AP 1000 and IRIS are to perform the experimentation and code development necessary to confirm compliance with 10 CFR 50.46 and to determine if there are conditions or accident scenarios that have unacceptable risk. For the AP 1000, an integral effects test facility exists, and separate effects tests are being conducted to develop data for models of critical importance. To fulfil these objectives for the AP 1000, a series of confirmatory tests, run under both design basis and beyond design basis accident conditions should be conducted in the APEX facility. These tests should be run at a power scaled to the AP 1000, and should be used as part of code development and validation for TRAC-M.

To meet these objectives for IRIS, a comprehensive test and analysis program should be conducted. While it is the applicant's responsibility to generate and provide experimental data sufficient to justify and license the design, the staff intends to supplement that data with confirmatory verification. As was done for AP 600, the staff intends to perform several independent, confirmatory tests at design basis and at beyond design basis conditions as a means of insuring the validity of regulatory decisions based on the applicant's submittals. Improved models for two-phase flow and heat transfer in helical coils need to be developed and implemented in the TRAC-M code, and the capability to predict the overall system performance demonstrated. The applicant's data, along with confirmatory NRC data will be used to develop these models. To simulate transients with strong vessel - containment interaction, it will be necessary to couple TRAC-M to a containment code such as CONTAIN. Models in the CONTAIN code for passive cooling, condensation, film coverage and non-condensible distribution would need to be assessed and improved.

VI.2.2.1.3.3.2 Advanced Light Water Reactors. APEX-AP 1000 Confirmatory Integral Testing: Provide data for code validation and to confirm safety margins. The APEX facility (currently being upgraded to represent AP 1000) will be used to develop an independent set of experimental data that can be used by the NRC to develop and refine its thermal-hydraulic tools so that they can be extended to AP 1000 plant analysis.

In addition, the tests will include accident scenarios and beyond design basis accidents that are beyond the scope normally addressed by the applicant. The tests, currently planned by DOE, are to confirm the safety margin that is expected in the AP 1000 design, and help identify any new processes or concerns not adequately addressed by thermal-hydraulic codes. The deliverables are experimental data and evaluation reports describing the tests themselves.

- AP 1000 Model Development and Separate Effects Testing: Obtain experimental data and develop thermal-hydraulic models for phase separation in hot leg - branch line connection necessary to benchmark analyses in support of AP 1000. Deliverables are separate effects test data, technical report(s) describing the data, and a technical evaluation report describing thermal-hydraulic models & correlations developed from the data and needed to represent important AP 1000 processes. This work is on-going at Oregon State University.

- AP 1000 code development and assessment: Assess TRAC-M for large & small break LOCA analysis in AP 1000. Insure that TRAC-M can produce reliable results for AP 1000 suitable to confirm licensing calculations and to explore beyond design basis behavior of the plant. Main objective is to qualify TRAC-M for independent assessment of AP 1000 behavior during LBLOCA, SBLOCA, and LTC. Deliverables include TRAC-M input decks for APEX-AP 1000 integral tests, code assessment reports, and a TRAC-M code version validated for AP 1000.

- IRIS code development and preliminary assessment: Develop special models (or at least a first-cut if data is insufficient) and perform initial independent assessment of IRIS behavior to wide range of design basis and beyond design basis scenarios. The code development and simulations will be used to identify major uncertainties and questionable plant behavior where experimental testing will be necessary to confirm margins and to develop improved models for thermal-hydraulic processes that need to be understood for IRIS. Special models and code issues that will need to be addressed for IRIS will likely include two-phase heat transfer and fluid flow in helical coils, critical flow, containment heat transfer, and primary-containment coupling. Assumes IRIS submittal in late 2003. (Staff does only technical follow and planning until submittal.) Main objective is development of thermal-hydraulic tools to perform independent assessment and to confirm safety margin. Deliverables include IRIS plant input deck, workable TRAC-M code version for IRIS application.

- IRIS Helical SG thermal-hydraulics: One of the important new features in IRIS is the integral helical SG. Some applicable data may currently exist from heat exchanger design data produced by the chemical and process industries. However, the geometric scale and conditions for those data are likely not sufficient for the NRC to develop and assess the IRIS SG in its code(s). Construction of a large scale test facility that can operate at high pressure (1000 psia) and acquisition of data for a series of two-phase tests, is expected to cost several million dollars. It is the applicant's responsibility to obtain data necessary to justify the IRIS SG design and its behavior during accident conditions. The NRC may find it cost efficient to participate in tests conducted

by industry to obtain independent data or to explore thermal-hydraulic conditions beyond those of interest to the applicant. The cost estimate as part of this plan assumes participation by the NRC is limited to technical follow in an industry sponsored facility.

- IRIS Integral Testing: The integral behavior of the IRIS primary system and the containment is new and not well understood. Like other plant designs, integral test facilities are vital in investigating accident scenarios, producing data necessary to validate thermal-hydraulic codes, and confirming safety margins. Such data will be needed by the NRC for independent confirmation and assessment of the IRIS design. It is the applicant's responsibility to obtain data or perform analysis that support the design and its behavior during accident conditions. The NRC may find it cost efficient to participate in tests conducted by industry to obtain independent data and to explore thermal-hydraulic conditions beyond those of interest to the applicant. The cost estimate as part of this plan assumes participation by the NRC early in the construction phase of the facility, and use of the facility for an independent series of experimental tests. The approach is similar to the NRC's participation in APEX, which was constructed by industry and later the staff.

- IRIS code and model development: Assess TRAC-M for LOCA (and possibly SGTR) analysis in IRIS. Insure that TRAC-M can produce reliable results for IRIS suitable to evaluate licensing calculations and to explore beyond design basis behavior of the proposed design. Main objective is to qualify TRAC-M for independent assessment of IRIS behavior using integral and separate effects test data from industry sponsored test programs applicable to IRIS. Deliverables are code validation reports, and a code version validated for the IRIS plant design, and several re-calculations of the IRIS plant using the now more refined code version.

VI.2.2.1.4 Application of Research Results

This research will be applied to the developing and demonstrating the ability to predict the behavior of the new plant designs under normal and accident conditions. Results from the research activities described above will be applied to enable and support the staff's independent assessment of thermal hydraulic issues associated with the respective advanced reactor designs. The importance of the research results is heightened by the fact that the NRC has had little recent experience at analyzing issues associated with new reactor designs that differ significantly from current LWRs with regard to the safety-related phenomena encountered in in-reactor and out-of-reactor nuclear analysis.

As outlined in the preceding sections, the thermal hydraulic research activities will result in developing the staff's technical insights in these areas and applying those insights toward establishing and qualifying independent analysis tools and capabilities. The development activities include the assessment of validation issues and modeling approximations in order to inform the staff's evaluation and treatment of potential biases and uncertainties.

VI.2.2.2 Nuclear Analysis

VI.2.2.2.1 Background

The term "nuclear analysis" describes all analyses that address the interactions of nuclear radiation with matter. Nuclear analysis thus encompasses the analysis of: (a) fission reactor neutronics, both static and dynamic, (b) nuclide generation and depletion as applied to reactor neutronics and to the prediction of decay heat generation, fixed radiation sources, and radionuclide inventories potentially available for release, (c) radiation transport and attenuation as applied to the evaluation of material damage fluence, material dosimetry, material activation, and radiation protection, and (d) nuclear criticality safety, (i.e., the prevention and mitigation of critical fission chain reactions ($k_{eff} \geq 1$) outside reactors).

This section of the advanced reactors research plan addresses nuclear analysis issues encountered in the evaluation of reactor safety. Nuclear analysis issues concerning radiation protection, material safeguards, and out-of-reactor materials safety at the front and back ends of the advanced reactor fuel cycles (i.e., fuel enrichment, fabrication, transport, storage, and disposal) are discussed in other sections of the plan.

VI.2.2.2.2 Purpose

The purpose of the research activities described in this section of the plan is to provide the nuclear analysis tools, data, and knowledge bases that will be needed in conducting the staff's reactor licensing safety evaluations for the respective advanced designs. In identifying the necessary research efforts, the staff has first sought to identify the nuclear-analysis related issues that affect reactor safety.

The following subsection begins with a brief discussion of the nuclear data libraries that are fundamental to all areas of nuclear analysis. Subsequent subsections discuss specific analysis issues grouped under the headings "Reactor Neutronics and Decay Heat Generation" and "Material Activation and Damage Fluence."

All areas of nuclear analysis make use of nuclear data libraries derived from files of evaluated nuclear physics data, such as ENDF/B in the U.S., JEF in Europe, or JENDL in Japan. The nuclear data files include, for example, fundamental data on radionuclide decay as well as neutron reaction cross sections, emitted secondary neutrons and gamma rays, and fission product nuclide yields, all evaluated as complex functions of incident neutron energy. The neutron reaction evaluations also provide cross-section uncertainty information in the form of covariance data that can now be processed and used with advanced sensitivity and uncertainty analysis techniques, as developed in recent years under RES sponsorship, to assist in the identification and application of appropriate experimental benchmarks for problem-specific code validation.

Many of the processed nuclear data libraries in use today were developed in the 1980s or earlier. For example, the PBMR design team in South Africa now relies on the German VSOP reactor physics code with multi-group nuclear cross section libraries derived in the early 1980s from the evaluated physics data in ENDF/B-IV. Pre-1990s

cross section libraries are similarly being used for preparing the LWR nodal physics data used by the NRC's reactor spatial kinetics code, PARCS, and for the criticality, depletion, and shielding analysis sequences in the NRC's SCALE code system. While these legacy cross section libraries have proven largely adequate in a variety of applications, they have known limitations and shortcomings that need to be considered in application.

In response to a 1996 user need memorandum, RES has sponsored ORNL to upgrade the AMPX code suite to enable its eventual use in creating new cross section libraries that would take full advantage of the expanded resolved resonance ranges and the improved/corrected nuclear data and covariance evaluations now available in the latest releases of ENDF/B-VI and its foreign counterparts JEF2.3 and JENDL-3. With the recently completed AMPX upgrades as well as continued improvements to the NJOY nuclear data processing codes, both opportunity and motivation now exist to produce and test state-of-the-art nuclear data libraries for use in the analysis of reactor safety and material safety and safeguards issues associated with conventional and advanced reactor technologies.

The nuclear heat sources of importance in all reactor safety analyses are primarily those arising from nuclear fission and the decay of radionuclides produced by nuclear fission and neutron activation. Reactor neutronics codes are used to predict fuel burnup and the dynamic behavior of neutron-induced fission chain reactions in response to reactor control actions and system events. Under subcritical reactor conditions, where the self-sustaining fission chain reactions have been terminated by passive or active means, the decay of radioactive fission fragments and activation products becomes the dominant nuclear heat source.

The results from accident sequence analyses provide information that may be used in plant PRAs for assessing event consequences and their probabilities. Core neutronics codes, generally coupled with thermal-hydraulic (T/H) and/or severe-accident (SA) systems codes, are needed for evaluating the dynamic progression of accident sequences that involve reactivity transients. For accident sequences in which the self-sustaining fission chain reaction is terminated by active or passive means, the T/H and SA codes used in evaluating the thermal response of the subcritical system (e.g., maximum fuel temperatures) must employ algorithms that represent the intensity, spatial distribution, and time evolution of the decay heat sources.

VI.2.2.2.2.1 HTGR Core Neutronics and Decay Heat Generation

The defining features of HTGRs include their use of fission-product retaining coated fuel particles, graphite as the moderator and structural material, and neutronically inert helium as the coolant. Both the PBMR and GT-MHR are modular HTGR designs that are fueled with low-enrichment uranium (LEU) instead of the high-enrichment uranium (HEU) and thorium used in earlier HTGRs. Both also have long annular core geometries and locate control and shutdown absorbers in the graphite reflector regions. In many respects, the PBMR and GT-MHR designs therefore have similar code modeling and validation issues for the prediction of reactor neutronics phenomena and decay heat generation.

Reactor neutronics and decay heat analysis issues unique to the PBMR relate mainly to its use of multiple-pass on-line fueling, its pebble-bed annular core with statistical packings of fuel pebbles of varying burnups, the intermixing of graphite pebbles and fuel pebbles near the boundaries between the fueled core region and the central graphite region, and the potential for seismic compaction events, misloading events, anomalous local packing and clustering of pebbles, and anomalous flow patterns of pebbles through the core such as might be caused by localized pebble bridging, jamming of chipped or fractured pebbles, unanticipated funneling effects near the core exit, or unanticipated radial gradients of pebble flow velocity resulting from the strong temperature dependence of pebble-to-pebble friction (i.e., as seen in the THTR-300 pebble bed reactor). Related research activities on the mechanics of pebble beds, including pebble flow and intermixing, statistical packing, bridging, and seismic pebble-bed compaction, are included in the plan section on Materials Analysis.

Physics analysis issues unique to the GT-MHR relate mainly to the effects of burnable poisons, the presence of both "fissile" and "fertile" coated fuel particles (with 19.9% enriched and natural uranium, respectively) in the fuel compacts, reactivity control for cycle burnup effects, and the power shaping effects of zoned fuel and poison loadings.

Nuclear analysis issues anticipated in evaluations of PBMR and GT-MHR reactor safety include the following:

VI.2.2.2.2.1.1 Temperature coefficients of reactivity. Ability is needed to confirm that the reactivity feedback effects from temperature changes in the fuel, moderator graphite, central graphite region, and outer reflector graphite are appropriately treated in the applicant's safety analyses. Based on sensitivity analyses and validation against representative experiments and tests, the evaluations should assess and account for computational uncertainties in the competing physical phenomena, including for example the positive contributions to the fuel and moderator temperature coefficients associated with ^{135}Xe and bred fissile plutonium.

VI.2.2.2.2.1.2 Reactivity control and shutdown absorbers. The reactivity worths of in-reflector control and shutdown absorbers are expected to be sensitive to tolerances in the radial positioning of the absorbers. The tests and analytical evaluations for reactivity control and hot and cold shutdown should also account for absorber worth variations through burnup cycles (GT-MHR) and the transition from initial core to equilibrium core loadings as well as absorber worth validation and modeling uncertainties and absorber worth variations caused by temperature changes in the core and reflector regions, xenon effects, variations or aberrations of pebble flow, and accidental moisture ingress.

VI.2.2.2.2.1.3 Moisture ingress reactivity. Although the absence of high-pressure, high-inventory water circuits in closed Brayton cycle systems makes this issue less of a problem than in earlier steam cycle HTGRs, the effects of limited moisture ingress will nevertheless have to be evaluated for depressurized or underpressurized accident conditions in the PBMR and GT-MHR. Effects to be evaluated include the moisture reactivity itself (i.e., from adding hydrogenous moderator to the undermoderated core) as well as the effects of moisture on temperature coefficients (e.g., from spectral

softening), shortened prompt-neutron lifetimes (i.e., faster thermalization), and reduced worths of in-reflector absorbers (i.e., fewer neutrons migrating to the reflector).

VI.2.2.2.2.1.4 Reactivity transients. T/H-coupled spatial reactor kinetics analyses will be needed for assessing axial xenon stability as well as reactivity transients caused by credible events such as overcooling, control rod ejection, rod bank withdrawal, shutdown system withdrawal or ejection, seismic pebble-bed compaction, and moisture ingress. Of particular importance in the safety evaluations for PBMR and GT-MHR is the need to identify, through safety analysis and risk assessment efforts, any credible events that could produce a prompt supercritical reactivity pulse. Should any such prompt-pulse events be identified as credible, their estimated probabilities and maximum pulse intensities should be considered in establishing any related plans or requirements for pulsed accident testing and analysis of HTGR fuels (see Section on Fuels). For loss-of-cooling passive-shutdown events with failure of the active shutdown systems (i.e., ATWS), the delayed recriticality that occurs after many hours of xenon decay may also require spatial kinetics analysis models to account for the unique spatial power profiles and feedback effects caused by the higher local reactivity near the axial ends and periphery of the core where temperatures and xenon concentrations are lower.

VI.2.2.2.2.1.5 Pebble burnup measurements and discharge criteria. The PBMR designer states that selected fission-product gamma rays will be measured to determine the burnup of each fuel pebble and that this measured burnup will serve as the criterion for discharging the pebble or passing it back through the reactor. The particular burnup value used as the discharge/recycle burnup criterion will be chosen to limit the maximum pebble burnup, which is stated as nominally 80 GWd/t. Therefore, determining a suitable value for discharge/recycle burnup criterion (<80 GWd/t) will require consideration of in-core pebble residence time spectra, together with supporting neutronics calculations, in order to statistically characterize the maximum burnup increment that might accrue during a pebble's final pass through the core. Burnup measurement uncertainties will also have to be considered. Furthermore, since pebble burnup measurements (unlike the pebble reactivity measurements used in THTR-300) cannot distinguish pebbles with different initial fuel enrichments, the same discharge burnup criterion will need to be applied to the initial charge of 4%-enrichment fuel pebbles as to the 8%-enrichment pebbles that are added in transitioning to an equilibrium core. Neutronics calculations will be needed to bound the higher neutron fluence experienced by the 4%-enrichment pebbles in reaching the maximum burnup levels allowed in the transitional cores.

VI.2.2.2.2.1.6 Pebble-bed hot spots. The results of melt-wire experiments conducted in the German AVR test reactor demonstrated the existence of unpredicted local hot spots under normal operating conditions in pebble bed cores and that such hot spots determine the maximum normal operating temperatures of the fuel. These hot spots may arise from a combination of higher local power density (e.g., due to moderation effects near the reflector wall or from chance clustering of lower burnup pebbles), lower local bed porosity due to locally tight pebble packings, and reduced local helium flow due to the increase of helium viscosity with temperature. Whereas the slow evolution of loss-of-cooling heatup transients in the PBMR will tend to wash out any effects of pre-accident local flow starvation on subsequent peak fuel temperatures, the effects of

higher local fission power densities will be retained throughout the heatup transient in the form of higher local decay heat powers. Therefore, the effect of decay-power hot spots, in particular, may need to be considered in evaluating the maximum fuel temperatures arising in pressurized or depressurized loss-of-cooling accidents.

VI.2.2.2.2.1.7 Pebble fission power densities and temperatures. The computational models may therefore need to account for pebble-to-pebble burnup and power variations within nodes. Note that in calculating operating temperatures inside a pebble, the reduction of pebble power with pebble burnup may tend to be offset by the reduction of graphite thermal conductivity with neutron fluence.

VI.2.2.2.2.1.8 Pebble decay heat power densities. Much as with fission power densities (see previous item), each node in the core calculational model will contain pebbles with a broad range of decay heat power densities. Computational studies may therefore be needed to establish technical guidance on accepted modeling approximations (e.g., nodal averaging methods) and assumptions (e.g., local hot spots, power histories) for calculating decay heat sources in pebble bed reactors while accounting for validation uncertainties associated with the shortage of applicable experimental data.

VI.2.2.2.2.1.9 Graphite annealing heat sources. Although continuous annealing effectively prevents any significant buildup of Wigner energy at the high operating temperatures of HTGR graphite, there is nevertheless a significant accumulation of higher-energy graphite lattice distortions that anneal out only at the elevated graphite temperatures encountered in loss-of-cooling accidents (e.g., conduction cooldown events). This high-temperature annealing heat source should be evaluated and, where significant, added to the nuclear decay heat sources used in the analysis of loss-of-cooling heatup events. (Note that the recovered thermal conductivity caused by high-energy lattice annealing during slow graphite heatup accidents can substantially reduce the peak fuel temperatures reached during the accident, an effect that has traditionally been credited in the heat removal models used for MHTGR accident analyses).

VI.2.2.2.2.2 ALWR Core Neutronics and Decay Heat Generation
Reactor neutronics and decay heat analysis issues for AP 1000 are essentially identical to those for AP 600 and the current generation of PWRs, with, for example, their gradual evolution to the higher initial enrichments and new burnable poison designs needed for higher burnups and longer cycles. Neutronics and decay heat analysis issues specific to the IRIS design include the following:

VI.2.2.2.2.2.1 Fuel depletion modeling. Depletion analysis of the IRIS fuel designs, with their >5% initial enrichments, significantly higher moderator-to-fuel ratios, novel burnable poison designs, and higher design burnup levels, may call for flux-solver methods and modeling practices more advanced than those traditionally used in analyzing conventional PWR fuels. Modeling studies with higher order methods (e.g., Monte Carlo) will be needed to assess such depletion modeling issues and develop appropriate technical guidance.

VI.2.2.2.2.2.2 Fuel depletion validation. The available experimental data base for validating LWR fuel depletion analysis methods consists largely of destructive radiochemical assays performed in the 1970s and 80s on rod segments from a dozen or so discharged PWR and BWR fuel assemblies. The database includes essentially no data from fuel rods with integral burnable poisons, initial enrichments above 4%, or burnups beyond 40 GWd/t. Sensitivity analyses, based on methods developed in recent years under RES sponsorship, will be needed to help assess the applicability of the existing validation databases to the IRIS fuel designs (with their >5% enrichments, significantly higher moderator-to-fuel ratios, advanced burnable poison designs, and burnup levels to 80 GWd/t) and to assist in the prioritization of further data needs and the estimation of remaining validation uncertainties.

VI.2.2.2.2.2.3 Neutronics of high-burnup cores. The IRIS concept of a 5- to 8-year straight-burn core without fuel shuffling poses a number of issues concerning the neutronics analysis of its initially highly poisoned and subsequently highly burned core. Current LWR experience makes relatively modest use of burnable poisons and is limited to shuffled core-average burnup values less than 35 GWd/t, whereby fresher fuel assemblies are typically placed in close proximity to those approaching design burnups of 60 GWd/t or less. Cumulative uncertainties associated with poison and fuel burnup effects, even at moderate burnups, will have greater neutronic significance in IRIS than in shuffled PWR cores. Neutronic phenomena affected by such analysis uncertainties would include temperature coefficients, spatial power profiles, control worths, shutdown margins, and kinetic parameters like effective delayed neutron fraction and prompt neutron lifetime.

VI.2.2.2.2.2.4 Decay heat power. Due to depletion modeling issues and the apparent shortage of available radioisotopic or calorimetric validation data applicable to the IRIS fuel designs at high burnup (see related items i and ii above), specific technical guidance will likely be needed on accepted methods for computing decay heat sources with appropriate consideration of validation uncertainties.

Nuclear analysis issues may arise concerning in-reactor radiation shielding analysis, material activation, damage fluence and dosimetry. Such analysis issues might concern, for example, the prediction and monitoring of local fluence peaks and the material damage or activation caused by radiation streaming through complex geometries, including any gaps that may develop over time between HTGR graphite reflector blocks. The importance of such nuclear analysis issues will depend on an assessment of related materials performance issues, such as the safety margins and uncertainties associated with graphite deformation and damage or the radiation-induced embrittlement of the pressure vessel or other metallic components.

VI.2.2.2.3 Objectives and Planned Activities

The NRC research objectives are to establish and qualify the independent nuclear analysis capabilities that are needed to support the evaluation of an applicants' reactor safety analyses for the respective advanced reactor designs.

VI.2.2.2.3.1 Related NRC Research

- For PBMR, GT-MHR, and IRIS, relevant past, ongoing, and planned NRC research efforts include the following:

- RES in-house analysis and contractor projects conducted in the late 1980s and early 1990s in supporting the staff's preapplication safety evaluation of the DOE MHTGR.

- Recently completed RES-sponsored work on (1) upgrading the AMPX code system for use in creating state-of-the-art nuclear data libraries, (2) the development of sensitivity and uncertainty analysis methods that utilize cross section covariance data, (3) modeling and validation guidance for computing radionuclide inventories in high-burnup LWR fuels, and (4) guidance on modeling and validation uncertainties in computing the reactivity of spent PWR fuel.

- Ongoing RES projects and tasks: (1) Modular HTGR Accident Analysis (ORNL), (2) TRAC-M code model development for modular HTGRs, (3) Initial PARCS code modifications to incorporate the R-Theta-Z geometry needed for PBMR analysis, (4) MELCOR code model development for modular HTGRs.

- Future NRC research on the HTGR and ALWR technical areas described in other chapters of this research plan (e.g., Materials Analysis, Thermal-Fluid Dynamic Analysis, Fuel Performance, Severe Accident Analysis).

- Ongoing RES tasks at ORNL to complete the development of 2D-depletion lattice physics analysis sequences (NEWT/ORIGEN-S) in the NRC's SCALE code system for use in exploratory studies and preparing design-specific nodal physics data tables for input to the NRC's PARCS spatial kinetics code.

VI.2.2.2.3.2 Related Domestic and International Cooperation

Opportunities for HTGR-related domestic and international cooperation include the following:

- Establish a cooperative research agreement with MIT that includes sharing of pebble-bed reactor physics codes and models as well as related code development and analysis tasks.

- Acquire HTGR physics benchmark data from the international HTR-PROTEUS program conducted in the early 1990s at PSI, Switzerland. (Room temperature only, ordered and random pebble beds, 15-20%-enriched LEU fuel, Pu sample worths, moisture ingress worths, in-reflector absorber worths)

- Acquire HTGR physics benchmark data from Russia, including GROG and ASTRA experiments as well as any newer physics experiments supporting the design and safety analysis for the Pu-burning GT-MHR in Russia. [Also pulsed test data on fresh HTR fuel.]

- Evaluate feasibility and technical merits of acquiring existing benchmark data from British Magnox, AGR, and early HTR programs, including BICEP, Dungeness B, and various HTGR-related experiments done in the 1970s by Winfrith and British Energy.

- Where relevant, acquire existing HTGR physics benchmark and test data from Fort Saint Vrain testing and operations; the CNPS experiments at LANL; the THTR-300 testing and operations, AVR testing and operations, and the KAHTR experiments in Germany; and the CESAR experiments in France.

- Acquire existing and new HTGR physics benchmark data from HTR-10 in China.

- Acquire existing and new HTGR physics benchmark data from VHTRC and HTTR in Japan.

- Join and add new physics benchmarking activities to the IAEA's ongoing CRP on safety performance of HTGRs. Such activities could include code-to-code benchmarks, but might also introduce additional experimental benchmarks taken from various sources such as recent and planned benchmark measurements at HTR-10 in China and HTTR in Japan, as well as a number of potentially relevant past experiments and operating tests from British activities with Magnox, AGR, and HTR technology. Note that the proposed additional benchmarking efforts would fill a number of validation gaps not addressed by programs to-date, including the international HTR-PROTEUS experiments described in the recently issued IAEA TECDOC and its references.

- Participate in existing and propose new physics benchmarking efforts within the OECD/NEA's Nuclear Science and/or Nuclear Safety activities related to HTGRs. (Note that OECD has recently taken over some HTGR activities formerly conducted by the IAEA).

- Participate in selected existing and planned HTR-N activities of the European Commission.

- Participate in efforts to expand the existing International Criticality Safety Benchmark Evaluation Project to include the documentation and evaluation of existing and new graphite-moderated benchmark experiments relevant to PBMR and GT-MHR neutronics.

Potential areas of ALWR-related interoffice, domestic, and international cooperation include the following:

- Through a PIRT process, identify and acquire relevant insights from recent and ongoing efforts to assess biases and uncertainties in computing the isotopic composition and reactivity of moderate- and high-burnup PWR fuels. RES staff could seek interoffice cooperation with staff in NMSS/DWM and NMSS/SFPO, as well as cooperation with the DOE Yucca Mountain Project, concerning the application of burnup credit in the criticality safety analysis for spent fuel

management systems. (See also related NRC research in previous section, items (i)(3) and (i)(4)).

To fill and technology gaps above and beyond an applicant's responsibility, RES could:

- Identify and acquire relevant LWR physics benchmark data from the international LWR-PROTEUS program now underway at PSI, Switzerland, and explore possibilities for extending the cooperative program to include specific IRIS-related benchmarks.

- Identify and acquire relevant LWR physics benchmark data from the ongoing international REBUS program in Belgium (formerly co-sponsored by RES) and from recent work at the ECOLE and MINERVA facilities of CEA/Cadarache in France, and explore possibilities for cooperative work on additional benchmark experiments to address specific IRIS validation issues.

- Pursue active NRC participation in relevant international programs, including experiments (e.g., items ii and iii above), code-to-data benchmarks, and code-to-code benchmarks, conducted by the IAEA, the European Commission, OECD/NEA.

VI.2.2.2.3.3 Planned NRC Research Activities
Listed below are the planned research activities pertaining to the nuclear analysis issues described previously:

VI.2.2.2.3.3.1 Preparation of modern cross-section libraries. Using the upgraded AMPX code system, supplemented by NJOY as needed, prepare state-of-the-art master cross section libraries for use in performing exploratory and confirmatory analyses on reactor safety and material safety issues. Test and verify the resulting cross section libraries by using them in selected benchmark calculations pertaining to reactor neutronics, criticality, depletion, and radiation shielding. The resulting cross section libraries will be generically applicable for nuclear analyses involving all conventional and advanced reactor technologies.

VI.2.2.2.3.3.2 Familiarization with applicant's codes and methods for core neutronics and decay heat in (1) PBMR, (2) GT-MHR, and (3) IRIS. In coordination with preapplication review activities, gain familiarity with the reactor neutronics codes and decay heat algorithms and associated analysis assumptions, validation data, and uncertainty treatments that are being used on behalf of the pre-applicants for their intended use in licensing-basis safety analyses. Incorporate insights and questions arising from this familiarization process into the prioritization, planning, and execution of the NRC's overall research efforts in this and related technical areas.

VI.2.2.2.3.3.3 Initial exploratory and scoping studies for core neutronics and decay heat in (1) PBMR, (2) GT-MHR, and (3) IRIS. Use available independent codes (e.g., GRSAC, MCNP/MonteBurns, SCALE/NEWT/SAS2D, WIMS/MONK, Venture 2000, PEBBED), and available applicant codes where needed, to perform exploratory and scoping analyses on selected issues such as described in Section (a) of this

chapter. Incorporate insights and questions arising from these exploratory and scoping studies into the prioritization, planning, and execution of the NRC's overall research efforts in this and related technical areas.

VI.2.2.2.3.3.4 Preparation and testing of spatial kinetics models of (1) PBMR, (2) GT-MHR, (3) IRIS, and (d) AP 1000. Develop PARCS input models and, using appropriate lattice physics and depletion analysis tools with state-of-the-art cross section libraries (see previous item), prepare the design-specific nodal data tables needed for performing spatial kinetics analyses with the PARCS code (coupled with a thermal-hydraulics code).

VI.2.2.2.3.3.5 Validation and testing for core neutronics in (1) PBMR, (2) GT-MHR, and (3) IRIS. Review the planned reactor startup and operational tests and measurements related to reactor neutronics. Review existing and planned validation databases (e.g., critical experiments, worth measurements, reactor tests) and perform sensitivity analyses, based on methods developed in recent years under RES sponsorship, to help assess their applicability to design-specific reactor neutronics phenomena and to help prioritize further data needs and assess remaining validation uncertainties. Participate in cooperative programs for acquiring new experimental data and conducting relevant code-to-data and code-to-code benchmarking activities.

VI.2.2.2.3.3.6 Validation for depletion and decay heat in (1) PBMR, (2) GT-MHR, and (3) IRIS. Review existing and planned validation databases (e.g., spent fuel isotopic assays and decay heat calorimetry) and perform sensitivity analyses, based on methods developed in recent years under RES sponsorship, to help assess their applicability to the respective fuels and operating parameters and to help prioritize further data needs and assess remaining validation uncertainties. Participate in cooperative programs for new experimental data as well as code-to-data and code-to-code benchmarking activities.

VI.2.2.2.3.3.7 Shielding and material fluence analyses for PBMR and GT-MHR. Specific HTGR shielding and material fluence issues will be identified in coordination with assessment activities described in the sections on High-Temperature Materials and Nuclear-Grade Graphite. Issues for which specific nuclear analysis tools and models may be needed include fluence damage to the vessel and other metallic components, fluence dosimetry requirements and interpretation, radiation streaming through gaps between radiation-warped graphite reflector blocks, and radiation shielding and protection of plant workers.

VI.2.2.2.4 Application of Research Results

Fundamental to reactor safety analysis is the ability to predict the fission and decay heat sources that arise under credible normal and accident conditions. Results from the research activities described above will be applied to enable and support the staff's independent assessment of nuclear analysis issues associated with the respective advanced reactor designs. The importance of the research results is heightened by the fact that the NRC has had little recent experience at analyzing issues associated with

new reactor designs that differ significantly from current LWRs with regard to the safety-related phenomena encountered in in-reactor and out-of-reactor nuclear analysis.

As outlined in the preceding sections, the nuclear analysis research activities will result in developing the staff's technical insights in these areas and applying those insights toward establishing and qualifying independent analysis tools and capabilities. The development activities include the assessment of validation issues and modeling approximations in order to inform the staff's evaluation and treatment of potential biases and uncertainties in the computed nuclear heat sources. Especially important in this context is the development of state-of-the-art master cross section libraries. The resulting master cross section libraries will play a fundamental role in all nuclear analysis activities for reactor safety (and out-of-reactor material safety) and will be generically applicable to all technologies associated with conventional and advanced reactors.

VI.2.2.3 Fission Product Transport (source term) Analysis

VI.2.2.3.1 Background

The NUREG-1150 study and subsequent reactor risk studies performed by NRC and industry have shown that public risk from reactor operation is dominated by accidents involving severe core damage coupled with containment bypass or containment failure. These accidents result from sustained loss of core cooling and can release quantities of radioactive fission products into the environment. The ability to model progression of severe accidents and estimate releases of fission products into the environment is required to be able to quantify risk to address severe accident issues. The NRC has developed several codes to model severe accidents. These codes have been used to develop and improve NRC regulations dealing with severe accident issues, such as 10 CFR 50.67, "Accident Source Term."

The NRC's severe accident codes are based on a large number of experiments performed in the 1980's following the Three Mile Island 2 accident and include MELCOR, SCDAP/RELAP5, CONTAIN, VICTORIA, and IFCI. Except for the MELCOR code the other codes are not being activity maintained. MELCOR is an integrated code which can model most aspects of a severe accident including thermal hydraulics, core melt progression, and fission product release. A number of experiments have also been carried out in support of development of a fundamental of the phenomena of severe accident and fission product transport. The recent NRC focus in severe accident has included developing of an upgrade and benchmarking of MELCOR against the more specialized severe accident codes and experimental results.

As part of the NRC's review of advanced reactors development of source terms will play an important part in the policy issues such as the need for leak tight containments, the need for and size on emergency planning zones, and the choice of design bases accidents. There is a need for data and modeling methods for the new materials and configurations that will be use in the advance reactors (particularly high temperature gas cooled reactors). Research will be needed to support both the development of infrastructure to preform confirmatory analysis and to identify and resolve many of the source term driven policy issues discussed above.

VI.2.2.3.2 Purpose

Severe accidents leading to fission product release need to be modeled. For today's LWRs, such accidents include a loss of coolant coupled with the failure of safety systems, reactor coolant system boundary failure, and containment failure or bypass. Accordingly, severe accident codes have been developed and used to estimate the probability and timing of the failure of the reactor coolant system boundary and the failure or bypass of the containment. Severe accident analysis methods using codes such as MELCOR have been developed to estimate the magnitude and timing of fission product release to the containment and subsequently to the environment.

Severe accident and source term analysis will likewise be needed for advanced reactors to support the development of limiting sequences and to confirm applicants' analysis of the plants. Therefore, severe accident codes, data and the expertise to apply them will be needed for advanced reactors to estimate overall plant risk as well as to address individual safety issues.

For advance light water reactors, the evolution of severe accidents and source terms will be similar to the current generation of plants. For HTGRs, both the types of sequences and the process by which fission products may be released from HTGR fuel will be different. As a result of diffusion during normal operation, rupture of coated fuel particles as a result of accidents, and vaporization during high-temperature degradation of the fuel, fission products may be released. The section on Fuel Analysis covers releases from diffusion during normal operation and from rupture of coated fuel particles as a result of accidents. This section covers the experimental database for release of fission products during high-temperature fuel degradation.

The fission products are transported through the primary system and containment prior to reaching the environment. During this transport, the fission products are largely in aerosol form and therefore may deposit in the primary system and containment. This deposition can significantly reduce the amount of fission products released to the environment. The research in this area covers the development of an experimental database for deposition of fission products in the primary system and containment following rupture of coated fuel particles and during high-temperature fuel degradation.

The risk from HTGR operation is the risk from releases during normal operation, from accidents involving rupture of coated fuel particles, and from accidents involving high temperature fuel degradation. Technical expertise in the area of fission product release during high temperature fuel degradation is needed to be able to assess the risk from HTGR operation. Because fission products released from the fuel are transported through the primary system and containment as aerosols, the offsite releases and offsite radiological consequences may be significantly reduced by fission product deposition in the primary system and containment. Aerosol deposition occurs through a variety of mechanisms such as gravitational settling, thermophoresis, and diffusiophoresis. Therefore, technical expertise is needed on the deposition of fission products in the primary system and containment. Ignoring deposition may result in overestimating the offsite releases and offsite radiological consequences.

VI.2.2.3.3 Objectives and Planned Activities

The MELCOR code contains sufficient modeling detail to be used to analyze most severe accident issues for operating reactors. It has been used during the past ten years to analyze a number of severe accident issues for operating reactors and advanced reactors including the AP 600 reactor. Therefore, MELCOR also can be used for the AP 1000 advanced reactor.

MELCOR also has most of the capabilities needed to analyze severe accident issues for HTGRs. However, four modifications to MELCOR are needed to model these reactors, because of the different fuel design and the different reactor internal structure design. The first modification, which is needed because of the different fuel design, is the capability to model the fission product release from the core and deposition in the reactor coolant system and containment. The second modification, which also is needed because of the different fuel design, is the capability to model the oxidation of graphite fuel under accident conditions potentially leading to higher fission product releases from the fuel. The third modification, which is needed because of the different reactor internal structure design, is the capability to model graphite structures. Finally, while MELCOR is used to analyze severe accidents in operating reactors that can extend over several days, some severe accidents in advanced gas cooled reactors can last considerably longer due to the lower power density. Therefore, because run time can be a problem for long duration accidents, MELCOR needs to be modified to allow use of longer time steps.

These modifications are described below, together with an activity to assess MELCOR against available experimental data and other codes.

VI.2.2.3.3.1 Validate Bottom Head Heat Flux Models

An issue of concern in the AP 600 review was the ability of an external pool of water to keep the bottom head of the AP 600 vessel cool and intact in the event that core damage should cause a debris bed to form inside of the vessel. The AP 1000 core is of considerably higher power density and may cause some concern with regard to the ability of the water pool to carry away enough heat to keep the bottom head of the vessel from failing. At present the OECD MASCA experiment is being performed to look at the melt chemical and thermal behavior is a simulated RPV lower head. The MELCOR models will be validated against this data and other data ensure the capability to assess sequences that include this phenomena.

VI.2.2.3.3.2 Extend Fission Product Release Models

Extend fission product release models in the code by expanding current fission release models which are based on CORSOR, CORSOR-M, or Booth formulation to predict release from advanced gas cooled reactor fuel (e.g., spherical fuel pebbles, block/prismatic fuel configurations). Where deemed appropriate, the effects of air or steam oxidation as well as burn-up should be included.

VI.2.2.3.3.3 Expand Oxidation Models
Expand the current oxidation models for various materials in the code to include a graphite oxidation model. Oxidants to be considered for the model should include oxygen, steam, and moist air. The oxidation model should account for CO and CO_2 as well as H_2 in the case of steam oxidation, where CO may further react with O_2. The model should be able to predict self-sustaining graphite fire. In addition to the graphite fire, smoke and particulate formation should be considered.

VI.2.2.3.3.4. Update Materials Properties Models
Expand the fuel and structural material components in MELCOR to include graphite. Graphite/fuel degradation and relocation modeling should be considered, as well as strength and integrity of core supporting structures. Core description considered should be general enough to allow description of both prismatic as well as PBMR core design.

VI.2.2.3.3.5 Improve Numerics
Improve MELCOR's numerics to allow use of longer time steps in order to carry out reasonable execution times for slowly developing accidents. This may involve changing the numeric solver for MELCOR to implement the SETS (semi-explicit-two-step) algorithm. This could be done as part of the MELCOR consolidation and modernization process.

VI.2.2.3.3.6 Assess Code Against Available Experimental Data and Other Codes
When model implementation in the MELCOR code is complete, assess the code against available experiments. Also, prepare input decks for selected advanced reactor designs, and demonstrate code capabilities for selected performance scenarios.

VI.2.2.3.3.7 Experimental Investigation of Fission Product Deposition and Transport to Support Database Development and Expertise in Fission Product Release and Transport for HTGRs Under Accident Conditions
To achieve this objective, a literature review will be performed of HTGR experiments on fission product release during high temperature fuel degradation and deposition in the primary system and containment under accident conditions. Because fission product aerosol deposition is increased by the release of non-fission-product aerosols from the core, this literature review would include experiments on aerosol releases of other core materials under accident conditions. Based on the results of the literature review, the need for additional experiments will be assessed. This literature review and assessment of the need for additional experiments will be performed by NRC staff over several months. Then, additional experiments would be performed as needed.

VI.2.2.3.3 Application of Research Results

The result of the above research will be a version of the MELCOR integrated severe accident code that could be used to analyze the progression of severe accidents in advanced reactors. This version of MELCOR could be used to independently confirm an applicant's safety calculations, identify the need for safety enhancements or other regulatory action, provide guidance for NRC reviewers, and provide the technical basis of criteria for acceptability. The major issues covered by MELCOR are the probability and timing of the failure of the reactor coolant system, the probability and timing of

containment failure or bypass, and the magnitude and timing of fission product release to the containment and subsequently to the environment.

The results of the database work will be used to develop and assess fission product release and deposition models in the MELCOR integrated accident analysis code. The development, validation assessment, and application of the MELCOR code to perform safety analysis for HTGRs will provide an essential capability supporting the staff's independent evaluation of the applicants' safety cases for PBMR and GT-MHR licensing.

VI.2.3 Fuel Analysis

VI.2.3.1 Background

MHTGRs, such as the Pebble Bed Modular Reactor (PBMR) and the Gas Turbine Modular Helium Reactor (GT-MHR) have unique safety features and safety characteristics. Foremost among these is the all-ceramic fuel element containing high integrity high performance TRISO coated fuel particles (CFPs).

The design of modular HTGRs involves many billions of CFPs contained within hundreds of thousands of graphite fuel elements (e.g., fuel pebbles, fuel compacts) that comprise the fueled core. The intended safety characteristic of the TRISO CFPs within these fuel elements is to provide the principal barrier and the primary containment function against the release of fission products to the environment during normal operation and design basis accidents. The release of fission product is a function of the sum of initial CFPs defects and heavy metal contamination from manufacture; the CFP failures that occur during normal plant operations, including anticipated operational transients; and the CFP failures that occur during design basis accidents or beyond the design basis (i.e., "severe") accidents.

HTGR applicants can propose that the accident source-term be based on models and methods that mechanistically predict fission product release from the fuel. Should this be the case, then it would be different from the traditional deterministic licensing approach to source term used by LWRs, which involves a pre-determined conservative upper bound for the accident source term. As in the past (MHTGR), HTGR applicants for modular HTGR will likely propose that these plants utilize non-leak-tight "confinement" structure rather than a traditional leak-tight and pressure retaining containment structure. Accordingly, for modular HTGRs, the licensing basis, and the safety analysis will hinge in large part, on the applicant's capability to confirm, as well as the NRC's capability to confirm, fuel fission product release, and address associated uncertainties.

The qualification of HTGR fuels will be based on a wide range of technical areas and specific factors that are known to influence fuel performance, such as fission product release and particle failure rates. The technical areas include: fuel design; fuel manufacturing process – including process specifications and statistical product specifications and; design-specific core operating conditions as well as the design-basis accident conditions and postulated beyond design-basis accident conditions. Key specific factors within the design-specific plant operating conditions that are known to effect fuel (particle) performance include: fuel operating temperature, fuel burnup, particle fast fluence, particle power and residence time in the core.

The key factor effecting fuel (particle) performance during accidents is the peak temperature in the particle(s) during the accident (following the prior degrading effects of the operating conditions). Temperature increases can occur due to heatup events, such caused by the loss of normal cooling or by core power increases, or by significant local reactivity increase events. Other factors potentially effecting fuel (CFP) performance during accidents can include the effects of chemical attack (e.g., oxidation) on the fuel element (and possibly) CFPs.

To have an effective capability to predict CFP performance and a deterministic approach to the source term capabilities in a number of interfacing technology areas will be needed. These include: time and space dependent nuclear analysis for: (1) fuel burnup, fast fluence (for particle coating behavior) and thermal fluence (for particle power and fuel kernel behavior) and fuel particle power during reactivity events; thermal-hydraulic analysis of both normal operating core temperature distributions, time and space dependent accident core temperature distributions, and time-dependent core temperature and flow distributions (for the onset and magnitude of fuel oxidation during postulated air intrusion events). The fission product release rate(s) from the fuel during normal operation and accidents are key inputs to the accident source term calculation which are addressed in another part of the plan.

Additionally, it will be essential to understand the safety margins, both qualitatively and quantitatively, to increases in CFP failure rates and increases in fission product release. These margins of safety will need to be known for normal operation design-basis accidents potential accidents beyond the design-basis. The first margin should be demonstrated by the applicant. The latter margin is on top of the first and involves the margins to the conditions that are beyond the fuel design-basis (e.g., fuel design specifications, fuel manufacturing specifications, fuel operating temperature limits, fuel burnup limits, particle fast fluence limits, particle power limits and residence time limits) Aspects of the second margin are also expected to be a pursued by an applicant, but would likely require additional RES staff investigation and study.

There is a range of significant fuel design, fuel manufacture, fuel quality and fuel performance issues which will require research initiatives by the respective applicant/vendor. Exploratory and confirmatory NRC research will also be required to support safety findings and conclusions.

The following paragraphs are provided to add further background insights that bear on the extent to which additional NRC regulatory research is needed in the area of HTGR fuel performance analysis. These paragraphs are intended to recognize that considerable worldwide research involving HTGR fuels with TRISO CFPs has already been conducted over the last 30 years and is currently underway. NRC's research in the area of fuel performance analysis should capitalize on this body of knowledge to establish the infrastructure of knowledge, data, and tools needed for HTGR reviews. It provides a base and context for which NRC research should be pursued to fill the infrastructure gaps without duplicating previous applicable reference work.

VI.2.3.1.1 Past Research

The design of HTGR fuels with TRISO coated fuel particles (CFPs) has evolved empirically over the last four decades. This evolution began with fuel elements utilizing fuel particles with a single anisotropic carbon layer. Later, fuel elements with BISO CFPs involving a layer of buffered isotropic pyrolytic carbon were developed, and, more

recently, fuel elements with TRISO CFPs have been qualified. This most recent design involves CFPs with a fuel kernel, a porous buffer layer, an inner pyrolytic carbon layer, a silicon carbide layer and an outer pyrolytic carbon layer. The fundamental characteristics of ceramic CFPs for HTGRs have also been investigated over this period. Several countries initiated fuel development and qualification programs with the coated particle as the basic unit. These efforts have addressed the design, design-analysis, manufacture irradiation testing, accident performance and utilization of these fuels in HTGRs.

In the early 1960s, the United Kingdom Atomic Energy Agency (UKAEA) initiated a CFP development program. The objective of the program was to define the essentials of CFP production and to identify the important process parameters which determine CFP properties, and thus its irradiation and accident performance. The fuel and materials development efforts included testing of a variety of CFPs in prismatic fuel elements which were involved the UK-OECD DRAGON project.

In the 1970s, in the Federal Republic of Germany (FRG), the production process for spherical fuel elements with BISO fuel was developed, established and licensed for use in the AVR and THTR. Later, in the early 1980s, a TRISO coated particle design with low enriched UO_2 was developed. This TRISO CFP design was later established as the reference fuel for the new FRG modular HTGR designs such as the HTR-Modul. The qualification program for the FRG TRISO fuel included a range of irradiation experiments in materials test reactors (MTRs) and the AVR and included aspects such as accident simulation testing. The FRG program was aimed at establishing the concept of a 1600°C limit for pebble fuel elements with TRISO CFPs. The concept was that TRISO CFP failures would not occur until well above1600°C, while the peak transient fuel temperature for a modular HTGR design would not exceed 1600°C during the most severe postulated accident. The FRG MTR fuel irradiation testing research on CFPs investigated such aspects as: particle performance (i.e., failure), fission product (FP) transport in the fuel kernel and FP transport in coating layers of intact particles, FP release from broken particles and the effects of chemical attack (e.g., moisture and air ingress) on particles. Fuel element (i.e., pebble) testing investigated aspects such as pebble surface wear and FP transport through the graphite matrix and included large scale demonstration tests in the AVR. "Proof" tests under simulated HTGR operating conditions were also carried out with test parameters chosen to envelope the selected HTGR's design conditions (e.g., operating temperature, burnup, fast fluence) followed by accident simulation heatup tests. Although the FRG HTGR developmental efforts were phased out during the 1990s, a significant number of unirradiated archive FRG reference fuel elements that were fabricated for use in the AVR are currently in storage at the Julich Research Center. This fuel is stated to be of the reference design and manufacture for the PBMR pebble fuel, but of higher enrichment. A number of these archive elements may be made available to NRC and other third parties for use irradiation testing programs.

Until recently, the International Atomic Energy Agency (IAEA) had a number of coordinated research programs related to the technical basis and safety performance aspects of HTGR fuels utilizing CFPs. These research programs are part of the broader International Working Group on Gas Cooled Reactors. The working group and the constituent programs, including the HTGR fuels program area, have served as fora for

the international exchange of technical information. Several meetings of technical specialists working in the area of HTGR fuels research and development have taken place, beginning in the early 1980s, and continuing during 1990s. Meeting topics have included, HTGR fuel development (1983), fission product release and transport in HTGRs (1985), behavior of HTGR fuel during accidents (1990), response of fuel elements and HTGR cores to air and water ingress (1993) and retention of FP in CFP and transport of FP (1992-1996). The proceedings from these meetings have been published and are publically available. Most recently, IAEA support for the gas reactor working group and associated coordinated research programs has substantially declined although limited periodic meetings among the international experts in different HTGR technology areas including HTGR fuels may still continue for some time.

Since 1985 the Japanese Atomic Energy Agency (JAERI) Research Institute conducted an HTGR R&D program in cooperation with the DOE under a DOE-JAERI memorandum of agreement. Under this agreement joint CFP fuel experiments were conducted and information was exchanged. However, the agreement was terminated in September 1995. Also since 1995, JAERI and the Julich Research Center (KFA) have carried out exchange of information in several HTGR safety arenas including fuel performance. The JAERI-KFA agreement ran from 1996 to 2001. Currently the NRC has an agreement with JAERI covering the exchange of technical information involving safety research and includes aspects such as HTGR fuel technology. A JAERI fuel irradiation test program to qualify the CFP fuel for HTTR operation has been completed and documented. The results were reviewed by the Japanese regulatory authorities in connection with the safety review and licensing of the HTTR. The JAERI fuel testing program has now entered the operational phase in which CFP fuel performance will be assessed on a large-scale as part of HTTR power operations.

In Japan, the reference HTGR fuel involves hexagonal prismatic graphite blocks utilizing graphite fuel rods containing fuel compacts with TRISO CFPs. The CFPs utilize a UO_2 kernel with customized coating layer thicknesses to achieve optimum performance for the operating and postulated accident conditions of the HTTR. The burnup limit for the HTTR fuel is significantly lower than the FRG or US designs. This is intended to accommodate the HTTR's higher fuel operating temperatures and higher peak fuel temperatures for a postulated reactivity insertion (rod ejection) accident. The Japanese fuel qualification program for the HTTR has been completed and included a range of bounding irradiation conditions in MTRs. This fuel is currently operating in its first cycle in the HTTR, which achieved full power operation in late CY 2001.

VI.2.3.1.2 Current Research

The U.S. DOE has announced its intention to pursue support for the development and implementation of a targeted fuel qualification program for PBMR fuel based on the German fuel fabrication process. This program is part of DOE's Advanced Gas Reactor Fuel Development and Qualification Program. In support of an application, an international fuel testing program has been proposed to jointly develop data that will be necessary for the technical basis for the safety analysis of the fuel. The testing program is intended to involve both German archive fuel in the near- term, and PBMR production fuel, when it becomes available for later testing later in the decade. The purpose of the

German archive fuel experiments is to develop fuel performance data for German fuel of their reference design and manufacture, but for PBMR design conditions. The tests are intended to establish a benchmark and validate the performance of the German fuel for the more demanding design conditions (e.g., higher operating temperature) of a PBMR compared to the earlier German pebble bed reactor designs. If this concept demonstration is successful, the later qualification and proof testing program for PBMR production fuel would then have to be demonstrated as achieving these same performance capabilities for these PBMR conditions. The test program is also intended to provide opportunities to explore the margins to failure of HTGR pebble fuels for test conditions that are well beyond the conditions associated with the fuel design basis.

In China, the current reference fuel design is very similar to the FRG reference pebble fuel with TRISO CFPs. The fuel manufacturing methods are also very similar to those used in Germany to manufacture their reference fuel. The fuel is designed for operation in the HTR-10 pebble bed reactor which is located at the Institute for Nuclear Energy Technology. Irradiation qualification testing of the fuel is currently underway in an MTR and its successful performance is a licensing requirement for power escalation of the HTR-10. As of late CY 2001, power escalation of the HTR-10 had not yet been authorized.

The European Commission (EC) is currently sponsoring approximately a $16M, 4-year research program on high temperature gas cooled reactors. The European HTGR program includes a project on HTGR fuel technology. The objectives of the program are: to re-establish the know-how that existed in the past in the areas of fuel design and fuel fabrication; to assess the performance of fuels with TRISO CFP at very high burnups; to develop a code for modeling HTGR fuel behavior under irradiation and; retrieve and evaluate data from past HTGR experiments with the aim of constructing a fuel database. Irradiation experiments on German archive fuel and GA compacts fabricated using a new manufacturing procure are expected to begin in CY 2002. The irradiation experiments will be followed by accident heat up simulations with fission product release measurements and post irradiation examinations. The purpose of the German archive fuel experiments is to develop fuel performance data for German fuel of their reference design and manufacture, but for conditions which go significantly beyond those previously tested for under the German fuel qualification testing programs. The conditions involved are for more demanding and go even beyond the design conditions expected for a modern modular pebble bed reactor. The EC tests are intended to establish a benchmark and validate the performance of the German fuel under these demanding conditions (e.g., very high burnup). If successful, the qualification and proof program for PBMR production fuel would then have to be demonstrated as achieving these same performance capabilities for these PBMR conditions.

The fuel modeling is aimed at developing a deterministic calculation capability for particle behavior under irradiation. The fuel fabrication aspect is aimed at re-establishing know-how in the fabrication of fuel kernels and particle coating technology.

In China, the Institute for Nuclear Energy and Technology (INET) is currently conducting an HTGR fuel irradiation qualification testing program for the HTR-10. This testing is being performed on both CFPs and fuel elements that were produced for use in the HTR-10. The fuel is currently being irradiated in a materials test reactor. The fuel

elements will be irradiated to burnups of 30,000, 60,000 and 100,000 MWd/t. At each of these burnups, the fuel pebbles will be subject to a temperature increase to simulate design-basis accident temperature conditions. The irradiation testing is a license condition for initial power escalation and long term power operation of the HTR-10. Once the fuel qualification testing is completed, it is expected that the INET fuel testing program will enter the operational phase in which CFP fuel performance will be assessed on a large-scale as part of HTR-10 power operations.

The Massachusetts Institute of Technology (MIT) has established a high temperature pebble bed reactor research project for student research. One area of student research is improved CFP performance modeling. CFP modeling aspects being pursued include migration of fission products through coatings, and chemical attack (Pd) of SiC. Other areas of interest which could lead to research collaborations include calculation of temperature distributions inside pebbles, models to predict the mechanical behavior, including failure, of CFPs, finite element models of CFPs, and fracture mechanics based failure models to predict CFP failure probability.

The PBMR fuel design is intended to be the same as the FRG reference fuel design. PBMR fuel is also to be manufactured using feed materials, processes and equipment which are "equivalent" to those that were used to manufacture the FRG reference fuel. The expectation on the part of the PBMR design team is that the PBMR fuel can achieve the same quality, irradiation performance and accident performance as the FRG fuel. This expectation also extends to fuel performance under PBMR service conditions. Plans are under way to conduct fuel irradiation tests using German AVR archive fuel and subject it to operating conditions and accident conditions that are applicable to the PBMR design. These tests are intended to provide part of an empirical data base which demonstrates that the German fuel design made with the German fuel manufacturing process perform satisfactorily in conditions simulating a PBMR operating conditions and postulated accident conditions and to establish a fuel performance benchmark for PBMR fuel that will be produced in the future at a PBMR fuel fabrication facility. In this regard, plans are currently being implemented to develop and establish the process, equipment and production facilities to be used to manufacture the production fuel for the PBMR demonstration plant and initial commercial PBMR plants. It is not expected that fuel from manufacturing facility will be available for irradiation testing until the first quarter of CY 2005.

VI.2.3.2 Purpose

The regulatory research plan in the area of HTGR fuel performance analysis is to establish NRC's infrastructure of knowledge, data, and tools needed for the performance analysis of HTGR fuels with TRISO CFPs. This infrastructure is needed to support the staff's review of a PBMR or GT-MHR application. The plan for establishing the infrastructure will capitalizes on world-wide research that has been conducted in this area over the last 30 years.

In summary, research is needed to qualitatively and quantitatively understand the safety margins associated with fuel performance for normal operation, design-basis accidents potential accidents beyond the design-basis. The research plan focuses on the ability of the staff to achieve the requisite level of knowledge in the areas of HTGR fuel design, manufacture, operational performance and accident performance, necessary to independently and

authoritatively assess the applicant's technical and safety basis. Tools will need to be developed and validated to independently predict fuel performance (including CFP failure rates and fission product releases) during normal operation, design-basis accidents and potential severe accidents. Research will also be needed to support mechanistic predictions of the source term for normal operation, and time dependent source term applicable to the relatively slow-moving design-basis accidents and potential severe accidents.

VI.2.3.3 Objectives and Planned Activities

The overarching objective of the NRC research in the HTGR fuel performance and qualification arena is directed toward developing a sufficient technical basis for the NRC to effectively review and resolve the significant technical and regulatory issues in the area of performance and qualification of HTGR fuels utilizing CFPs. The specific objectives are as follows:

VI.2.3.3.1 NRC HTGR fuels (PBMR and GT-MHR) testing

The purpose of the testing would be to:

- Provide the data needed to verify an applicant's fuel performance and fission product release;
- Provide the data which explores the limits (i.e., margins) of fuel performance and fission product release for parameters which are important to the fuel safety margins such as fuel operating temperature, maximum fuel accident temperature, fuel oxidizing environment, fuel burnup, energy deposition and deposition rate in the fuel (due to reactivity accidents), etc;
- Provide the knowledge and insights needed to provide the basis for judging the acceptability of an applicant's fuel irradiation test program (e.g., test methods, QA program, data analysis methods), and
- Provide data for use in developing/validating NRC analytical models and methods.
-

VI.2.3.3.2 NRC fuel analytical model and methods development

The purpose would be to:

- Independently evaluate HTGR fuel behavior, including CFP failure, fission product release and margins of safety, and
- Evaluate the effects of variations in irradiation service conditions, and uncertainties (i.e., sensitivity studies).

VI.2.3.3.3 NRC Interactions and Cooperative Research with Other International Organizations on the Aspects of HTGR Fuel Fabrication Processes That Are Critical to Ensuring Fuel Performance

The Purpose would be to:

- To stay current on design, development, testing, operational experience and research work directed at identifying the critical aspects of fabrication HTGR fuels utilizing TRISO CFP, and

- Capitalize on previous experience.

VI.2.3.3.4 The NRC Should Participate in National and International Workshops and Meetings Having Substantive Technical Content Relating to HTGR Fuel Performance Analysis Research Issues Discussed in this Section

The purpose would be to leverage NRC resources in pursuit of developing the NRC's fuel performance analysis infrastructure development needs.

VI.2.3.3.5 HTGR Fuel Irradiation Testing Plan

VI.2.3.3.5.1 Issues

Virtually all of the past and ongoing worldwide irradiation testing research of HTGR fuel designs with TRISO CFPs involved accelerated irradiations in MTRs. Although there subsequently was significant large-scale operating experience with these fuels in plants such as the AVR in Germany, accident simulation tests (i.e., fuel heat-up test following irradiation) to qualify the fuel involved accelerated irradiations in MTRs. There is not a well-established and thorough understanding of the mechanics and properties (e.g., creep) of CFP behavior, failure and fission product release to conclude with certainty that fuel accident simulation tests following accelerated irradiations are conservative as compared to the rate of fuel irradiation in a power reactor. Accident simulation heatup tests either after realtime MTR fuel irradiations or after fuel irradiations in a power reactor would be required to resolve this issue.

Virtually all of the accident simulation tests for TRISO CFPs involved so called "ramp and hold" temperature increases. These typically consist of increasing fuel temperature at about 50°C/hr up to a set temperature (e.g., 1600°C, 1700°C or 1800°C) and then holding the fuel at the set temperature for several hundred hours while fission product release measurements are taken. The results of ramp-and-hold tests up to 1600°C, for qualified fuel, show that no additional CFP failures occur. However, in the Federal Republic of Germany, there was at least one test in which the temperature was controlled to closely simulate the predicted accident heat-up curve to 1600°C for a design basis reactor coolant pressure boundary failure. For this test, CFP failures were observed to occur. Additional post-irradiation accident simulation tests that closely simulate the predicted temperature curve for a design basis reactor coolant pressure boundary failure would be required to determine if the traditional ramp and hold test accident simulation approach is conservative with respect to establishing CFP failure rates for postulated accidents.
Among the most limiting events that could challenge HTGR CFP integrity are those involving large scale chemical attack such as air intrusion following a pipe large break in the reactor coolant pressure boundary (RCPB) and moisture intrusion for a postulated heat exchanger tube failure with the reactor helium pressure falling below the heat exchanger tube pressure. While there have been experiments on oxidation of unirradiated HTGR fuel in air and water at HTGR accident temperatures and

measurements of HTGR fuel oxidation due to air or moisture impurities in helium during fuel experimental irradiations, there are no known experiments on fully irradiated HTGR fuels that simulate the effects of large air or water ingress events. Additional post-irradiation accident simulation tests that closely simulate air or water intrusion events and take the fuel to the onset of CPF failures would be needed to fully assess the adverse effects of air and water corrosion on HTGR fuels and the margins to failure for such events.

Very limited testing has been conducted on fuels with TRISO CFPs to assess the capabilities and the margins to CFP failure for reactivity events involving a large energy deposition in the fuel over a very short time interval (<< 1 second). Some limited testing was conducted in Japan for a postulated control rod ejection accident in support of the HTTR licensing and was one of the limiting licensing basis events. Although the staff has been told that the PBMR design does not have a potential for such large rapid reactivity events, this may not be the case for the GT-MHR with control rods located in the central core (fueled) region. In order to fully understand the margins to failure for reactivity events, fuel irradiation experiments involving such reactivity insertion events would need to be conducted.

Only limited worldwide testing has been conducted on previously qualified FRG or US HTGR CFP fuel for conditions that went well beyond the maximum qualification operating temperature and maximum qualification fuel burnup. In order to fully understand the margins to CFP failure and fission product release for fuel operations beyond the maximum allowed operating temperature (e.g., 1250 °C for PBMR) and design fuel burnup limits (e.g., 80 GWd/t for PBMR) fuel experiments involving irradiation conditions beyond such limits would need to be conducted.

VI.2.2.3.5.2 Plans

It is assumed that PBMR and GT-MHR applicants/vendors will conduct all fuel testing necessary to support their license application. Such fuel testing would be expected to address all significant aspects of the licensing basis and should address: a sufficient range of parameters to cover uncertainties and variations; the plant-specific service conditions of the PBMR and GT-MHR (core maximum operating temperature, fuel design burnup, fast fluence, number of fuel passes through the core, daily load follow), include a sufficient quantity of fuel elements and CFPs to establish a sufficient statistical database; and cover the range of potential CFP failure mechanisms and performance factors (e.g., fission product release) applicable to or potentially applicable to the licensing basis. It is also expected that such testing will use fuel from fabricated by the fuel production facility, utilizing equipment, processes and methods that are identical to those that are to be used to fabricate the production fuel for the (GT-MHR or PBMR) fuel cores. However, some test objectives supporting these fuel irradiation test plans may use counterpart German or US archive fuel or pre-production fuel.
It is important that the NRC staff and contractors have expertise on the proper conduct of fuel irradiation experiments, including a thorough understanding of good testing practices as well as testing limitations and potential opportunities for oversights and omissions. Such knowledge and experience will provide the staff with a sound basis for judging the acceptability of the applicant's fuel irradiation program methods, quality assurance practices, etc.

The NRC HTGR fuel irradiation testing program plan has three elements. These are testing of unirradiated German archive pebble fuel fabricated for the AVR, testing of PBMR production fuel for the PBMR demonstration plant and initial PBMR plants that may be built in the US, and testing of GT-MHR production fuel compacts that will be used for the initial GT-MHR plants that may be built in the U.S. Table 1 at the end of this section summarizes the irradiation testing plan for German archive pebble fuel, Table 2 summarizes the testing plan need for PBMR production fuel, and Table 3 summarizes testing plan needs for GT-MHR production fuel compacts. These test plans will be implemented in connection with the following cooperative agreements, and any proposed testing will not duplicate but capitalize on testing performed by DOE.

The NRC will participate in a cooperative HTGR fuel test program with the Department of Energy (DOE). The NRC emphasis for this cooperative fuel testing program will be on understanding the safety margins, by exploring conditions that are well beyond the fuel design-basis conditions associated with normal operations and postulated accidents. It is expected that participation in the program will also provide: test data which can be used for developing and validating fuel performance analysis models, data that can be used to confirm applicant fuel performance analysis, and data on which it is based, increased staff knowledge of fuel testing for later application to the review of an applicant's fuel qualification program technical basis documents.

The NRC will participate in the European Commission (EC) research HTGR program project on HTGR fuel technology. The NRC will provide support for the irradiation experiments on German Archive fuel and GA compacts fabricated using a new manufacturing procure as well as the accident heat up simulations with fission product release measurements and post irradiation examinations. The NRC will also support the retrieve of data from past HTGR experiments with the aim of constructing a fuel database.

VI.2.3.3.6 HTGR Fuel Analytical Model and Methods Development

VI.2.3.3.6.1 Issues

The body of irradiation and accident simulation (heating) tests have enabled the development of analytical tools of HTGR fuel performance during reactor operating conditions and postulated accident conditions. These tools have endeavored to model the various particle failure mechanism that have been identified. These mechanisms include: internal over pressure and tensile stress failure of the SiC layer; chemical attack of the dense coating layers due to migration of the fuel kernel; thermal dissociation and failure of the SiC layer at very high particle temperature; chemical interaction of fission products with the SiC layer leading to SiC degradation and failure and; mechanical overstress of the SiC layer due to external loading on the particle layers. Models have been developed for each of these potential failure modes. These models have been used by fuel designer to help quantify margins and by safety analysts in calculating mechanistic source terms.

VI.2.3.3.6.2 Plan:

The NRC, as a first step, will conduct a search and review of: (1) ongoing research aimed at developing tools for performing mechanistic analyses of HTGR fuel performance and (2) existing HTGR fuel performance analysis models and methods tools. The NRC would plan to enter into a cooperative agreement with a university or a cooperative agreement with the European Union to develop and validate analytical tools for assessing CFP behavior and fuel element performance, including fission product release and CFP failure. The developed tool would be benchmarked against existing empirical CFP fuel performance data, other codes and the results of NRC and applicant/vendor fuel performance and qualification test data. A user guide will be developed for use of the analytical tool. Sensitivity calculations could then be conducted to assess the effects of variations and uncertainties in fuel characteristics and reactor core conditions which may not be fully simulated (e.g., daily load follow) in the fuel irradiation testing programs.

VI.2.3.3.7 HTGR Fuel Fabrication Process Expertise:

VI.2.3.3.7.1 Issues:

The manufacture of the fuel kernels for the CFPs involves a process of gel precipitation droplet formation process using uranyl nitrate, followed by droplet aging, washing, drying, calcining and sintering steps. The coatings for the CFPs involve a chemical vapor deposition (CVD) process. By its nature, the CVD process and fuel element manufacture involves distributions of the attributes of the CFP layers (e.g., density, anisotropy, thickness, microstructure, stoichiometry) and distribution of attributes of the fuel pebbles or fuel compacts. Therefore the quality and the performance of billions of CFPs that make up an PBMR or GT-MHR core are both very statistical in nature. Consistent, reliable and repeatable fabrication process steps and reliable QC characterization of finished products against product specifications based on reliable characterization techniques, statistical analysis and acceptance criteria are critical to making fuel that will consistently perform satisfactorily during normal operations and licensing basis events. History shows that there is a learning curve associated with making fuel of this type and that experience is important to achieving and sustaining consistently high fuel quality and a consistently high level of fuel performance. This learning curve can involve as much as 10 years or more for fuel production to achieve and sustained fuel quality and performance standards. In this regard, a significant period of time has elapsed since HTGR fuel with TRISO CFPs was last successfully manufactured in large quantities for the nuclear fuel supply of an operating HTGR. Over this time critical manufacturing know-how has declined and will need to be re-established by fuel suppliers. In parallel, the NRC will need to develop an adequately high level of understanding of the aspects of fuel manufacturing that are critical to achieving and sustaining consistently high quality manufactured fuel. This knowledge will be needed to independently review the critically important area of fuel manufacture and that the manufacturing process and facilities that will be developed have addressed the critical issues and has established the controls and the specifications that will ensure that the requisite levels of quality will be sustained over the life of the fuel supply of the HTGR plant.

VI.2.3.3.7.2 Plans:

The NRC should participate in the European Commission (EC) research program project on HTGR fuel technology. The fuel fabrication aspect of this project is aimed at re-establishing know-how in the fabrication of fuel kernels and particle coating technology. It is expected that this program will help identify the critical aspects and the necessary controls and tolerances for the fabrication of HTGR fuels needed to achieve consistently good fuel quality for consistently good fuel performance.

Table 1. German Archive Fuel Irradiation Tests

#	Irradiation Purpose	Burnup Increment (GWd/t)					Safety Test Δ	PIE
		0 to 25	25 to 50	50 to 75	75 to 100	100 to 125		
1	Archive Pebble						N/A	
2	Archive Pebble						N/A	
3	Design Max Fuel Temp+Ramp Hold	-----Accel-----	-------Accel------	------Accel--------	------Accel--------	Δ	1600°C Ramp Heatup	Y
4	Design Max Fuel Temp + Acc Temp	-----Accel-----	-------Accel------	------Accel--------	------Accel--------	Δ	1600°C Accid Simulation	Y
5	Design Max Fuel Temp+Ramp Hold	-----Accel------	-------Accel------	------Accel--------	------Accel--------	Δ	1800°C Ramp Heatup	Y
6	Design Max Fuel Temp+Real Time	-- Real-Time--	---Real-Time--	---- Real-Time---	--- Real-Time----	Δ	1800°C Ramp Heatup	Y
7	Design Max Fuel Temp+50° C	-----Accel------	------Accel-------	------Accel--------	------Accel--------	Δ	1800°C Ramp Heatup	Y
8	Design Max Fuel Temp+Air Ingress	-----Accel------	------Accel-------	------Accel--------	------Accel--------	Δ	1600°C+ Air Ingress	Y

Δ = Burnup at which the safety test is conducted.

All irradiation tests are at the upper bound on burnup with margin (i.e., ~100 MWd/t).

All irradiation tests are for the upper bound on temperature with margin and simulate a sawtooth temperature history.

Irradiations should involve a conservative fast fluence vs burnup history (fluence > max expected fluence vs BU line) for the plant.

Accel = the burnup rate is accelerated compared to the burnup rate expected in the core.

Real time = the burnup rate is about the average real time burnup rate expected for the core.

PIE = Post Irradiation Examination (e.g., leach-burn-leach, micrograph).

Air Ingress = simulates the worst case oxidation expected for the worst case air ingress event with margin.

Table 2. PBMR Production Fuel Irradiation Tests

#	Irradiation Purpose	Burnup Increment (GWd/t)					Safety Test Δ	PIE
		0 to 25	25 to 50	50 to 75	75 to 100	100 to 125		
1	Archive Pebble						N/A	
2	Archive Pebble						N/A	
3	Design Max Fuel Temp	-----Accel------	-------Accel------	-----Accel-------	-----Accel------	Δ	1800°C Ramp Heatup	Y
4	Design Max Fuel Temp+50° C	-----Accel------	-------Accel------	-----Accel-------	-----Accel------	Δ	1800°C Ramp Heatup	Y
5	Design Max Fuel Temp+20K BU	-----Accel------	-------Accel------	-------Accel------	-------Accel-----	-----Accel------Δ	1600°C Ramp Heatup	Y
6	Design Max Fuel Temp+Real Time	--- Real-Time--	--- Real-Time---	---Real-Time---	--- Real-Time--	Δ	1800°C Ramp Heatup	Y
7	Design Max Fuel Temp+Air Ingress	-----Accel------	-----Accel-------	-----Accel-------	-----Accel------	Δ	1600°C+ Air Ingress	Y
8	Design Max Fuel Temp +RIA	Δ					Reactivity Insertion	Y
9	Design Max Fuel Temp +RIA	-----Accel------	-------Accel------	Δ			Reactivity Insertion	Y
10	Design Max Fuel Temp +RIA	-----Accel------	-------Accel-----	-----Accel-------	-----Accel------	Δ	Reactivity Insertion	Y
11	Design Max Fuel Temp+Rmp Hold	-----Accel------	-------Accel------	-----Accel------	-----Accel------	Δ	1600°C Ramp Heatup	Y
12	Design Max Fuel Temp +Acc Temp	-----Accel------	-------Accel------	-----Accel-------	-----Accel------	Δ	1600°C Acc Simulation	Y

Δ = Burnup at which the safety test is conducted.
All irradiation tests are at the upper bound on burnup with margin (i.e., ~100 MWd/t).
All irradiation tests are for the upper bound on temperature with margin and simulate a sawtooth temperature history.
All Irradiations involve a fast fluence vs burnup which is conservative (fluence above the maximum expected fluence vs BU line) for the plant.
Irradiations should involve a conservative fast fluence vs burnup history (fluence > max expected fluence vs BU line) for the plant.
Accel = the burnup is accelerated compared to the burnup rate expected in the core.
Real time = the burnup rate is about the average real time burnup rate expected for the core.
Air Ingress = simulates the worst case oxidation expected for the worst case air ingress event with margin.
PIE = Post Irradiation Examination (e.g., leach-burn-leach, micrograph).
RIA = Reactivity insertion accident TBD; energy deposition spike TBD (temperature increase over delta time); RIA time history simulation includes later core and fuel heat-up profile to simulate longer term fuel heat-up (e.g., loss of helium cooling due to loss of forced circulation following a reactivity insertion pebble compaction).

Table 3. GT-MHR Production Fuel Irradiation Tests

#	Irradiation Purpose	Burnup Increment (GWd/t)					Safety Test Δ	
		0 to 25	25 to 50	50 to 75	75 to 100	100 to 125		
1	Archive Compact						N/A	
2	Archive Compact						N/A	
3	Design Max Fuel Temp	-----Accel------	-------Accel-----	-----Accel-------	-----Accel------	Δ	1800°C Ramp Heatup	
4	Design Max Fuel Temp+50° C	-----Accel------	-------Accel-------	-----Accel--------	-----Accel------	Δ	1800°C Ramp Heatup	
5	Design Max Fuel Temp+20K BU	-----Accel------	-------Accel-------	-------Accel-------	-------Accel-----	-------Accel-----Δ	1600°C Ramp Heatup	
6	Design Max Fuel Temp+Real Time	-- Real-Time--	--- Real-Time---	--- Real-Time---	-- Real-Time--	Δ	1800°C Ramp Heatup	
7	Design Max Fuel Temp+Air Ingress	-----Accel------	-------Accel-------	-----Accel--------	-----Accel------	Δ	1600°C+ Air Ingress	
8	Design Max Fuel Temp +RIA	Δ					Reactivity Insertion	
9	Design Max Fuel Temp +RIA	-----Accel------	-------Accel-------	Δ			Reactivity Insertion	
10	Design Max Fuel Temp +RIA	-----Accel------	-------Accel-------	-----Accel--------	-----Accel------	Δ	Reactivity Insertion	
11	Design Max Fuel Temp+Rmp Hold	-----Accel------	-------Accel-------	-----Accel--------	-----Accel------	Δ	1600°C Ramp Heatup	
12	Design Max Fuel Temp +Acc Temp	-----Accel------	-------Accel-------	-----Accel--------	-----Accel------	Δ	1600°C Acc Simulation	

Δ = Burnup at which the safety test is conducted.
All irradiation tests are at the upper bound on burnup with margin (i.e., ~100 MWd/t).
All irradiation tests are for the upper bound on temperature with margin and simulate a sawtooth temperature history.
All Irradiations involve a fast fluence vs burnup which is conservative (fluence above the maximum expected fluence vs BU line) for the plant.
Irradiations should involve a conservative fast fluence vs burnup history (fluence > max expected fluence vs BU line) for the plant.
Accel = the burnup is accelerated compared to the burnup rate expected in the core.
Real time = the burnup rate is about the average real time burnup rate expected for the core.
Air Ingress = simulates the worst case oxidation expected for the worst case air ingress event with margin.
PIE = Post Irradiation Examination (e.g., leach-burn-leach, micrograph).
RIA = Reactivity insertion accident TBD; energy deposition spike TBD (temperature increase over delta time); RIA time history simulation includes later core and fuel heat-up profile to simulate longer term fuel heat-up (e.g., loss of helium cooling due to loss of forced circulation following a reactivity insertion pebble compaction).

VI.2.4 Materials Analysis

VI.2.4.1 Background

A key research area important to safety is the behavior of metallic and graphite components with structural, barrier, and retention functions under normal and off-normal conditions expected in HTGRs. A sound technical basis must be available for evaluating expected lifetime and failure modes of reactor pressure vessel materials and components whose failure would result in loss of core geometry and/or an ingress of air, water, or steam into the pressure boundary. High temperature materials are required to maintain core geometry, adequate cooling of the core, access for reactivity control and shutdown systems and, in the case of the PBMR, a defueling route. This section emphasizes the need for research to establish a technical understanding of the metallic and graphite components under high temperature operating and accident conditions. Integrity of the pressure boundary and structural components is linked to nearly all other research areas and in fact determines the useful life of the plant. Information from the materials research area is needed for conducting probabilistic risk assessments (PRA). Since failure probability data for components of advanced reactors is not available from experience, the information will be developed from materials research on potential degradation processes and quantification of their progression. Evaluation of component service life, safety margins, and behavior under accident conditions is dependent on spatial and temporal variations as well as the constant values of inputs such as temperature, pressure, gas composition, fluence determined by reactor systems analysis, and fuels analysis. Outputs of the materials component analyses would include stable configuration of the core, available operating time, temperature, pressure, fluence, and gas impurity limits. Research areas such as fuel integrity, neutronics, and reactor system analysis will need to be integrated into this area of research.

The operating conditions, materials, and coolant environments used in ALWRs are not significantly different from those of conventional LWRs. Therefore, lessons learned from the design, materials choices, and environments of LWRs should be taken into account for ALWR applications. Because of the similarities in materials and environments, there is not a great need for new research in the materials area specifically for ALWRs. However, a large body of research data, from both the US and Japan, has shown a detrimental effect of the coolant environment in reducing the fatigue life of LWR components. Methods have been developed and are widely available in the literature (NRC NUREG reports and PVRC report) for taking into account the effects of the operating environment in the fatigue design of components. Although the ASME code has on-going activities to address the issue of the effects of the environment, it has not yet incorporated changes in its design rules and correlations. Therefore we should ensure that during design and review of ALWRs that the effects of the environment are appropriately accounted for in the fatigue design and evaluation of components. We should also continue to work with ASME to ensure that its rules for fatigue design of components are updated. In addition, two aspects of the HTGR and some ALWR designs raise the potential for the need for improved ISI program and for continuous monitoring. First, more components are enclosed in pressure vessels making access for inspection difficult. The second aspect is the longer operating cycles between scheduled, short-duration, refueling outages during which ISIs can take place. This brings up the need for evaluating effectiveness of the less frequent ISIs for timely detection of cracking and degradation of components and the potential for excessive growth of cracks before the next ISI.

VI.2.4.2 Purpose

The NRC staff needs to develop independent research capability in the high temperature materials area for HTGRS to develop the staff's technical expertise in order to evaluate and establish a technical basis for regulatory acceptability regarding the safety capacity of these advanced reactor designs. The advanced reactor designs are significantly different from LWRs, where the staff has experience, in terms of the materials used, such as high-temperature metals and graphite; higher coolant temperatures; a coolant that does not change phase; and different degradation mechanisms such as creep, and behavior of metallic and graphite components in this environment.

In HTGRs, graphite acts as a moderator and reflector as well as a major structural component that may provide channels for the fuel and coolant gas, channels for control and shutdown, and thermal and neutron shielding. Additionally, graphite components are employed as supports. Graphite also acts as a heat sink during reactor trip and transients. During reactor operation, many of the physical properties of graphite are significantly modified as a result of temperature, environment, and irradiation. Significant internal shrinkage, bowing, and stresses can develop which may cause component failure, and/or loss of core geometry. Additionally, when graphite is irradiated to very high radiation dose, ensuing swelling causes rapid reduction in strength, making the component lose its structural integrity. In the event of an accident causing air ingress, subsequent graphite oxidation causes further changes in its physical and mechanical properties.

Research had progressed through the 1980's on the high-temperature design (creep, fatigue) of metal components for the Liquid Metal Fast Breeder Reactor. This research formed the basis for some ASME code cases and requirements for the design of high temperature components. The NRC staff needs to review and evaluate this research and that which has progressed since the 1980s/1990s, in particular with respect to the temperatures, coolant environment and materials to determine applicability to current HTGR designs and develop its own capability.

The NRC staff needs to develop independent research capability in the materials area beyond the licensing basis to understand safety margins and failure points, and reduce uncertainties. To conduct independent probabilistic risk assessments of advanced reactors, the staff will need information on the probability of failure of various reactor components. Because of the lack of operating experience, this information will have to be developed analytically using probabilistic fracture mechanics. To do this, potential degradation mechanisms of metallic and graphite components need to be identified and progression of degradation quantified under the operating reactor conditions. Potential technical issues that need to be addressed are: (1) availability and applicability of national codes and standards for design and fabrication of metallic and graphite components for service in HTGR high temperature helium environments; (2) lack of appropriate data bases for calculating fatigue, creep, and creep-fatigue interaction lifetimes of components in high temperature applications; (3) the effects of impurities, including oxygen, in the high-temperature helium on degradation of components; (4) aging behavior of alloys during elevated temperature exposures; (5) sensitization of austenitic steels; (6) degradation by carburization, decarburization, and oxidation of metals in HTGRs; (7) issues related to inspection of HTGR and ALWR reactor components; (8) performance and degradation of graphite under high levels of irradiation; (9) lack of knowledge for prediction of irradiated graphite properties from as-received virgin graphite properties; (10) lack of data on

oxidation kinetics of reflector grade graphite, fuel pebble matrix graphite, and graphite dust; (11) applicability of graphite sleeve properties to large block graphite properties; and (12) lack of standards for nuclear grade graphite. Each of these potential technical issues is addressed in the following paragraphs. Another potential issue for the PBMR is the understanding and prediction of the mechanics of pebble flow including temperature effects on pebble friction and flow, mixing of fuel and graphite pebbles at the central reflector core, compaction, hang-up, bridging, etc. This issue is discussed in the section on Nuclear Analysis (V.C-2). The NRC staff needs to develop independent research capability for the high temperature behavior of materials in HTGRs beyond the licensing basis to reduce uncertainty, gain confidence and understanding of defense-in-depth.

VI.2.4.2.1 Description of Issues, Metallic Components

The availability and acceptability of national codes and standards for the design and fabrication of metallic components for service in HTGRs is a key issue. Background studies and activities for eventual development of codes and standards were conducted in the 1980's for application to the liquid metal breeder reactor. Of particular note is the work conducted by the Pressure Vessel Research Council (PVRC) in their preparation of several technical reports that provided the basis for development of high temperature design codes by the ASME. These reports give background and procedures for design of components to resist fatigue, creep and creep-fatigue failures. However, the effects of the helium environment, including impurities such as oxygen were not addressed. In addition, improved correlations for creep and creep-fatigue have been developed from research of the 1990s. These improvements are not included in the PVRC reports and the procedures need to be updated before they are included in National Codes and Standards.

Although methodologies could be assembled from existing knowledge for calculating fatigue, creep, and creep-fatigue lives of components in high temperature applications, appropriate data bases for fatigue, creep, etc. are needed for these calculations. Based on past experience and research, we have found that environmental effects play an important role in reducing fatigue lives and in enhancing degradation of materials. For example, small levels of impurities such as less than 1 part per million of oxygen in the high purity water coolant of LWRs can greatly decrease fatigue life and resistance to stress corrosion cracking of metallic components. These effects were not originally addressed in the ASME Code. For example, the design data for fatigue was obtained from materials tests in air. Because helium is inert, there has been a tendency to obtain design data in pure helium, in impure helium, but not all impurities included, or in air. The effects of all important impurities, such as oxygen, in helium need to be taken into account with respect to reductions in fatigue and creep life and such data and understanding need to be developed. Environmental effects on fatigue under ALWR operating conditions need to be addressed as well.

To address degradation and aging of metals in HTGRs, the effects of high-temperature helium with impurities including oxygen at levels present in HTGRs need to be evaluated with respect to stress corrosion crack initiation and growth rate, crevice corrosion crack initiation and growth rate, and cyclic crack growth rate. Low levels of impurities in high-temperature, high purity aqueous environments are known to cause these types of

degradation and to accelerate the crack growth rates. The potential exists for these phenomena to occur in a high-temperature helium environment with low levels of impurities.

Many alloys undergo solid state transformation and precipitation during elevated temperature exposures. These transformation reactions are known as aging and can lead to embrittlement of the alloy. Aging and embrittlement occurs, for example, in cast stainless steel components under temperatures and time conditions experienced in operating LWRs. At the operating temperatures of HTGRs, the reaction rates are much higher, (i.e., the aging and embrittlement would occur sooner). The different alloys and higher temperatures of HTGRs would indicate potentially different aging reactions and mechanisms, some of which could occur relatively rapidly and render the material embrittled and susceptible to cracking. The aging reactions, as a function of time and temperature, in the different alloys used in important components of HTGRs need to be studied to establish the potential for material property degradation and embrittlement during the lifetime of operating HTGRs.

Another solid state reaction that occurs in stainless steels (and austenitic alloys) is called sensitization. Sensitization is caused by the precipitation of chromium carbides at the grain boundaries of the stainless steel. This precipitation normally occurs during slow cooling of the metal through high temperatures such as when cooling from the high temperatures following welding. Formation of the carbides depletes the chromium from the grain boundary areas rendering the stainless steel susceptible to intergranular stress corrosion cracking (cracking along the grain boundaries) in oxidizing and impurity environments. A less well known method for producing sensitization is through low-temperature sensitization. This occurs over long periods of exposures to relatively low temperatures. Low-temperature sensitization in stainless steel has been studied under temperature conditions relevant to LWRs. Under these conditions, low-temperature sensitization would not occur in times less than 40 years. However, the sensitization rate is exponential with temperature, and at the higher operating temperatures of HTGRs, there is a real potential for sensitization during the lifetime of these plants thus rendering the stainless steel components susceptible to stress corrosion cracking.

Carburization, decarburization, and oxidation of metals in HTGRs are other phenomena that can lead to degradation caused by the operating gaseous and particulate environment. Carburization is a phenomenon where carbon either as a particulate or from carbon containing gases diffuses into steel to form a surface layer with high carbon content. This surface layer may be hard, brittle, and have higher strength than the substrate. Differences in strength and other physical properties between the surface layer and substrate may lead to high stresses in the surface layer when the component is under load. In addition, carbides may form in the high carbon surface layer of stainless steel leaving the matrix depleted of chromium and susceptible to stress corrosion cracking and oxidation. Cracking, stress corrosion cracking, and oxidation can more easily develop in the surface layer which could then propagate into the component. Decarburization is a process whereby carbon is depleted from the steel depending on the composition of the gaseous environment. Depletion of carbon results in a softer steel and in reduced fatigue and creep lives. The presence of oxygen results in the formation of scale and general corrosion of metallic components and more importantly it can oxidize the graphite and render metallic components susceptible to

stress corrosion cracking. To control the phenomena of carburization, decarburization and oxidation, a very careful control of the level of different impurities is required. Conditions that lead to avoidance of one of the above phenomena can lead to development of another. For example, to avoid carburization, some HTGRs might use slightly oxidizing conditions by addition of oxygen to the gas stream. However, this can lead to oxidation of graphite, general corrosion of metals and an increased susceptibility to stress corrosion cracking. Some research has been conducted to study the phenomena described above; however, NRC needs to conduct confirmatory research and better define the conditions under which the phenomena occur for important metallic components of HTGRs. In addition, much of the available research did not include oxygen in the gaseous environment. Since oxygen will be present in HTGRs at high enough levels that can affect the progression of the above phenomena and can reduce fatigue, creep lives, and resistance to stress corrosion cracking, oxygen needs to be included in new experimental studies.

VI.2.4.2.2 Description of Issues, ISI and Monitoring

There are a number of potential issues related to the inspection of some HTGR and ALWR reactor components. Because some of these reactors are designed to operate for long periods of time between scheduled short-duration shut-downs for maintenance, ISI intervals may be too long and the amount of inspection conducted too limited. Therefore, there is a need to evaluate the effectiveness of various ISI programs as a function of frequency of inspection and the number and types of components inspected. Additionally, many internal components are not easily accessible for inspection, and the impact of not inspecting these components needs to be assessed. An alternative to conducting periodic in-service inspections during reactor shut-downs is to conduct continuous on-line, nondestructive monitoring for structural integrity and leakage detection of the entire reactor or reactor components during operation. Techniques for continuous monitoring have been developed, validated and codified for use in LWRs. If ISIs of HTGRs and ALWRs cannot be conducted on a frequent enough basis and certain components cannot be inspected, then continuous monitoring may become necessary. The continuous monitoring techniques need to be evaluated and validated for the materials, environments, and degradation mechanisms of the HTGRs and ALWRs.

VI.2.4.2.3 Description of Issues, Graphite

To be able to effectively review the new HTGR designs, there is a need to conduct confirmatory research to establish an information base related to the long-term performance and behavior of nuclear-grade graphite under the temperatures, radiation, and environments expected during normal operating and accident conditions. Potential loss of strength and of resistance to fatigue and creep, shrinkage, swelling, cracking, and corrosion during operation could impact the performance and function of the graphite core structural elements, reflector (side and bottom), and moderator balls. Various graphite variables, including coke source, size, impurity, and structure; manufacturing processes; density; grain size; crystallite size and uniformity determine the virgin and irradiated properties of the graphite component.

Some irradiation studies have been conducted on older graphites that are no longer available due to loss of raw materials supply and/or manufacturers. In addition, limited results are available at high levels of irradiation exposure. Thus, two key issues are the lack of data on irradiated properties of current graphites, and the lack of data at higher doses of irradiation. As discussed earlier, the irradiated material properties are heavily dependent on the particular make-up of the graphite and the manufacturing process; therefore, at issue is whether the irradiated materials properties from the "old graphites" can be assumed to be the same as the "new graphites." Irradiation affects, and in many cases, degrades physical and mechanical properties of the graphite. Important properties that change with irradiation are thermal conductivity, strength, and dimensions. These changes have safety implications since they may degrade structural integrity, core geometry and cooling properties. Some of these changes are not linear with irradiation dose. Strength of graphite initially increases with irradiation dose, then, at higher levels, it begins to decrease. With respect to dimensional changes, graphite initially begins to shrink with increasing dose, then beyond turn-around, graphite begins to swell with increasing dose. During operation, thermal gradients and irradiation induced dimensional and strength changes result in significant component stresses, distortion, and bowing of components. These can lead to loss of structural integrity, loss of core geometry, and potential problems with insertion of control rods. At still higher doses, beyond turn-around, where the swelling is considerably greater than the original volume, graphite structures and fuel balls will start to disintegrate and experience total loss of integrity.

To evaluate the suitability of a particular graphite for HTGR application, irradiation property change data is needed in addition to the as-received virgin properties. Development of adequate irradiation data on graphite is difficult, expensive, and time consuming. Therefore, reactor designers/vendors propose to use radiation data from studies conducted on older graphites and attempt to use graphites produced in a similar manner. However, the virgin and irradiated graphite properties depend strongly on the raw materials and manufacturing processes. Small variations in these may have strong effects on the graphite properties. Since the exact raw materials and processes have changed and may continue to change in the future, the NRC may need to independently confirm whether a particular graphite will behave the same as the old graphites under operating irradiation conditions. To accomplish this without irradiation testing every time a change occurs in the graphite raw materials or processing, correlations are needed for predicting irradiated graphite properties and changes from the virgin graphite raw materials characteristics, composition, processing, and properties.

Graphite corrosion and oxidation can occur in HTGRs from oxidizing impurities in the helium coolant from in-leakage during normal operation or from air or water ingress during accidents. The oxidation of graphite is an exothermic reaction, and it is important to know the rate of heat generation particularly during accidents. Oxidation also will remove the surface layers of graphite components resulting in loss of structural integrity. Further, oxidation will change the thermal conductivity and reduce the fracture toughness and strength of graphite components. The loss in strength may be due to intergranular attack of the binder. The oxidation rates vary for different graphites, and can be greatly affected by the impurities in the original graphite. Therefore, oxidation rate data is needed for the graphites proposed for new reactors.

The PBMR will use AGR type fuel sleeve graphite for the replaceable and permanent structures in the core. The proposed graphite properties used for design, operating, and accident analyses of these structures will have the same values as those for the sleeves. The sleeves are relatively thin structures manufactured differently than the large structural blocks of the PBMR, and the mechanical and other properties will be different. Furthermore, the properties of the large block graphite will vary through the thickness of the block. The difference in properties between the sleeves and large blocks and through-thickness variations need to be established. The potential for different irradiated properties of sleeve graphite and large block graphite also needs to be evaluated.

There is a lack of standards for nuclear grade graphite. Designers of HTGRs intend to use measured properties of the particular graphite in their design calculations. However, nuclear graphites should meet certain minimum requirements with respect to important properties such as strength, density, thermal conductivity, etc. as is the case for materials used in other reactor systems. If a particular graphite has excessively low strength and the designer uses that value in designing various components, that may not result in a suitable component for the intended service. There are underlying reasons why the strength may be excessively low. For example, the graphite might contain excessive cracking and porosity resulting in low strength. Although the component might have been designed using the low strength (resulting in possibly a thicker component), the excessive cracks in the component may grow during service and cause failure. Specific elements in the graphite might be detrimental to irradiation properties of the component, and they should be limited in nuclear graphites. Other elements , such as halides, which can degas during operation and cause degradation of other components in the reactor should also be limited in nuclear grade graphite. Thus, there is a need to develop standards on the acceptable physical and mechanical properties, composition, and manufacturing variables for nuclear grade graphite.

VI.2.4.3 Objectives and Planned Activities

The NRC research is aimed at developing an independent capability for NRC to evaluate the integrity of important components in advanced reactors under operating and accident conditions. Research on metallic components will be conducted to evaluate and quantify degradation processes, metallurgical aging and embrittlement, carburization, decarburization, nondestructive examination, and ISI. In addition, currently available (international) procedures for design against fatigue, creep, and creep-fatigue will be reviewed and evaluated. The objective of this review is to evaluate current code design rules and procedures and to provide input for improvements as necessary. The best procedures will be updated to incorporate correlations developed from more recent research. Research on graphite will be conducted to evaluate performance under high levels of irradiation, develop correlations for irradiated properties from virgin properties, develop data on oxidation kinetics, evaluate variation in properties through the thickness of large blocks, develop standards for nuclear grade graphite, and to develop an understanding of the mechanics of pebble flow. A description of this research for metallic components, ISI, and graphite components follows.

VI.2.4.3.1 Metallic Components

Carburization, decarburization, and oxidation of HTGR high-temperature metals will be studied as a function of time and temperature in helium gas with impurities including oxygen. Different levels and ratios of impurities will be studied. Metallographic studies and mechanical testing will be conducted on the exposed samples to determine the degree of deterioration and loss of strength. The objective is to define the environmental conditions under which the phenomena can occur, to what degree they occur under the different conditions, the potential for occurrence under the operating conditions of HTGRs, and the significance on structural integrity of components.

Research will be conducted on the effects of helium environment impurities, especially the effects of oxygen, temperature, and strain rate on the fatigue life of HTGR metallic components. Similarly, the effects of impure helium environments on the creep and creep-fatigue life of HTGR components will be investigated. The objective of this research is to ensure that the design rules and procedures available are adequately conservative and address reductions in life due to the operating environment. If the codes and procedures are not adequate, then the data base developed can be used to update the codes and procedures to provide adequately conservative design procedures and rules to avoid failure of HTGR components during service. In addition, research will be conducted to quantify the effects of carburization and decarburization on the reduction of fatigue and creep life to ensure that these reductions are adequately accounted for in the design procedures and analyses.

Research will be conducted on the effects of the high-temperature helium environment containing impurities, including oxygen, at levels typical of HTGRs on stress corrosion crack initiation and growth rates, crevice corrosion crack initiation and growth rate, and cyclic crack growth rate. The tests will be conducted on materials in the as received condition and in carburized and decarburized conditions. The objective of this research is to either confirm that these degradation mechanisms do not occur and crack growth rates are not enhanced in the environments of interest or to quantify the crack initiation times, increases in growth rates, and define the environmental conditions under which these occur.

Thermal aging and sensitization research will be conducted on high temperature alloys used in HTGRs on samples in the as-received and in the welded condition. Samples will be exposed for different times to temperatures at and above the operating temperatures of the HTGR components. Exposure to higher temperatures will provide an acceleration in the aging and sensitization reactions. As long as the aging mechanisms at the higher temperatures are the same as at the operating temperatures, correlations can be developed for quantifying the times required to reach different levels of aging and sensitization at the operating temperature. Mechanical property testing will be conducted on the aged samples to quantify the degree of embrittlement and other property changes as a function of aging time and temperature. Metallographic and microscopy studies will be conducted to identify the aging and precipitation reactions if they occur, to ensure that the reactions are the same at the operating and higher temperatures, and to evaluate the potential for and degree of low temperature sensitization. The objective of the research is to identify the potential and the degree to

which thermal aging, embrittlement, and sensitization can occur during operation of HTGRs and to evaluate the impact of these changes on the structural integrity of reactor components.

A number of potential degradation and aging mechanisms in the operating environment of HTGRs have been discussed. There is an opportunity to evaluate and validate these potential degradations by conducting research on components removed from operating reactors. An international research program will be conducted on components removed from the AVR to include microstructural studies and mechanical tests. Microstructural studies will be conducted to determine if solid state changes and precipitation have occurred during operation to produce thermal aging, sensitization, carburization, and decarburization. In addition, metallographic studies will establish if stress corrosion cracking, crevice corrosion, general corrosion, and oxidation have occurred. Mechanical tests on materials removed from the AVR will be conducted to determine if any degradation in materials properties has occurred. Fatigue and creep tests will determine if fatigue and/or creep damage have occurred, if the design codes and methods correctly predict the damage, and if the coolant environment had an effect in reducing fatigue and creep lives. The results will help determine if and how the design codes/procedures need to be changed to take into account the potential degradation mechanisms.

VI.2.4.3.2 ISI and Monitoring

In the nondestructive examination area, research will be conducted to evaluate the impact of different ISI plans on structural integrity and risk. The key variables in the study will be the length of time between inspections, the reliability of the inspection methods, and the number of components and locations tested for HTGRs and ALWRs. Different degradation mechanisms will be considered appropriate to the reactor design and operating environment along with the inspection variables in probabilistic fracture mechanics analyses to evaluate the impact of potential failures on risk. Results of this work will be used to support the evaluation of proposed ISIs of HTGRs and ALWRs, and to determine the technical basis for improved, more frequent , or more extensive ISIs. The results will also provide guidance on the need for continuous on-line monitoring of structural integrity.

Because some components are inaccessible and because ISI periods may be too long, research will be conducted to evaluate continuous monitoring of reactor components for crack initiation, crack growth, and for leak detection. Acoustic emission techniques will be used on laboratory testing of specimens under simulated HTGR and ALWR conditions (respective temperature, noise sources, coolant flow, etc.) in fatigue, creep, and stress corrosion cracking. Correlations will be developed for crack initiation and crack growth rates with the acoustic emission signals for the materials and environments of the HTGRs and ALWRs. Similar research was conducted by the NRC in the 1980s and 1990s where acoustic emission techniques were developed, validated, and codified for application to LWRs. The research, methods, and techniques for HTGRs and ALWRs will take advantage of the knowledge gained in earlier work. Similar acoustic emission techniques will be evaluated for detection, location, and quantification of coolant leakage from the pressure boundary and internal components

under the operating conditions of HTGRs and ALWRs. Again, similar work was conducted for LWR applications and the research for HTGRs and ALWRs will benefit from this. Once the laboratory research is completed and correlations of acoustic emissions to crack initiation and growth developed, an operating or test HTGR will be instrumented with acoustic emission sensors and monitored during its operation to validate the methods and correlations developed in laboratory testing. The result from this work will provide an alternative to periodic ISIs and the advantages of continuous on-line monitoring of reactor structural integrity and leakage. The results will also provide technical data bases for incorporating the techniques into codes and standards.

VI.2.4.3.3 Graphite

Research will be conducted to qualify graphite for HTGR application. This will involve evaluation of the performance and degradation of graphite under high levels of irradiation. A review will be conducted of available high dose irradiation data for nuclear grade graphite, including data from ORNL taken under the DOE NP-MHGTR program that has not been published. High dose irradiation data on "old graphites" will be evaluated to determine its applicability to "new graphites." The data will be utilized to determine the behavior of current graphites planned for HTGRs under operating conditions. In general, there is a lack of data in the high dose, high-temperature regime of HTGR operating environment, additional research will be conducted on current graphites planned for HTGRs to determine high dose material behavior, properties, and degradation. Experiments will be conducted at three different temperatures at high dose irradiation in a high flux test reactor. Microstructural evaluations such as microscopy and spectroscopy, dimensional measurements, mechanical testing, and physical property testing of the irradiated specimens will determine the effects of high dose and high temperature on new graphites.

Research will be performed to determine irradiated graphite properties from as-received virgin graphite properties. As received graphite material properties are determined by the raw materials and manufacturing process. Important parameters will be identified such as coke source, pitch, and sintering to develop graphites with carefully varied parameters within a range reasonable for HTGR graphite. Studies will be conducted to quantify the as received graphite. This will include mechanical properties such as strength, fracture toughness, density, thermal conductivity, level of chemical impurities, and absorption cross-section. Due to the anisotropy of manufactured graphite, the materials properties will be determined for three principle directions. The graphite will then be irradiated at systematically varied irradiation doses and temperatures significant to HTGRs. Following irradiation, the materials properties will be reevaluated to determine effect of irradiation and establish a correlation between the initial properties and the post-irradiation properties for any particular graphite that may be used in HTGRs.

Investigations will be undertaken to understand oxidation effects on the physical characteristics of nuclear graphite. There is a lack of data on oxidation kinetics of reflector grade graphite, fuel pebble matrix graphite, and graphite dust. Experiments will be conducted to determine reduction in weight of graphite due to oxidation indicating degradation and loss of mechanical integrity. The heat generated from oxidation of

graphite dust and the detrimental effect on surrounding components due to this elevated temperature will be studied. Research will be performed to determine the reduction in strength of graphite due to oxidation along binder paths through the bulk graphite which leads to diminished fracture, fatigue, and creep resistance.

Research on through-thickness variability in large block graphite will be conducted to characterize the key physical properties of full size blocks of current graphites planned for application in HTGRs (based on AGR fuel sleeve graphite), to establish in-block uniformity variation and variability between graphite batches. Large graphite blocks to be used for reflector material will be sectioned, tested, and evaluated to determine if sleeve properties can be extrapolated to large block. Due to the manufacturing process, graphite materials properties are typically anisotropic and vary with the forming method and size of the final fabricated component. The sectioned large block specimens will be tested to determine the important parameters such as strength, fracture toughness, density, thermal conductivity, level of chemical impurities, isotropy, and absorption cross-section. Based on the above results, an assessment will be conducted to estimate if the large block bulk properties would vary under high-temperature and high dose irradiation in a manner similar to the thin sleeve graphite material. This research will determine if the bulk large block material would exhibit the same behavior as the AGR fuel sleeve graphite.

Staff efforts will be directed toward development of consensus standards for nuclear-grade graphite. Design and fabrication standards are also needed. The NRC will work with the international community, industry organizations, and professional societies to develop a material specification consensus standard. The standard will set limits on important parameters for nuclear grade graphite planned for HTGR application. The standard will specify limits on density, strength, fracture toughness, thermal conductivity, coefficient of thermal expansion, absorption cross-section, impurities, and any other appropriate parameter. The staff will also work with the codes and standards organizations to develop the design and fabrication requirements for nuclear-grade graphite to address processes such as strength, fracture, fatigue, creep, irradiation damage, stability, and oxidation for HTGR service.

An effort will be conducted to review and evaluate experimental data, analyses, and appropriate models for predicting pebble flow through and across a PBMR reactor core. Evaluations will be conducted on how the predictive models were validated and how well they predict field experience. Pebble flow, temperature effects, friction, mixing of fuel and graphite pebbles in the central reflector core, compaction, hang-up, and bridging will be considered in the above evaluations. Conclusions will be reached regarding the application of currently available methods and codes, and recommendations developed for any necessary follow-on studies.

VI.2.4.4 International Agreements

There is considerable research that has been performed or is ongoing in the Commission of the European Communities (CEC) and Japan on high temperature metals for HTGRs. To make use of this research and establish cooperative research efforts, it is necessary to establish what research has been completed and what efforts are currently underway. Much of the current CEC research concentrates on advanced or future HTGRs that would operate at higher

temperatures and use different materials than the HTGRs of current interest. Regardless of this initial hurdle, there are some areas that were identified during the workshop on high-temperature gas-cooled reactor safety and research issues, October 2001, US NRC, Rockville, and during follow-up interactions that would be of potential use in accomplishing the NRC research objectives. The following areas, which are of interest to the NRC for review of current designs, should be explored with the CEC to determine if they can be addressed under on going HTR-related research projects in Euratom FP5[1] or whether they could be included in other or future CEC programs:

- The effects of impurities in the gas stream on creep behavior, fatigue behavior, creep/fatigue behavior, SCC, crevice corrosion, oxidation, and carburization/decarburization, of metal components in the temperature range of the HTGRs.
- Thermal aging and sensitization transformations of metallic components over the planned reactor lifetime at the temperature of interest for PBMR and GT-MHR.
- Microstructural analysis and mechanical testing evaluation of components removed from service, such as AVR, to determine the effectiveness of component lifetime design codes and standards, and information about effects of the operating environment on aging and degradation of components.

CEC efforts that may address these needs are their review of RPV materials, focusing on previous HTRs in order to set up a materials property database on design properties. Specific mechanical tests will be performed on RPV welded joints (Framatome facilities), and irradiated specimens (Petten HFR) covering tensile, creep and/or compact tension fracture. Compilation of existing data about materials for reactor internals having a high potential interest, selection of the most promising grades for further R&D efforts, and development and testing of available alloys will be part of the plan. Mechanical and creep tests will be performed at CEA on candidate materials at temperatures up to 1100° C with focus on the control rod cladding. The NRC needs to determine which aspects of this research are useful and then establish an agreement to make use of this information.

Other areas of international cooperation and experience that would be useful to the NRC staff involve how ISI can be performed to ensure safety and confidence considering component accessibility and the long cycle times of up to 6 years between refueling shut downs proposed for PBMR, GT-MHR, and ALWR plants.

The NRC staff has a need to establish the effects of irradiation on the predictability of graphite materials properties and establish whether the materials properties are within acceptable bounds. The CEC effort for potential cooperation with the NRC is currently reviewing the state of the art on graphite properties in order to set up a suitable database and perform oxidation tests at high temperatures on: (1) a fuel matrix graphite to obtain kinetic data for advanced oxidation (THERA facility at FZJ) and (2) advanced carbon-based materials to obtain oxidation resistance in steam and in air respectively (INDEX facility at FZJ).

Other work important to understanding high temperature materials in the Power Conversion System (PCS) could be accomplished through a cooperative effort with CEC. Current CEC

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[1]Georges VAN GOETHEM, EC, letter to Thomas L. King, US NRC, October 3, 2001

effort in this area is focusing on compilation of existing data about turbine disk and blade materials, selection of the most promising grades for further R&D efforts, and development and testing of available alloys. Tensile and creep tests (in air and vacuum) from 850° C up to 1300° C and fatigue testing at 1000° C will be performed at facilities at CEA while creep and creep/fatigue tests in helium will be performed at JRC. This cooperation would apply if the materials being considered by the CEC are applicable for use in HTGRs of interest in the US. Degradation of PCS components under high temperature environments leading to catastrophic failure could compromise the primary circuit pressure boundary in HTGRs leading to air and water ingress with commiserate degradation of core and safety components.

The UK is conducting ongoing research on graphite properties and has had experience with operating gas cooled reactors which may be useful for NRC cooperation. As part of international cooperation with the UK, the NRC plans to assign a staff member from RES to the NII in the UK to develop expertise on graphite behavior under high temperature and irradiated conditions and develop knowledge of experience with, and inspection of graphite in HTGRs. The NRC staff member would spend approximately 3 months in the UK to discuss with experts the reasons and causes for a lack of available correlations of "as-received" graphite properties with irradiated graphite properties. NRC staff work while on this assignment would include discussing, reviewing, and obtaining input from experts on the important manufacturing parameters, physical and mechanical properties, composition, etc. of the as-received graphite that should/could have an effect on irradiated graphite properties. With input from the UK (and other) experts, the staff would devise a matrix of tests/research plan for developing correlations between irradiated graphite properties from initial as-received properties. The NRC staff would also obtain details from UK experts of graphite operating experience and degradation, and details of UK inspection and monitoring programs.

Additional work for the NRC staff member during this international effort with the UK includes gaining a better understanding of ongoing and past research results at the University of Manchester and exploring potential cooperation in their program. In this effort, the staff would obtain information on the scope and objectives of NII's center of excellence for graphite research at the University of Manchester. The staff can obtain details from University of Manchester researchers on the graphite research being conducted for NII and other cooperating partners. The staff will then be able to evaluate potential benefits to the NRC of the research conducted at the University of Manchester, and explore different methods for NRC participation as appropriate.

The staff member will develop recommendations for the minimum acceptable values of parameters to be included in codes and standards regarding manufacturing and properties of graphite including design codes for structural analyses, and fatigue and creep analyses. This would be done in collaboration with NII and other experts to outline one or more potential standards for the manufacture, composition (i.e., limits on certain detrimental effects), and minimum properties for nuclear grade graphite. The NRC staff member would be in a position to obtain, review and discuss with NII and other experts different codes available for structural, fatigue, and creep analyses for design of high temperature graphite components. The staff will evaluate these codes and the need to update these codes, based on service experience and more recent research results produced after the codes were developed.

Finally, the NRC staff member will have the opportunity, with the help of NII staff, to gather data and information on the DRAGON experiments performed on graphite and fuels in the UK, and evaluate relevance of this information for application to currently proposed HTGRs.

Perhaps other international efforts, such as work in the UK where the issue has been raised, would be useful for determining the long term degradation mode of glass fiber encased insulation components which were discussed at the workshop on HTGR safety and research issues. The objective would be to conduct studies of the effects of vibrations and service conditions to determine the reliability of this insulation since it protects the metallic components and pressure boundaries in the HTGR designs from unacceptable high temperatures.

VI.2.4.5 Application Of Research Results

Research results will provide input on component probability of failure for NRC probabilistic risk analyses to independently confirm and support safety evaluations.

Due to the high temperatures and environments with which the industry has relatively little experience, careful analysis of the proposed materials needs to be carried out to indicate whether these materials are prone to degradation and provide the technical basis or criteria for materials acceptability. Aging effects and degradation due to the high temperature helium environment and radiation need to be considered. Evaluation of potential degradation mechanisms and rate of progression for materials used for connecting piping between the reactor pressure vessel and the power conversion systems will provide the NRC an independent basis to determine the validity of the contention that pipe break analysis does not need to be evaluated.

The research on nondestructive examination (NDE) and evaluations of ISI programs for HTGRs and ALWRs is applicable to independently confirm if an applicant's inspection plans are technically sound, or if additional requirements are needed. Currently accepted NDE and ISI programs may not detect materials degradation due to inaccessibility of components and long time periods between inspections. Research in this area may lead to regulatory requirements to modify NDE techniques and/or the use of continuous online monitoring of structural integrity for structures and components of advanced reactors.

VI.2.5 Structural Analysis

VI.2.5.1 Background

Historically, the NRC has been committed to the use of U.S. industry consensus standards for the structural analysis, design, construction, and licensing of commercial nuclear power facilities. The existing industry standards are based on the current class of light water reactors (LWRs) and as such may not adequately address analysis, design and construction features of the advanced light water cooled reactors (ALWRs) such as AP 1000 and International Reactor Innovative and Secure (IRIS) and other types of High Temperature Gas Cooled Reactors (HTRGs) such as Pebble Bed Modular Reactor (PBMR) and Gas Turbine Modular Helium Reactor (GTMHR). As part of its commitment to participate in the development of industry standards, the NRC plans to conduct research that will involve the review and study of the new and unique features of design basis documentation of the ALWRs and HTGRs.

The staff research effort will evaluate the containment, confinement, aging, material aspects, and challenge of external events for the PBMR, GTMHR, AP 1000, and IRIS reactor designs. Based on the findings of the proposed research plan, the staff will be able to determine the need to maintain current deterministic LWR requirements for containments, structures, systems and components or recommend that performance based and/or risk-informed criteria be used to evaluate the acceptability of proposed Advanced Reactor designs.

In 1996 and 1997, the NRC updated the seismic and geological criteria for siting nuclear power plants (NPPs). Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," is one of the new guides. It lists both the Lawrence Livermore National Laboratory (LLNL) and Electric Power Research Institute (EPRI) probabilistic seismic hazard methodologies as acceptable to the NRC staff for determining the safe shutdown earthquake (SSE) for NPPs. For the NPP sites in the central and eastern United States (US), the estimates from the two methodologies often differ by more than a factor of two. This has led to difficulties in cases where it was important to use the absolute value of the estimate. Research is needed to help ensure an efficient and effective advanced reactor licensing process.

In the proposed PBMR reactor vessel internal structure design, the ceramic reflector structure consists of graphite blocks with holes for control rods, and it is necessary to retain alignment through vertically arranged blocks, supported vertically by a dowel system, and circumferentially by a radial keying system. Research is needed on these structures since they are considerably taller than existing designs and consequently subject to nonlinear response during horizontal and vertical earthquakes.

Current soil-structure interaction computer codes are based on structures founded at or near the ground surface. Research is needed to evaluate the responses of new reactors that may be deeply or completely buried in ground.

In the new HTGRs, concrete structures may be subjected to sustained high temperature. Research is needed to accumulate and expand existing data on effects of high temperatures on properties of concrete. This data is available in various transactions and proceeding as well as in earlier research by Sandia National Laboratory (SNL).

In the mid 1990's, the use of structural modules was proposed for advanced nuclear power plants (AP 600, ABWR and System 80+). The objective in utilizing modular construction is to reduce the construction schedule, reduce construction costs, and improve the quality of construction. During the 1995-1997 time frame, NRC conducted research which evaluated the proposed use of modular construction for safety-related structures in the advanced nuclear power plant designs. The research program included a review of current modular construction technology, development of preliminary licensing review criteria for modular construction, and initial validation of currently available analytical techniques applied to concrete-filled steel structural modules proposed for the AP 600. The program findings were documented in NUREG/CR-6486, "Assessment of Modular Construction for Safety-Related Structures at Advanced Nuclear Power Plants." The key findings of this research were the need for supplementary review criteria to augment the Standard Review Plan and the need for verified design/analysis methodology for unique types of modules, such as the concrete-filled steel module.

Because of new reactors commitment to risk-informed processes, it is anticipated that existing ISI requirements for containment structure and structural components will be replaced or augmented by risk-informed ISI (RI-ISI) programs. Independent research is needed to work with the industry to develop methodologies for RI-ISI of containment and associated components such as liners, bellows, and prestressing hardware.

VI.2.5.2 Purpose

The purpose of this research activity is to develop the criteria for the evaluation of the structural/seismic analysis and design of the structures, systems and components of the new advanced reactors. The new reactor designs that deviate from current practice need to be reviewed to ensure that a level of safety equivalent to that of currently operating LWRs is provided, and that uncertainties in the design and performance are taken into account. For those unique features or areas that are not similar to existing operating nuclear reactors, the staff will need to conduct research to provide the technical basis for regulatory decision-making on these advanced reactor designs. Research is also needed to improve NRC's knowledge and understanding of new phenomena for which analytic methods and analyses are not currently available to the staff. The areas in which research should be conducted include: (1) seismic hazard methodology, (2) nonlinear seismic analysis of reactor vessel and core support structures, (3) seismic soil-structure interaction analysis of deeply embedded or buried structures, (4) effects of high temperature on properties of concrete, (5) issues related to modular construction, and (6) RI-ISI methodologies for containment and associated structures.

The ALWRs (AP 1000, IRIS) designs are upgrades, advancements, and simplifications to currently operational reactor designs. The majority of the advancements and simplifications are in the areas of systems, components and operations. These advancements include the use of passive safety systems, reduction in the number of components such as pumps, valves, and tanks, reduction in the amount of piping required, and the use of digital distributed control systems. The ALWRs structural design basis and the structural components, although in some cases different in appearance, are similar in nature to the existing domestic operating nuclear power plants. There have been attempts to enhance the structural analysis, design, fabrication, and construction criteria and processes including: (1) offsite prefabrication (called modular construction), (2) the elimination of the Operating Basis Earthquake as a design basis event, and (3) the use, in some cases, of more recent industry consensus and non-consensus codes and standards for Safety Class design and construction applications. However, not withstanding these features, the majority of the analysis, design, fabrication, construction criteria, and methods are similar to those applicable to recent domestic commercial nuclear power plants.

The unique design features of the HTGRs (PBMR, GTMHR) include the operational cycles such as helium gas cycles for heat and power generation and changes in the operational aspects of systems and components. In addition, in some cases, the safety classification and seismic categorization is based on probabilistic methods in lieu of the deterministic approach that has been used in current commercial power reactor designs. This approach results in power reactor designs which do not have "containments" designed to ASME, Section III, Division 1 and/or Division 2 (American Concrete Institute-359) as currently utilized in domestic operating nuclear power plants. While these reactors provide some structural design and construction processes similar to the Advanced LWRs reactors and to the existing operating

nuclear power plants, there are some unique structural design aspects that need to be evaluated.

In the area of probabilistic seismic hazard assessment, research will be conducted to update the current two seismic hazard assessments (LLNL and EPRI) for the central and eastern US making use of a set of guidelines developed by the NRC and DOE with EPRI, also called the Senior Seismic Hazard Analysis Committee (SSHAC) methodology.

Current Soil Structure interaction (SSI) analysis techniques and criteria used in the industry have been based on structures which only have partially embedded foundations. Analytical and experimental research will be conducted to develop independent capability for SSI analysis of completely deeply embedded or buried structures.

A key area of analytical and experimental research for PBMR is the nonlinear structural behavior of the reactor vessel and internals including its core and supports during horizontal and vertical seismic events. There is also a need to assess high contact point stresses between the spherical fuel pebbles due to dead weight as well as due to seismic events.

For concrete performance under high temperatures, research will be conducted to focus on accumulating the existing database, expanding the database, and evaluating the impact of high temperature on concrete properties.

The purpose of research in modular construction technology is to augment the earlier research performed by NRC and documented in NUREG/CR-6486, "Assessment of Modular Construction for Safety-Related Structures at Advanced Nuclear Power Plants." The key findings of this research were the need for supplementary review criteria to augment the Standard Review Plan and the need for verified design/analysis methodology for unique types of modules, such as the concrete-filled steel module.

Research will be conducted to develop methodologies for RI-ISI of containment and associated components such as liners, bellows, and prestressing hardware. This research will be built upon recent experience with applying the RI-ISI methodologies to piping. Components of this research include compiling database on degradation mechanisms for containment structures, developing methodologies for identifying risk-significant locations, identifying inspection techniques suitable for specific degradation mechanisms, and investigating methodologies for extending inspection intervals.

VI.2.5.3 Objectives And Planned Activities

The overall objective of this research is to assess new advanced reactor design concepts, and investigate the margins of safety in structures, systems, and components to support regulatory decisions that may be necessary in the design review phase. Industry codes and standards will be reviewed and evaluated to determine their applicability to the proposed reactor designs. This objective also includes investigating state-of-the-art analytical techniques in order to develop regulatory guides and regulatory criteria to reflect the latest knowledge and to confirm the licensing decisions made during the design reviews. The plan to carry out this overall objective is based on the following overall research objectives:

VI.2.5.3.1 Seismic Hazard Assessment:

The objective of this research activity is to update the two current seismic hazard assessments for the central and eastern US making use of a set of guidelines developed by the NRC and DOE with EPRI, also called the SSHAC methodology. With a single update methodology accepted by the NRC, the controversy associated with picking between the current two methodologies, developed by LLNL and the EPRI, will be reduced, if not eliminated.

The planned activity, implementation of the SSHAC methodology, is to be carried out, primarily, by the NRC making use of panels of seismicity and ground motion experts. The NRC staff, with contracted assistance, will (a) assemble the expert panels, (b) elicit from them the basic seismic hazard data, (c) compute the individual seismic hazard assessments for individual sites, (d) analyze and interpret the results, and (e) be the experts in the methodology and its use for licensing proposed advanced reactor designs.

VI.2.5.3.2 Seismic Analysis of Reactor Vessel and Core Support Structures:

The NRC research is aimed at developing an independent capability to evaluate the seismic integrity of the unique and new design features of advanced reactors. Due to the nonlinear configuration of the PBMR reactor components consisting of nonductile graphite core reflectors and supports, research will be conducted to develop seismic and structural analysis models of reactor vessel internals and core support structures and perform seismic analyses for horizontal and vertical earthquakes. The assumptions and limitations of existing finite element analysis codes will be evaluated for applicability to the PBMR design configuration. Due to the first-of-a kind design of PBMR internals, the need to perform experimental verification of the design's seismic response will also be investigated.

For the PBMR reactor, fuel pebbles will be piled into a considerably tall configuration resulting in nonlinear responses during horizontal and vertical components of earthquakes. Research will be conducted to perform linear and nonlinear elastic and plastic stress analyses due to the dead weight and seismic events taking into account contact stresses between the spherical pebbles of the tall piles of fuel pebbles.

VI.2.5.3.3 Seismic Soil-Structure Interaction Analysis

The objective of this research is to investigate the applicability of existing seismic SSI computer codes to deeply embedded or buried structures and to modify the computer codes as necessary. For two of the new reactor designs, the entire reactor building and a significant portion of the steam generator building will be partially or completely embedded below grade. For the analysis of seismic events, the SSI effects for these types of deeply embedded structures will have a significant influence on the analytically predicted seismic response.

Current seismic SSI analysis computer codes have been developed for and applied to coupled soil-structure models where the structures are founded at or near the ground

surface with shallow embedments. These computer codes have been developed to determine the seismic responses such as amplified response spectra, forces, and moments, that are required for the detailed analysis and design of structures, equipment and piping, taking into account the interaction between the soil and the structure during seismic events. These computer codes will need to be modified for applicability to deeply embedded structures. It is likely that kinematic (vertical and horizontal motion of the structure) interaction effects are more important for deeply embedded structures during seismic events than for conventional plants. It is also likely that dynamic soil pressures on deeply embedded structures will be more important and may require better definitions than are now available.

This research will focus on developing independent analytical capability to determine the coupled seismic SSI responses for deeply or completely buried structures during horizontal and vertical earthquakes. The research will also include shake table studies for the experimental verification of analytical results.

VI.2.5.3.4 Effect of High Temperature on Concrete

The objective of this research is to investigate the change in concrete properties when subjected to sustained high temperatures. In the current American Concrete Institute (ACI) Code, the temperature limits specified for concrete are 150°F for long term, 200°F for short term, and 300°F for local effects.

The operating temperatures of the primary reactor vessels for some of the new advanced reactors designs being considered are greater than those of currently licensed nuclear power reactors. Therefore, depending on the effectiveness of the reactor vessel insulation and cooling system, the concrete reactor building could experience a high temperature environment. Elevated temperatures can reduce the strength of concrete due to de-watering effects as well as cause degradations such as cracking and spalling.

This research will include data accumulation and expansion of existing data bases. Significant information regarding high temperature effects is available in literature, including journals, and conference transactions, and proceeding. SNL's earlier research on LWR severe accidents work also accumulated significant data on the effects of high temperatures on properties of concrete. Lessons learned from facilities where concrete was found to be subjected to high temperatures for long durations will also be investigated and utilized.

VI.2.5.3.5 Modular Construction

Modular construction has not been used in the USA for nuclear power plants but some techniques have been used in Japan. It has been proposed by the PBMR, GTMHR, AP 1000 and the IRIS to use modular techniques in structural elements inside the containment which must survive seismic loading events. Technical issues relate to the strength and ductility of joints and connections as well as appropriate damping values for seismic analyses.

This research effort will focus on developing evaluation criteria that will facilitate review of reactors that use modular construction. The NRC staff will use the results of earlier research described in NUREG/CR-6486. Also calculation methods will be verified based in part on available test data on structural module such as concrete-filled steel modules. Recommendations on the acceptability of industry codes (ACI 349, "Nuclear Safety Related Concrete Structures," and AISC, N690, "Nuclear Facilities-Steel Safety Related Structures-Design Fabrication and Erection") and required code changes will be made. Regulatory guidance will be established or revised as necessary to reflect the state of the knowledge.

VI.2.5.3.6 Aging and Inservice Inspection of Structures

Because of new reactors commitment to risk-informed processes, it is anticipated that existing ISI requirements for containment structure and structural components will be replaced or augmented by risk-informed inservice inspection (RI-ISI) programs. Research will be conducted to develop RI-ISI methodologies for ISI of containment and associated components such as liners, bellows, and prestressing hardware. Recent experience with the application of RI-ISI methodologies to ISI of piping has concluded that inspection resources need to be focused on risk-significant areas and inspection methods should be tailored to the potential degradation mechanisms. Existing inspection requirements have been found to be excessive and not focused on locations where cracks and leaks have been discovered.

American Society of Mechanical Engineers (ASME) has formulated a Task Group to develop methodologies for RI-ISI of containments. The staff will actively participate in this Code activity while independently developing the methodologies for RI-ISI of containments. Research for this item will include compiling data on degradation mechanisms for structures, developing appropriate inspection strategies for these degradation mechanisms, and defining risk categories based on potential degradation mechanisms and consequences of failure. ISI parameters such as the amount of inspection and frequency of inspection will be based on the risk categorization of the structural component. It is expected that the RI-ISI approach will result in focusing inspections on risk-significant areas while reducing unnecessary regulatory burden.

VI.2.5.4 International Agreements

The Japanese nuclear industry has made use of modular construction techniques and has traditionally invested a great deal of resources in testing to demonstrate the design's capabilities. To make use of this research and establish cooperative research efforts it is necessary to establish what research has been completed and what efforts may be underway. In 1997 the NRC staff published NUREG/CR-6486, "Assessment of Modular Construction for Safety-Related Structures at Advanced Nuclear Power Plants," which discusses some of the Japanese test results and efforts at that time. One of the recommendations of NUREG/CR-6486 was that a "cooperative program be developed to share information...which would provide valuable data useful in verifying the safe application of structural modules in nuclear power plants within the United States."

VI.2.5.5 Application Of Research Results

The end product of this work will be guidance in a NUREG for each task and updates of RGs and SRPs, as necessary. In addition, the completion of Task 1 will result in a revision of Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Ground Motion." A probable outcome will be that the probabilistic hazard estimates from the implementation of the SSHAC guidance and associated methodology will replace the LLNL and EPRI methodologies and provide an acceptable method for satisfying the 10 CFR Part 100.23 requirement for uncertainty analysis of the SSE determination. Possible outcomes of Tasks 2 and 3 will be new or revised computer codes that may be utilized by the staff for the review of new reactors submittals. The results of the efforts in Tasks 4 and 5 will result in staff interactions with the industry to help develop Code revisions to address effects of elevated temperatures on concrete and structural analysis and design methodologies for modular construction. In a manner similar to RI-ISI of piping, the research on RI-ISI of containments will lead to regulatory guidance for RI-ISI of containments and staff input for developing appropriate Code Cases.

VI.2.6 Consequence Analysis

VI.2.6.1 Background

Off-site consequence analysis is the final aspect of Probabilistic Risk Analysis, the so-called Level 3. The mix of radionuclides and the chemical forms in the releases from severe accidents occurring in advanced reactors may be different from those in releases during accidents in light water reactors. Therefore, comparisons of present and advanced technologies are likely to require the comparison of full Level 3 analyses. Past evaluations of light water reactor technology issues have often stopped at the stage of Large Early Release Frequency.

VI.2.6.2 Purpose

Normal input to NRC's Level 3 evaluation code, MACCS2 (MELCOR Accident Consequence Code System), is based on light water reactor technology. A review appears warranted to ensure that any important differences in user inputs to the code stemming from advanced reactor technologies are accounted for. The outcome of this effort will be an NRC choice of site- and technology-specific input parameters for the Level 3 analysis.

VI.2.6.3 Objectives And Planned Activities

There are 87 parent and daughter radionuclides presently considered in MACCS. The impact on off-site consequences in terms of early and latent fatalities, doses to specific organs, and economic consequences of these radionuclides is dependent on their chemical forms. The chemical forms are accounted for in dose conversion factors and other factors such as uptake in foodstuffs. If new biologically-important radionuclides are produced, they will be added to the library. If new chemical forms are important, revised dose and uptake factors will be made available. Other analyses will give a final list of radionuclides produced, but this research will evaluate the biological importance. In similar manner, the Level 2 analyses will give the chemical form of the released material, but this research will evaluate the needed factors.

VI.2.6.4 Application Of Research Results

The results will be incorporated into NRC's Level 3 code, MACCS2. Independent confirmation of risk (probability times consequence) will be available to NRC reviewers. For instance, a technical justification for a recommendation to the Commission on the policy question of the size of the Emergency Planning Zone (EPZ) may be needed. The supporting calculations will need to commensurate with the calculations utilized choosing a 10-mile EPZ for present light water reactor plants. These calculations are referred to in NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," where the choice of the size of the EPZ is discussed. The calculations are discussed more fully in NUREG-0396 (EPA 520/1-78-016), "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants."

VI.3 MATERIALS SAFETY AND WASTE SAFETY

VI.3.1 Nuclear Analysis For Materials Safety And Waste Safety: Criticality Safety, Radionuclide Inventories, Decay Heat, Radiation Sources, Shielding, and Detection

VI.3.1.1 Background

The term "nuclear analysis" refers to all analyses that address the interactions of nuclear radiation with matter. Nuclear analysis thus encompasses, for example, the analysis of: (a) fission reactor neutronics, both static and dynamic, (b) nuclide generation and depletion as applied to reactor neutronics and to the prediction of decay heat generation, fixed radiation sources, radionuclide inventories potentially available for release, (c) radiation transport and attenuation as applied to the evaluation of material damage fluence, material dosimetry, material activation, radiation protection, and radiation detection, and (d) nuclear criticality safety, (i.e., the prevention and mitigation of critical fission chain reactions ($k_{eff} \geq 1$) outside reactors).

This section of the advanced reactors research plan addresses nuclear analysis issues encountered in the NRC arenas of nuclear material safety and waste safety. Nuclear analysis research for the reactor safety arena is discussed in another section of this document.

While nuclear analysis is by no means the only technical discipline of importance to the regulation of material safety, and waste safety, it is a quintessential and cross-cutting discipline that appears repeatedly in regulated activities at the front and back ends of the respective advanced reactor fuel cycles. The nuclear analysis research issues and activities discussed in the following subsections are therefore cross-referenced, via footnotes, to other sections of the plan that address related technical areas and to sections that discuss multi-disciplinary research activities from the perspective of systems and processes (e.g., fuel enrichment, fabrication, transport, storage, and disposal).

VI.3.1.2 Purpose

The purpose of the research activities described in this section of the plan is to provide the nuclear analysis tools, data, and knowledge bases that will be needed in conducting the staff's out-of-reactor material safety evaluations throughout the fuel cycles of the respective advanced reactor designs. In identifying the necessary research efforts, the staff has first sought to identify the nuclear-analysis related issues that will arise in the technical evaluations of material and waste safety.

In the arenas of nuclear material and waste safety, nuclear analysis issues are expected to arise concerning (1) the out-of-reactor criticality safety analyses needed at the front end of the respective fuel cycles for the PBMR, GT-MHR, and IRIS designs, (2) the various safety analysis efforts that will be needed for at-reactor storage and away-from-reactor storage, transport, and disposal of the spent fuels to be discharged from PBMR, GT-MHR, and IRIS.

VI.3.1.2.1 Nuclear Criticality Safety at the Front End of the Fuel Cycle[2]

Enrichment plants, fuel fabrication facilities, and transportation packages for low-enriched uranium (LEU) commercial LWR fuel materials and fuel assemblies are not presently licensed to handle uranium enrichments significantly above 5 wt% ^{235}U. Criticality validation issues are expected to arise for HTGR materials safety due to the shortage of evaluated critical benchmark experiments involving neutron moderation by graphite, fuel materials with 5 to 20% ^{235}U enrichment, and particle fuel geometries. In addition, technical guidance may be needed on the criticality modeling of HTGR particle fuel forms, which are generally much more reactive than would be predicted by simplified computational models that smear the fuel particles and graphite into a homogeneous mixture.

Similar criticality safety analysis issues will arise for the higher-enrichment fuels (e.g., 8 wt% ^{235}U) produced for the IRIS reactor design, again because the enrichment plants, fuel fabrication facilities, and transportation packages now used for LWR fuels are not presently licensed to handle uranium enrichments above 5 wt% ^{235}U. Criticality validation issues are expected due to the shortage of applicable critical benchmark experiments involving materials with 5 to 20% enrichment and elements with high burnable poison loadings. Depending on details of the IRIS burnable poison designs, technical guidance may also be needed on the criticality modeling of fresh IRIS fuel elements in storage and transport in order to determine acceptable modeling approximations for granular or layered poisons.

VI.3.1.2.2 Safety Analyses for Spent Fuel Management[3]

Nuclear analysis issues for storing, shipping, and disposing of the high-burnup spent fuels and underburned fuels discharged from PBMR, GT-MHR, and IRIS will involve the assessment of modeling assumptions and approximations, needs for specific validation data, and validation uncertainty treatments in the prediction of (a) long-term decay heat sources for cooling, (b) radiation sources for shielding, and (c) spent-fuel reactivities (i.e., burnup credit) for criticality safety. Of course, technical safety issues for away-from-reactor management of spent fuel will generally have longer lead times for resolution than those for at-reactor handling and storage of irradiated fuels. It is anticipated that extensive burnup credit will be needed in performing criticality safety analyses for fuels discharged from PBMR, GT-MHR, and IRIS and that computational modeling and validation could become significant technical issues in this context.

VI.3.1.3 Objectives And Planned Activities

The NRC research objectives are to establish and qualify the independent nuclear analysis capabilities that are needed to support the evaluation of applicants' material safety and safeguards analyses for the fuel cycles of the respective advanced reactor designs.

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[2]See also separate Plan sections on Fuel Manufacture and Transportation & Storage.

[3]See also separate Plan sections on Transportation & Storage and Disposal.

VI.3.1.3.1 Planned NRC Research Activities

Listed below are planned research activities pertaining to the nuclear analysis issues anticipated in the assessments of nuclear material safety and safeguards for the respective advanced reactor fuel cycles.

VI.3.1.3.2 Nuclear Data Libraries

Preparation of Modern Cross-Section Libraries: (See reactor safety section on nuclear analysis)

VI.3.1.3.3 Nuclear Criticality Safety at the Front End of the Fuel Cycle:

Criticality Validation and Modeling Guidance for (a) PBMR, (b) GT-MHR, and (c) IRIS Fuel Materials: Identify and review existing and planned critical (and subcritical) benchmark experiments and use sensitivity methods to assess their applicability for validating criticality safety calculations involving fuel materials and fuel elements produced for the respective advanced reactor types. Develop options and recommendations for the evaluation and treatment of remaining validation uncertainties. Develop modeling guidance for PBMR and GT-MHR fuels to help ensure appropriate treatment of the resonance escape and self-shielding effects that make the particle fuel forms more reactive than would be predicted by simplified smeared models. Participate in cooperative programs for new experimental data as well as code-to-data and code-to-code benchmarking activities.

VI.3.1.3.4 Safety Analyses for Spent Fuel Management

Validation and Modeling Guidance for Applying Burnup Credit in Criticality Safety Evaluations involving Spent Fuel from (a) PBMR, (b) GT-MHR, and (c) IRIS: Identify and review existing and planned spent fuel isotopic assay databases as well as potentially relevant critical (and subcritical) benchmark experiments and use sensitivity methods to assess their applicability for code validation in applying burnup credit to criticality safety evaluations involving spent fuel from the respective advanced reactor types. Develop options and recommendations for the evaluation and treatment of remaining validation uncertainties. Develop modeling guidance for applying burnup credit to the respective fuel types to help ensure that accepted modeling approximations and assumptions will not lead to significant underpredictions of spent fuel reactivity. Participate in cooperative programs for new experimental data as well as code-to-data and code-to-code benchmarking activities.

Validation and Modeling Guidance on Predicting Decay Heat and Radiation Sources in Spent Fuel from (a) PBMR, (b) GT-MHR, and (c) IRIS: (See reactor safety section on nuclear analysis).

VI.3.1.4 Application Of Research Results

Results from the research activities described above will be applied to enable and support the staff's independent assessment of nuclear analysis issues associated with nuclear material

safety, waste safety, and safeguards in the respective advanced reactor fuel cycles. As outlined in the preceding sections, the nuclear analysis research activities will result in developing the staff's technical insights in these areas and applying those insights toward establishing independent review and analysis capabilities. The development activities include the assessment of validation issues and modeling approximations in order to inform the staff's evaluation and treatment of potential biases and uncertainties in the respective nuclear analysis areas. Especially important in this context is the development of state-of-the-art master cross section libraries as discussed section on Reactor Safety.

VI.3.2 Uranium Enrichment And Fuel Fabrication

VI.3.2.1 Background

The fuel elements for some types of advanced reactors will be substantially different in physical characteristics from those of existing light water reactor types. Therefore, new manufacturing facilities are likely to be required. Operating experience will provide valuable insights to ensure that those manufacturing facilities consider the accumulated knowledge from operating the existing facilities with a view toward minimization of hazards. Waste minimization and handling, criticality control, personnel exposure (ALARA), and contamination control are all candidates for the process. 10 CFR 20.1406 is the basis for this activity and the activity is consistent with the Commission's desire for risk-informed regulation.

VI.3.2.2 Purpose

Provide insights from activities at existing fuel manufacturing facilities in the areas mentioned above to identify safety issues and pathways to resolution.

VI.3.3.3 Objectives And Planned Activities

Reports to the NRC from the existing fuel manufacturing facilities will be surveyed and evaluated as a whole for insights into improvements that could be made. Further, the Independent Safety Analysis reports that will have been submitted by the fuel facilities will be reviewed for insights. In addition, the fabrication processes and materials for some advanced reactor fuel types (HTGR) may present a larger fire hazard than those in existing fuel fabrication facilities. Specific technical issues and research activities for criticality safety in facilities for enriching and fabricating the respective advanced reactor fuel materials and elements are identified and discussed in another section of this plan.

VI.3.3.4 Application Of Research Results

The reviewers responsible for the various aspects of the fuel manufacture, such as waste generation and handling, criticality control, ALARA, fire safety, and contamination control, will be provided with insights from existing facilities.

VI.3.3 Transportation And Storage

VI.3.3.1 Background

Regulatory requirements and technical guidance documents already exist for the packages and casks used in transporting fresh fuel and spent fuel under 10 CFR Part 71, for the at-reactor storage of fresh and irradiated fuel under Part 50, and for the storage of spent fuel in casks under Part 72. However, some advanced reactor fuels will differ substantially from existing LWR fuels both in physical form (for instance, pebbles versus rodded fuel bundles) and in enrichment (up to 20 wt% versus 5 wt%). Further, such technical issues as (a) the assessment of high-burnup (80 GWd/t) cladding integrity for IRIS spent fuels in storage and transport casks and (b) the application of burnup credit in the criticality safety evaluations for spent fuels from PBMR, GR-MHR, and IRIS[4] will take on significant new aspects in relation to the corresponding issues for conventional LWR fuels. Therefore, the continued applicability of existing requirements and technical guidance to the changed conditions may need review. Transportation and storage of spent fuel present issues of especially high public concern.

VI.3.3.2 Purpose

Evaluate the technical applicability of existing storage and transportation regulations and associated technical and regulatory guidance documents to new and existing package and cask designs for transporting and storing proposed advanced reactor fuels.

VI.3.3.3 Objectives And Planned Activities

A review of the data and analyses supporting existing storage and transportation regulations, and associated technical and regulatory guidance documents, will be undertaken to determine continued applicability for advanced reactor fuels. Physical differences between existing fuels and proposed fuels will be considered. If the existing data and analyses are found not to apply to proposed fuels, applicable data and analyses of similar types will be identified and provided where feasible. The review will identify any areas where changes or clarifications may be needed in the regulations and guidance documents. Certain aspects of this effort, including criticality safety evaluation with burnup credit, decay heat modeling, radiation shielding aspects of cask design, and the evaluation of radionuclide inventories available for release, will be addressed through the nuclear analysis efforts described elsewhere in this plan.[1]

VI.3.3.4 Application Of Research Results

Applicants and technical reviewers for the transportation and storage of proposed advanced reactor fuels will be given data and analyses to support the development and application of appropriate modifications to existing regulatory requirements and guidance.

VI.3.4 Waste Disposal

VI.3.4.1 Background

The NRC staff currently uses a risk informed and performance based approach to assess the disposal of high-level radioactive waste in a waste disposal repository to meet the design

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[4]See the Plan section on Nuclear Analysis for Material Safety, Waste Safety, and Safeguards.

objectives of 10 CFR Part 63. Basic knowledge limitations and conceptual, model, parameter and data uncertainties make it difficult to estimate the long-term dose and risk to the reasonably maximally exposed individual (RMEI) from the disposal of radioactive waste generated by advanced reactors. Where models are known to be oversimplifications of complex systems and uncertainties in these models are known to be large, the advanced reactor performance assessment dose and risk calculations could significantly underestimate or overestimate individual exposure. Underestimating the dose and risk to the RMEI could lead to decisions that realistic estimates would show to be inconsistent with Part 63 regulatory limits for the disposal of spent fuel from advanced reactors. In this case, opportunity and obligation exist to improve NRC's assessment capabilities. Over estimates of dose and risk to the RMEI from disposal of advanced reactor waste could cause unnecessary regulatory burden on stakeholders. Here, opportunity and obligation exist to improve the efficiency, effectiveness and realism of agency analyses and decisions.

VI.3.4.2 Purpose

The purpose of the advanced reactor waste disposal research is to provide more realistic data and information to support defensible estimates of radionuclide exposure to the RMEI from radionuclides released from a waste repository containing spent fuel and other radioactive waste from advanced reactors. Research is needed to quantify conceptual, parameter and data uncertainties in models used to estimate radionuclide source terms, transport of radionuclides in the environment, and transport of radionuclides through other biosphere pathways. Many computer codes use computational methods that attempt to compensate for uncertainty and lack of knowledge in a conservative manner with parameter and model selections that incorrectly predict potential exposure to 10,000 years. These conservative approaches generally lead to decisions that may be more restrictive than necessary and may incorrectly predict the locations and arrival times of radionuclides thereby overestimating the magnitude of potential radionuclide exposure to the RMEI.

Much of the data and information on fission products, transuranics and activated metals needed for establishing a technical basis and criteria for acceptability, are not available, or if available, are generally either of poor quality or have been obtained under conditions different from what could be expected in a high-level waste repository. The data are needed for establishing radionuclide inventories, determining source terms, understanding the chemical behavior of radionuclides in disposal environments, determining sorption parameters in the transport process, and evaluating pathways in advanced reactor performance assessment applications

Further research is needed to address uncertainties in performance assessment methodologies and computational tools applied to advanced-reactor wastes by updating existing computer codes where deficient, identifying analyses required for performance assessments, and validating computer calculations with experimental and field data derived from research investigations.

VI.3.4.3 Objectives And Planned Activities

The overall objectives of the advanced reactor waste disposal research program are to:
(1) Improve existing radionuclide source term, environmental transport and pathway computer codes for assessing the performance of a high-level radioactive waste repository containing

advanced reactor spent fuel, (2) Support the identification of long-lived radionuclides and their respective chemical forms in advanced reactor spent fuel, (3) Provide a technical basis for understanding the releases of radionuclides from spent fuel to the environment as a function of time to 10,000 years and peak dose from a repository containing advanced reactor high-level waste, (4) Validate analytical methods and all radiological, chemical and physical data used to predict radionuclide releases to and behavior in the environment against critical experiments in order to establish the calculational bias and uncertainty, (5) Obtain all laboratory and field data in probabilistic distribution format, (6) Quantify chemical effects that may impact the parameters that control radionuclide releases, mobility, solubility, sorption etc., (7) Identify appropriate environmental radionuclide migration pathways and model input for calculating plant uptake of radionuclides, (8) Quantify uncertainties of model calculations to predict dose and risk to 10,000 years, and (9) Evaluate direct radionuclide and fuel release by volcanism.

Certain important information will have to be provided by other areas of advanced reactor research:

1. Radionuclide inventories in spent advanced reactor fuel.
2. Potential for nuclear criticality in geologic disposal of advanced reactor fuel.
3. Chemical forms of radionuclides in spent advanced reactor fuel.
4. Fuel characteristics (e.g., microstructure, radionuclide distribution).

The first two of the above four items involve technical issues that will be addressed through the nuclear analysis research activities described elsewhere in this plan.[5] The other remaining issues will be addressed as part of the advanced reactor fuel program.

Other planned activities for this area of advanced reactor research include the following:

1. Obtain dissolution rates of advanced reactor fuel under varying chemical conditions.
2. Obtain radionuclide release rates from leaching experiments in varying chemical conditions.
3. Determine solubilities of important radionuclides released from advanced reactor fuel.
4. Obtain data on fuel cladding corrosion/dissolution under repository chemical conditions.
5. Evaluate repository near-field chemistry effects on spent fuel and cladding behavior
6. Determine presence of radiocolloids formed from cladding, material and repository particles.
7. Assess sorption characteristics of radionuclides in unsaturated and saturated groundwater.
8. Determine data to evaluate food-chain pathways impacts.
9. Determine direct fuel and radionuclide releases by volcanism.
10. Study accelerator transmutation of waste as an alternative to repository waste disposal.

VI.3.4.4 Application Of Research Results

Many results will be incorporated into NRC's high-level waste performance assessment computer codes. The research results are expected to be used to support evaluating and

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[5]See the Plan section on Nuclear Analysis for Material Safety, Waste Safety, and Safeguards.

auditing DOE's entire submittal, including data, information, models, computer codes, etc. The results are also expected to provide a base of physical data, information and scientific expertise that can be used by staff to quantify uncertainties in the technical bisis for supporting licensing reviews. In addition, the research results are needed to support the development of regulatory criteria and resolve NRC staff key technical issues associated with assessing a high-level waste repository containing advanced reactor waste.

VI.3.5 Personnel Exposure Control During Operation[6]

VI.3.5.1 Background

Since most of the facilities associated with advanced reactor concepts would be new facilities, the opportunity to design them from the beginning with attention to minimization of personnel exposure (ALARA) is unique. While most ALARA issues would not be new to advanced reactors, one unique issue has been identified for the Pebble Bed Modular Reactor and for the Gas Turbine Modular Helium Reactor: migration of the fission product silver from the grains of the fuel into the gas stream. ^{110m}Ag, with a 250-day half life, will present a continuing maintenance hazard as it plates out on down-stream equipment. Further, shielding designs for advanced reactors with graphite reflectors may develop streaming paths, posing a future exposure issue or vessel damage issue.

VI.3.5.2 Purpose

Ensure that the operational aspects of new reactor designs minimize personnel exposure. Systematically search new designs for different exposure issues, such as the ^{110m}Ag issue for the PBMR and GT-MHR and the issue of radiation streaming due to changes in graphite geometry.

VI.3.5.3 Objectives And Planned Activities

Evaluate the extent of the ^{110m}Ag hazard and plans for personnel exposure control. Evaluate the propensity for geometry changes in graphite components[7] and assess associated radiation streaming issues[8] in view of potential concerns over vessel fluence[9] as well as radiation protection. In addition, evaluate different advanced reactor designs to identify any other issues that may pose radiological hazards that differ from those in conventional LWRs.

VI.3.5.4 Application Of Research Results

Provide reviewers with insights from analyses.

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[6]Applies to Reactor Safety as well as Materials and Waste Safety

[7]See related activities described in the section on Nuclear-Grade Graphite.

[8]See related activities described in the section on Nuclear Analysis for Material Safety, Waste Safety, and Safeguards.

[9]See related activities described in the section on High-Temperature Materials.

VI.4 SAFEGUARDS

VI.4.1 Background

The focus of this section is on material control and accountability. The loss of two fuel rods at the Millstone plant following their separation from their large, well-identified fuel assemblies for example, has suggested that material control and accountability (MC&A) should be reviewed for certain of the advance reactor types. The fuel elements for PBMR and IRIS will be enriched up to 8 wt% and for GT-MHR up to 20 wt% ^{235}U. Therefore, these types of fuel elements may be more desirable for diversion than the less-enriched (3 to 5 wt%) fuel for conventional LWRs. Further, the fuel pebbles for the PBMR are relatively small in size (6 cm diameter), very large in number, and not individually marked with identifiers, thus making MC&A potentially more difficult. This research area addresses material diversion safeguards, including the nuclear analysis efforts needed for assessing proliferation potential and radiological threats, material security technology, and material control and accounting measures throughout the fuel cycles of the respective advanced reactor designs

VI.4.2 Purpose

Provide insights into potential MC&A activities that will provide safeguards for the enriched fuel material during manufacture (if the decision is made to manufacture in the U.S.), transport, storage on-site prior to irradiation, irradiation, and storage and transport as spent fuel. The safeguards activities should be commensurate with the relative ease and desirability of diverting the respective advanced reactor fuel types.

VI.4.3 Objectives And Planned Activities

Other industries produce valuable, seemingly-identical objects that are not specifically identified. Those industries can be surveyed to provide benchmarks for activities in MC&A for advanced reactor types. Literature surveys will be performed to develop a set of industries for the benchmarks. As part of the larger safeguards evaluation efforts, the relative ease and desirability of material diversion will be examined through nuclear analysis activities described elsewhere in this plan.[10] In addition, the technological barriers to extracting plutonium and other radionuclides from irradiated fuel materials will be described for the respective advanced reactor technologies.

VI.4.3.1 Material Diversion Safeguards

Nuclear analysis tools and methods will be used in the arena of material diversion safeguards for the assessment of weapons proliferation potential and radiological threats, material security technology, and the material control and accounting (MC&A) measures needed throughout the fuel cycles of the respective advanced reactor designs.

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[10]See the material safeguards issues and research activities described in the section on Nuclear Analysis for Material Safety, Waste Safety, and Safeguards.

For example, the PBMR's use of pebble fuel elements in a multiple-pass, continuous on-line fueling scheme will raise questions about the potential for overt or covert production and diversion of bred fissile plutonium and other radionuclides for use in nuclear weapons or radiation weapons. It is worth noting in this context that the higher burnup levels (e.g., 80 GWd/t) of spent fuel from a PBMR will yield plutonium isotopic compositions that are significantly less attractive for use in nuclear weapons than those in today's spent LWR fuels. Nevertheless, in view of the apparently greater ease of diverting 6-cm-diameter fuel pebbles (or 80-cm tall GT-MHR fuel blocks) in relation to 4-meter-long LWR fuel rods or assemblies, questions will arise about the potential for early discharge and diversion of standard fuel pebbles (i.e., with 4-8% initial ^{235}U enrichment), or of special plutonium-production pebbles fueled with natural uranium, and the predicted quantities and isotopic compositions of plutonium that could credibly be produced and diverted without noticeable disruption of operations or reliable detection under such postulated proliferation scenarios.

In addition to predicting plutonium production, various nuclear analysis methods (e.g., radiation shielding codes) will also be applied in modeling and assessing the performance of nuclear detection systems used in various MC&A and security settings for preventing and detecting the covert introduction or diversion of materials in fuel production, transport, reactor operations, and waste management.

No new nuclear analysis issues have been identified for assessing material diversion safeguards in the fuel cycle for AP-1000, whose fuel assemblies are essentially identical to those for convential PWRs. For IRIS, the only potential issues for material safeguards would be those concerning the presence of higher-enriched LEU materials at the front of its fuel cycle.

VI.4.3.1.1 Scoping Studies on Proliferation Resistance of (a) PBMR and (b) GT-MHR Fuel Cycles:

Analyze postulated scenarios for overt and covert production of weapons-usable plutonium in the respective fuel cycles. Develop credible postulated scenarios involving introduction, early discharge, and diversion of standard fuel elements as well as special Pu-production fuel elements. Perform calculations to predict associated radionuclide inventories, including the quantities and isotopic compositions of plutonium produced per fuel element. Using credible assumptions regarding specific material control and accounting and material security measures, compare the proliferation resistance of the PBMR and GT-MHR fuel cycles to that of the major reactor types in operation around the world today, including LWRs and CANDUs. The comparative analysis should consider the potential for using the respective reactor types for overt or covert production of materials for fission weapons as well as weapons that use chemical explosives or other means for dispersing radioactive materials.

VI.4.3.1.2 Assessment of Technical Requirements for Material Control and Accounting and Material Security in the (a) PBMR and (b) GT-MHR Fuel Cycles:

Using the material production results from the scoping studies described above (see previous item) and information on detector technology typically used in MC&A and security, assess the ability to detect the overt or covert diversion of significant quantities

of material, considering standard as well as special requirements for MC&A and material security technology. Compare the material diversion potential of the PBMR and GT-MHR fuel cycles to that of the major reactor types in operation around the world today, including LWRs and CANDUs. Develop recommendations and options regarding any special measures needed for reducing the diversion potential in the respective advanced reactor fuel cycles.

VI.4.4 Application Of Research Results

This research will provide reviewers with relevant MC&A benchmarks from other industries and will develop and analyze technical information needed for establishing a technical basis for new material safeguards and MC&A acceptance criteria in the proposed advanced reactor fuel cycles.

PART II

VII. PRIORITIZATION

As part of the overall objective to preparing the NRC for independent regulatory review of advanced reactor applications and to develop the associated regulatory infrastructure including data, codes and standard and analytical tools, a prioritization method is needed to help allocate available resources. The purpose of the advance reactor research program prioritization is to provide an effective method for allocating resource among the different elements in the research program, and takes into account the four performance goals used for the prioritization of research as a whole. Application within a particular technical area, a phenomena identification, and ranking table process will be used to focus resources on those tests and analysis that would contribute significantly to achieving, for example, the need for some projects to be completed on a particular schedule, the relative safety significance and the important of the research to the development of policy recommendations.

VII.1 PHENOMENA IDENTIFICATION AND RANKING TABLE (PIRT) PROCESS

RES has developed and used the PIRT (Phenomena Identification and Ranking Tables) process as a tool for identifying and prioritizing research needs. The PIRT process, and related approaches previously used by RES (e.g., CSAU=Code Scaling Applicability and Uncertainty), provide for the identification and ranking of safety-significant phenomena and associated research needs through the sequential consideration of:

→ 1. Designs
→ 2. Representative Scenarios
→ 3. Important Phenomena
→ 4. Important Data and Models
→ 5. Available Data and Models
→ 6. Gaps in Available Data and Models

For a given design (e.g., of a reactor system, fuel transport cask, storage facility, etc.), this kind of approach becomes risk-informed by employing PRA and/or other risk evaluation techniques (e.g., Hazops) to help guide and check the selection of representative scenarios or event sequences.

Such phenomena-based approaches to research planning and prioritization have been previously applied in the context of the four advanced reactor designs reviewed by RES during the early 1990s (MHTGR, PRISM, PIUS, and CANDU-3), with the goal of providing an initial comprehensive identification and assessment of significant gaps in the data and modeling needed for safety analysis of the respective reactor design. Results of those efforts were documented in several papers and reports, including for example the following:

(1) D.E. Carlson and R.O. Meyer, "Database and Modeling Assessments of the CANDU 3, PIUS, ALMR, and MHTGR Designs," paper presented at the 1993 WRSM.

(2) P.G. Kroeger, "Initial Assessment of the Data Base for Modeling of Modular High Temperature Gas-Cooled Reactors," Draft report (82 pages), Brookhaven National Lab, September 1993.

(3) D.E. Carlson and R.O. Meyer, NUREG-1502, "Assessment of Database and Modeling Capabilities for the CANDU-3 Design," 1994.

More recently, formalized PIRT processes have been conducted in which a panel of outside experts is tasked with considering a limited set of scenarios or associated safety-related phenomena in a given system. Recent examples include the PIRT processes conducted on (a) AP600 test and analysis need, (b) performance of high-burnup LWR fuels in reactor accidents, and (c) using burnup credit in predicting the subcritical margins for spent PWR fuel in shipping cask accidents.

Several PIRT activities will be conducted for each advanced reactor design or design type (e.g., HTGR). These activities are outlined and described below:

VII.1.1 Umbrella PIRT for Comprehensive Reactor Safety Evaluation

VII.1.1.1 Initial Strawman Umbrella PIRT

For each reactor design, a team of NRC staff and contractors, whose collective areas of expertise should largely cover the full range of anticipated processes and phenomena for that reactor design, will develop a draft PIRT document for high-level identification and prioritization of the specific data and model development activities that are needed to enable and support the staff's safety evaluation of that design. This PIRT team will consist of six to ten NRC staff and contractors or type (e.g., PRA, thermal and fluid flow, nuclear analysis, fuel fabrication and performance, fission product transport, materials, systems, structures, and components, containment/ confinement, human factors, I&C, maintenance and inspection). NRR will be invited to provide one or more technical staff to serve as team members and/or observers.

For the PBMR and GT-MHR, this umbrella PIRT activity will build upon results from (i) the October 2001 NRC Workshop on HTGR Safety and Research Issues, and (ii) relevant NRC preapplication review and research efforts conducted during the 1985-1995 time frame for the DOE MHTGR design, including Reference 2 above, an RES contractor's PIRT-like report on MHTGR safety evaluation.

Selected off-normal and accident event sequences will be chosen to represent the major safety-related processes and phenomena encountered in all anticipated licensing basis events (LBEs). The selected event sequences will initially encompass phenomena in the LBEs proposed by the preapplicant and will be supplemented as needed by additional or alternative sequences derived from the staff's framework activities, past NRC and international experience, and relevant PRA results as they become available from NRC and outside efforts. Accident sequences beyond the licensing basis will also be considered as needed for the NRC staff's assessment of safety margins, defense-in-depth, and the significance of uncertainties in the predicted frequencies and consequences of events. Normal operating conditions will be addressed as needed for establishing accident initial conditions, such as temperatures, pressures, flows, power densities, irradiated fuel characteristics, and properties and dimensions of irradiated materials.

Results from these initial umbrella PIRT activities will be considered in prioritizing, refining, and updating the remaining activities in the evolving research programs, including, as described

below, additional "topical" PIRT activities focused on particular subgroupings of phenomena, associated event sequences, and affected systems, structures, and components. With regard to prioritization, this umbrella PIRT activity will produce an initial identification and ranking of research efforts by their technical priority, with highest technical priority going to efforts that address the largest gaps in the most safety-significant data and analysis tools.

VII.1.1.2 Continuing Umbrella PIRT Activities

Results from the strawman umbrella PIRT activities for each design can be peer reviewed, leading to publication of a PIRT report. Any major additions or revisions emerging from the formal PIRT panel or peer review processes, or from the topical PIRT activities described below, will be reflected through appropriate additions or changes to the affected research activities and their relative priorities.

VII.1.2 PIRT Activities

Following and in some cases concurrent with the umbrella PIRT, NRC staff and contractors will conduct topical PIRT activities that focus on particular subgroupings of phenomena with their associated event sequences and affected systems, structures, and components.

Foremost among the NRC's topical PIRT efforts relevant to the PBMR and GT-MHR designs will be a PIRT activity focused on HTGR TRISO fuel performance (i.e., fission product retention and transport) as affected by fuel fabrication variables, irradiation parameters, and accident conditions such as power transients, loss-of-cooling heatup accidents, air ingress with oxidation, or moisture ingress with hydrolysis. This topical PIRT activity will be conducted in two phases, the first involving only NRC staff and contractors and running concurrently with the initial PBMR/GT-MHR umbrella PIRT exercise described above. The second phase will employ outside panel members in addition to the participants in the first phase and will incorporate relevant information from the initial umbrella PIRT activities.

As suggested by results from the umbrella PIRT exercises and other research efforts, additional topical PIRT efforts may be conducted to give closer attention to such areas as reactivity and power transients, graphite oxidation, passive decay heat removal, high-temperature materials, containment/ confinement performance issues, or human factors and I&C. To help conserve limited resources and meet schedules, such topical PIRT exercises will initially be limited to NRC staff and contractors. As warranted and possible within resource and schedule constraints, some of these less formal PIRT exercises may be followed in a second phase by formal PIRT panels or peer review processes.

Results from the topical PIRT activities will be combined with those from the umbrella PIRT exercises and reflected through appropriate refinements, additions, or changes to the affected research activities and their relative priorities.

VIII. IMPLEMENTATION

Successful implementation of an effective advanced reactor research infrastructure is essential to support an efficient licensing process. To achieve this, the NRC will have to consider the projected industry schedule as well as budget constraints. Tasks that would require sufficient lead-time (e.g., rulemaking, codes and standards development efforts) will have to be initiated well ahead of a formal license application. Other tasks that are technology-neutral or generic (e.g., regulatory framework) will have to be initiated and completed whether there be one or more license application. In this process, various research activities will be prioritized by addressing some basic questions. As discussed in Section VII, a systematic and logical PIRT process will be implemented to prioritize various research topics. Using these guidelines, the needed research activities can be ranked, available resources can be allocated, and schedules can be established.

Inevitably, to off-set costs, the NRC will have to continue to draw upon the existing international HTGR experience and research. Due consideration would have to be given to future cooperative efforts in both the domestic and the international arena. To alleviate the burden, some shared research with the industry is also expected. Most importantly, it is imperative that at the pre-application review phase, the applicant provide the NRC with complete and detailed plant-specific design-, safety-, and technology-related information so that at topical meetings NRC can raise specific issues. Early identification and resolution of key safety issues are essential to the efficient licensing of a plant design. Discussions between the NRC and the applicant during the pre-application review phase should lead to a clear understanding of mutual expectations. These discussions should also help identify the information gaps as well as the additional analytical tools and data that the NRC might need to develop to support the review of the applicant's submittal at the license application stage.

For implementation of an effective advanced reactor research infrastructure, the following critical elements need to be considered for each topical research area:

VIII.1 PRIORITY (e.g., high, medium, low)

- How important is it?
- What if the NRC does not develop the independent information?
- What are the implications if the desired information is not generated to the level desired or in the time frame required?

VIII.2 THE DESIRED END-PRODUCT (e.g., new or modified analytical code, experimental data)

- What independent analytical tool or experimental/operational data are needed?
- Is it generic (technology-neutral) or plant-specific?
- Can the applicant(s) be asked to share the cost of generating/developing the information?

- Can the applicant(s) be asked to provide part or all of the supporting data?
- Is the information available elsewhere, such as, from international partners or domestically funded (e.g, by DOE or US industry organizations)?
- What is the feasibility of a joint venture with the industry?
- Can the required information be purchased for reasonable cost or by making a contribution of the kind?
- Do we have the necessary performance/acceptance criteria for the final product? What levels of uncertainties would we accept?
- Would there be a need to do any sensitivity analysis?
- What means (e.g., experimental data, code-to-code validation) would we need for testing/validating/accepting the final product?

VIII.3 INTER-DEPENDENCE OF THE COMPLETED END-PRODUCT AND OTHER KEY RESEARCH AREAS (e.g., if this end product is not developed/completed on time, will it have implications on completion of another key research area)

- What other key research areas or development efforts would provide input to this information/product?
- What are the other key research or development efforts into which the desired information/product feeds in?
- How do the schedule constraints of other related key areas affect the outcome of this research project?

VIII.4 PLANNING

- When should the project be completed to support the licensing process?
- What are the industry projected time-frames for various license applications?
- What will be the impact of unanticipated delays in completion of the projects on the licensing process/schedules?

VIII.5 LEVERAGING (Is the desired information/product (or part of it) available from domestic or international partners?)

- Are there any domestic/international efforts in progress that may be relevant to our goals?
- If yes, what are the relevant ongoing domestic and international efforts?
- If not, should NRC be pro-active and take the initiative to formulate such domestic/international programs?

- Is NRC already participating in or have initiated steps to cooperate? Does DOE have a cooperative agreement where the information could be made available to the NRC?

- Do we have the required material (e.g., German pebble fuel or decommissioned AVR in-vessel specimens) to be able to conduct the tests ourselves? For that purpose, are experts and facilities available?

- Do the cooperative efforts fully support NRC research needs?

- If not, can those research programs be augmented to serve the NRC needs?

- If not, what part of the desired information would still remain to be developed? And, who (contractor/facility) would best serve our goals?

VIII.6 ANTICIPATED LEAD-TIME FOR THE DESIRED END-PRODUCT

Especially for the products involving long lead-time various consideration come into play:

- Is it technology-neutral (e.g., development of regulatory frame-work) or long-lead time products (e.g, fuel irradiation testing or specific rulemaking?

- Do we know of a pertinent ongoing international effort for which NRC does not at present have a cooperative agreement with that country/entity?

VIII.7 REQUIRED FISCAL/HUMAN RESOURCES AND SCHEDULE CONSIDERATIONS

- When does the NRC need to initiate the research efforts?

- Do we have required core staff expertise? If not, can we hire new staff/retirees to bridge the critical skill gap?

- Do we have appropriate contractor staff and facility to conduct and support the desired research, generate data, or develop the desired tool?

- How much time is needed for necessary quality check and/or independent testing/validation of the end-product?

- Are international experts available to NRC? What are the protocols for obtaining international experts? (On loan? As part of exchange program)?

- Do we have provisions in the budget for the next 5 years to support the research?

- What are the implications if we are not able to sustain the necessary research to completion?

IX COOPERATIVE RESEARCH

Unlike proven LWR technology where extensive LWR-related operational world-wide experience exists, the HTGR-related operational experience is limited and of the available data some may not be directly applicable. For instance, while the graphite-related advanced gas-cooled reactor (AGR) experience in the UK is expected to be valuable, extrapolation of some of the other AGR-related operational data to the new generation of HTGRs may be significantly restrictive or may have to be grossly approximated. Furthermore, inherent differences between the AGRs and the HTGRs in the context of reactor coolant chemistry (CO_2 vs Helium), operating conditions (higher temperatures in the HTGRs), as well as factors such as high enrichment and burn-up, would considerably limit direct application of some of the AGR operational data. In some instances (e.g., high-temperature materials performance or coolant chemistry issues), relevant data from other industrial experiences, namely, aviation and chemical, may have to be considered for developing insights. However, such data may be applicable only to a limited extent and will have to be used with caution.

IX.1 INTERNATIONAL COOPERATION

Inevitably, a great deal of HTGR-related needed data will have to generated in the laboratory settings under accelerated, simulated operational and post-accident conditions. Admittedly, this will be a time-consuming as well as an expensive venture. Consequently, it is expected that the NRC will have to continue to draw upon the existing domestic and international HTGR-related experience and research. Serious consideration of formal bilateral agreements or technology transfer arrangements with domestic as well as international partners in conducting future HTGR-related research and sharing data, will be an integral part of future planning. NRC's active participation in the ongoing research programs, as well as initiating new cooperative efforts with various international organizations, need to be designed so as to deliver optimum mutual benefits while off-setting costs.

IX.2 RELEVANT INTERNATIONAL EFFORTS

There is extensive gas-cooled reactor (GCR) operational experience in Germany and UK, including fuel performance and qualification data from the German AVR and the graphite behavior data from the British AGRs. Some of these data may be pertinent to the new reactor designs. There are other data that are needed and research efforts need to be focused to attain them. The existing AVR operational experience and data provide significant insights in identifying the future research needs. It is imperative that the NRC's future research programs take full advantage of the currently available information as well as on the future cooperative efforts both with the domestic as well as international partners. It is also believed that HTR-10 on China, HTTR in Japan, and HFR in the Netherlands will play a crucial role in providing to the international HTGR community the necessary experimental data and means for code validation. Other ongoing relevant efforts in various countries are considered to be vital to developing a thorough understanding of and establishing the necessary confidence in the HTGR design, safety and technology issues. Examples of such efforts include the following:

- air ingress and loss of forced circulation studies in Germany;

- high temperature materials qualification, including new graphite and new materials being tested, for example, under the Russian Federation and the European Commission's (E.C.'s) HTGR programs, respectively;

- fuel performance, neutronics, and equipment qualification related efforts sponsored by the E.C.;

- zero power neutronics experiments, fuel performance under reactivity insertion accidents, and other programs in support of GT-MHR and HTGR development for Pu disposition in Russia; and

- IAEA-sponsored Coordinated Research projects (CRPs) on code validation using data from HTR-10 and HTTR, as well the graphite database being developed under the sponsorship of IAEA.

IX.3 RECENT DEVELOPMENTS

In the year 2001, various advanced reactor workshops have been hosted by the NRC. On July 25, 2001, "Workshop on Future Licensing Activities" was sponsored by NRR. From October 10-12, 2001, an HTGR Safety and Research Issues Workshop was hosted by RES. The first two of these workshops were open to the public and were widely attended by the potential applicants and vendor representatives as well as consultants and the members of public. The RES workshop was by invitation only. It was intentionally kept free of parties with vested interest, such as, vendors, builders, potential licensee-applicants. Various HTGR experts from China, European Commission, Germany, Japan, Russia, South Africa, UK, US and IAEA, as well as representatives of the ACRS and MIT, and some consultants participated. Based on the workshop discussions, priorities were assigned to key HTGR safety issues, and future HTGR research needs as well as potential for several opportunities for international cooperative research were identified. Various international partners also offered to make available to the NRC their existing HTGR research experience and databases.

In February, 2002, the Director, RES, co-chaired a joint NEA-IAEA workshop on Advanced reactors, "Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs," held in Paris. At this week-long workshop, significant research topics related to advanced reactors as well as various research areas for possible future cooperation were identified.

IX.4 PROSPECTS FOR FUTURE INTERNATIONAL COOPERATION

Since beginning of the PBMR pre-application review process that was initiated in 2001, US delegations have visited South Africa, UK, Germany, China and Japan. There is considerable potential for future cooperative efforts with various countries. Additionally, technical information exchanges have recently been initiated between the NRC and the representative of the European Commission. The purpose of these exchanges is to understand the HTGR research programs and initiatives sponsored by the E.C. and to identify research items of common interest. Further dialogue will be necessary to identify areas of cooperative NRC-E.C. research. Formal agreements with the E.C. and other countries may materialize to affirm the commitment to share the existing HTGR-related data, and plan and implement future research efforts.

IX.5 PARTICIPATION IN IAEA-SPONSORED EFFORTS

The IAEA's documented data from various Coordinated Research Programs (CRPs), as well as international conference proceedings in various TECDOC, provide a significant information base. It is anticipated that the NRC will actively participate in the future HTGR-related CRPs. Participation in future specialists meetings similar to 1991 meeting on the subject of graphite development for gas cooled reactors at the Japan Atomic Energy Research Institute, and the 1995 meeting on graphite moderator life-cycle behavior (TECDOC-901) that was held in UK, will avail the most up-to-date information as well as resolution of various relevant issues and concerns. With support from Japan, South Africa, UK, and US, the IAEA has established a database related to irradiated nuclear graphite properties. The objective of this effort is to preserve the existing world-wide knowledge on the physical and thermo-mechanical properties of the irradiated graphite, and to provide validated data source to the member countries with interest in graphite-moderated reactors or development of the HTGRs, and to support continued improvement of graphite technology applications. The database is currently being developed and includes a large quantity of data on irradiated graphite properties, with further development of the database software and input of additional data in progress. Development of a site on the Internet for the database, with direct access to unrestricted data is also in progress.

Also under the auspices of IAEA, the objectives of the International Working Group on Gas Cooled Reactors (IWGGCR) are to identify research needs and exchange information on advances in technology for selected topical areas of primary interest to HTGR development, and to establish within these topical areas, a centralized coordination function for the conservation, storage, exchange, and dissemination of HTGR-related information. The topical areas identified include irradiation testing of graphite for operation to 1000° C, R&D on very high burn-up fuel, R&D and component testing of high efficiency recuperator designs, and materials development for turbine blades up to 900° C for long creep life. The duration of this CRP is from 2000 through 2005. Continued US participation in this and similar CRPs will be beneficial.

IX.6 PARTICIPATION IN OECD/NEA ACTIVITIES

The NRC anticipates a pro-active role in future NEA activities. In early 2002, the Director, RES co-chaired a joint NEA-IAEA workshop on advanced reactors, where key research topics were identified and future cooperative programs for their resolution were discussed (use Ashok and Charlie's trip report, when issued, for brief discussion on key items). Earlier, some useful conclusions at the First Information Exchange Meeting on Survey on Basic Studies in the Field of High Temperature Engineering, held in September 1999, identified various areas for future research. In a follow-on meetings it was re-affirmed that international collaboration should take full advantage of various reactors, (i.e. HFR in the Netherlands, HTTR in Japan, and HTR-I0 in China), to generate experimental data and to refine computer code qualifications. Irradiation tests were planned to take advantage of Russian reactors, IVV-2M, in particular. Integration of the European Program (HTRTN) with the Japanese and Chinese programs was strongly recommended in basic studies such as: core physics code qualification; fuel and material irradiation; graphite behavior and characterization. It was also recommended that

(1) A multinational group prepare a set of licensing and construction code guidelines specific to the new HTGRs that are commonly agreed upon;

(2) A set of internationally accepted safety guidelines for a modular HTGR be drafted;

(3) The design basis accidents and transients should be identified and simulated by appropriate code systems on the most elaborate modular HTR designs;

(4) Fuel performance and qualification be further explored; and

(5) The models that allow the prediction of irradiation damage in graphite using unirradiated material properties should be further developed.

It was concluded that the existing databases on irradiation damage effects on carbon-carbon composite materials and ceramic composite materials are not sufficient for designing nuclear fission. Since irradiation experiments need extensive time and resources, it is important that information exchange on irradiation experiment details should be done effectively.

IX.7 INTERNATIONAL COOPERATION THROUGH DOE

The NRC-DOE cooperative efforts encompass a wide range of HTGR issues. Both DOE and NRC are exploring opportunities for collaboration in international R&D efforts related to the GCR technology. A current DOE-NRC Memorandum of Understanding (MOU) may also be expanded to encompass future efforts in conducting the HTGR fuel testing and experiments. Currently, under the DOE sponsorship, as part of the Nuclear Energy Research Initiative (NERI) program, various reactor designs and high burn-up and enrichment related research projects are being conducted at various organizations, including U.S. universities (24), DOE national laboratories (10), industry organizations (20), as well as foreign R&D organizations (24). There are nine ongoing projects under NERI that relate to the GCR technology. The GCR fuel irradiation program and the GCR fuel technology R&D efforts are currently being planned. Of the NERI programs, the projects related to gas-cooled reactors that are of particular interest to the NRC include fuel component designs; researching better reactor materials; and basic chemistry. Under NERI, DOE is also supporting development of the IRIS design the research for which is being supported by, in addition to Westinghouse, various US universities, as well as Polytechnical Institute of Milan, Italy.

The International Nuclear Energy Research Initiative (I-NERI) efforts include collaborative agreements between US and France, and US and the Republic of Korea (ROC) on gas reactor technology. The US-France agreement of May 2001, relates to the joint development of advanced nuclear systems. This agreement is part of DOE's I-NERI to foster international collaborative research and development of nuclear technology, focusing on the development of advanced nuclear system technologies. The joint research awarded through this agreement will enable the US and France to move forward with leading-edge generic research that can benefit the range of reactor and fuel cycle designs anticipated in the future. DOE is currently developing a Generation IV Technology Roadmap that, when completed next year, will serve as the research and development plan for advanced reactor and fuel cycle system development. In a November 2001 US-ROC agreement, the areas of collaboration include R&D in the following areas: advanced I&C and diagnostics (including advanced digital I&C, software validation and verification; and advanced condition monitoring of components and systems); ALWR technology (including advanced materials for fuel, cladding, and reactor structures); advanced fuel technology (including high burn-up, thorium, particle fuels); and innovative safety research (including advanced computational methods for seismic, thermal-hydraulic, nuclear analyis).

IX.8 DOMESTIC COOPERATION

The current pre-application review of the PBMR design and the possible near-term pre-application review of the GT-MHR have heightened the urgency of some of the needed HTGR-related research, especially in those areas where long lead times are anticipated. Examples include development of a generic regulatory framework, TRISO-coated fuel irradiation testing as well as high-temperature materials performance issues that need to be addressed on an urgent basis. However, budget constraints and limited domestic resources would necessitate cooperative research efforts among the government agencies (e.g., DOE and NRC), national laboratories, industry (e.g., joint collaboration on experimental set-ups with applicants to generate the needed data for independent analysis), and various universities. Some of the ongoing efforts are purely domestic; however, others involve participation by many foreign Research and Development (R&D) organizations who have joined ongoing activities.

IX.9 DOE-SPONSORED RESEARCH AND OTHER INITIATIVES

For many years ending in the early 1990s, DOE sponsored the modular High Temperature Gas-cooled Reactor (MHTGR) Program. This program culminated in a draft safety evaluation review by the NRC of the MHTGR design in 1989 (NUREG-1338). Subsequently, in the late 1990's, due to the continued focus on nuclear energy being a viable energy source, DOE initiated a new program called the Nuclear Energy Research Initiative (NERI). NERI is intended to stimulate universities, industry, and national laboratories to innovate and apply new ideas to old problems. The DOE research funds for generic work on both HTGR and ALWR comes from NERI. The NERI budget for FY 2002 is $27.1 million; however, there is fierce competition for this pool of money from researchers involved in international activities, Generation IV activities, as well as current efforts to optimize the existing nuclear power plants.

The cooperative research efforts between DOE and the Electric Power Research Institute (EPRI) focus on advanced light water reactors and research to optimize the operations of the current operating fleet of nuclear plants. EPRI, in cooperation with the Nuclear Energy Institute and other nuclear industry organizations, developed "Nuclear Energy R&D Strategy Plan in Support of National Nuclear Energy Needs" and provided it to DOE to initiate joint planning and coordination efforts toward common R&D goals.

IX.10 INDUSTRY AND UNIVERSITY RESEARCH

General Atomics has an on-going joint project with Russia to build an HTGR for plutonium disposition. This project is intended to lead to the development, fabrication, and demonstration of key GT-MHR components such as the turbo machinery and its major components, reactor vessel and internal materials, and the fuel based on a plutonium oxide coated particle fuel. While the Russian plant is not a commercial venture, the research for this plant could be transferrable to the commercial GT-MHR design.

The Massachusetts Institute of Technology (MIT) is conducting research on a modular high temperature gas cooled pebble bed reactor. Students and faculty are engaged in research on core neutronics design, thermal hydraulics, fuel performance, economics, non-proliferation, and waste disposal. The objective of this research is to develop a conceptual design of a 110-Mwe pebble bed nuclear plant which could be used as demonstration of its practicality and competitiveness with natural gas. In addition to MIT with its consortium of US universities, national laboratories, and industries, this research involves international collaborations with Germany, Russia, China, Japan, and South Africa.