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# ADVANCED REACTOR

# RESEARCH PLAN

Revision 1
Office of Nuclear Regulatory Research
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Attachment

## *Executive Summary*


In a Staff Requirements Memorandum (SRM) dated February 13, 2001, the Commission directed the staff to assess its technical, licensing, and inspection capabilities, and identify any enhancements that would be necessary to effectively carry out the Agency's responsibilities in licensing new reactors. The Commission also directed the staff to critically assess the regulatory infrastructure supporting Part 50, Part 52, and other applicable regulations that may require improvement. In response, the staff prepared and issued a report, "Future Licensing and Inspection Readiness Assessment [FLIRA]," dated September 2001.

The FLIRA report committed the staff to the development of an advanced reactor research plan which is the subject of this report. During plan development, the staff focused on the critical information that will be needed to establish safety standards for new reactor designs. The plan captures both technical and potential safety issues that involve great uncertainties., and identifies capabilities that will enable the staff to independently ask the right questions. At this point, the plan does not delineate the research that will be conducted by the NRC. Rather, it identifies information gaps that exist at NRC in terms of the analytic tools and data shortcomings, and attempts to encompass activities that aim at either applying existing knowledge or creating new knowledge. It is also recognized that an applicant of a new reactor design has the primary responsibility to demonstrate the safety case of a proposed design.

The scope of the plan includes both confirmatory and anticipatory research, as it applies to four reactors identified in the FLIRA report. These reactors include the Pebble Bed Modular Reactor (PBMR), Gas Turbine-Modular Helium Reactor (GT-MHR), Westinghouse advanced pressurized water reactor AP-1000, and International Reactor Innovative and Secure (IRIS). Generation IV (Gen IV) reactor concepts have not been included in the plan at this time, because of their preliminary stage of development. The plan, however, is expected to be maintained as a living document, and will be modified to accommodate any new issues and technologies not previously considered. This will include, for example, the Gen IV concepts now under development and other designs now being considered for pre-application review (e.g., European Simplified Boiling Water Reactor (ESBWR), Boiling Water Reactor (SWR-1000), and Atomic Energy of Canada Limited ACR-700.)

The plan originates from a technology-neutral perspective. Discrimination between technologies was required, however, once design-specific safety issues needed to be considered, or when modifications to existing analytical codes for specific applications needed to be addressed. The key technical areas and topics include: (1) development of regulatory decision making tools based on the risk-informed, performance-based principles; (2) accident analysis (PRA methods and assessments, human factors, and instrumentation and control); (3) reactor/plant systems analysis (thermal-fluid dynamics, nuclear analysis, and severe accident and source term analysis); (4) fuels analysis and testing (fuel performance testing, and fuel qualification); (5) materials analysis (graphite behavior and high-temperature metal performance); (6) structural analysis (containment/confinement performance, external challenges); (7) consequence analysis (dose calculations, environmental impact studies); (8) nuclear materials safety (covering enrichment, fabrication, and transport) and nuclear waste safety (covering storage, transport, and disposal); and (9) nuclear safeguards and security.

It should be understood that not all the work in the plan has to be done by NRC.  Information can be obtained through domestic and international cooperation, as well as through work done by developers.  Accordingly, prioritization and budgeted resources will need to reflect the fact that some of this information will need to be obtained from others.  As more information becomes available, the plan will be updated to reflect only those activities that require NRC resources as presented in the FY 2003–2005 budget projections.

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# 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 early site permit (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, “Future Licensing and Inspection Readiness Assessment [FLIRA],” SECY-01-0188, October 12, 2001, which assessed the 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 envisioned that the research plan would help formulate and set directions for research programs, including programs to develop a regulatory framework for advanced designs, and analytical tools and experimental data to independently assess the safety capacity of the new reactor designs. 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. Implementation of this plan will include full participation of NRC staff from the Offices of Nuclear Reactor Regulation (NRR), Nuclear Material Safety and Safeguards (NMSS), Nuclear Security and Incident Response (NSIR), and Nuclear Regulatory Research (RES).

The Commission also issued a SRM on June 19, 2001, that approved the staff's plan (SECY-01-0070) to proceed with pre-application review of the Pebble Bed Modular Reactor (PBMR), and develop necessary research infrastructure to support the advanced reactor licensing process. Pre-application reviews provide a forum for early interaction between the NRC and the reactor designers, and facilitates the identification of key safety and policy issues, and potential paths for their resolution. Insights from pre-application reviews supported the development of the advanced reactor research plan. In addition, any specific research activities and infrastructure needs identified during the pre-application review get enter into the Planning, Budgeting and Performance Management process and be 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, and preserve the defense-in-depth philosophy. New reactor designs being pursued by industry include the Gas Turbine-Modular Helium Reactor (GT-MHR), the Westinghouse AP-1000, and the International Reactor Innovative and Secure (IRIS). All are considered to be within the scope of this plan. Generation IV (Gen IV) reactor concepts have not been included in the plan at this time, because of their preliminary stage of development. As discussed below, the plan is expected to be maintained as a living document, and will be modified at some later date to accommodate Gen IV reactor concepts, and other designs now being considered for pre-application review (e.g., ESBWR, SWR-1000, and ACR-700).

The existing NRC research and regulatory infrastructure primarily supports licensing of the current generation of LWRs. Although there are several areas in which the research infrastructure needs to be improved to address ALWRs, specifically with regards to the IRIS design, most research infrastructure gaps relate to HTGRs. These reactors present new challenges to the NRC, both from a technical and safety perspective. To effectively and efficiently address these challenges, modifications to the existing regulatory framework may be necessary. Any modification will require a sound technical basis. One of the key research activities within the scope of this plan, is to develop new risk-informed, performance-based regulatory decision-making tools that will support modifications to the regulatory framework.

In developing this plan, the staff focused on critical research areas and information that would be needed to technically support an advanced reactor license submittal review. At this point, the plan does not delineate what research, for example, will be conducted by the NRC versus the developers of new reactor designs, but rather focuses 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. The plan has also been developed from a technology-neutral perspective. At some level, however, the process needed to discriminate between different technologies (i.e., LWR vs. ALWR and/or LWR vs. HTGR), specifically when design-specific technical or safety issues needed to be considered, or when future modifications to the existing analytical codes for specific applications had to be addressed.

Most NRC regulations were developed for LWRs, and in certain cases these regulations may not apply to a future licensing application. In these cases, NRC may need to develop new safety limits, or upgrade its data base to assess safety margins or issues not previously considered. These activities are also captured under this plan.

It is envisioned that the plan would be maintained as a living document, and it will be updated as appropriate to accommodate any new issues. Future updates will (1) identify any new information from applicants and potential applicants, international research activities, and the Department of Energy (DOE), and (2) reflect plans to independently confirm an applicant’s findings. 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.

## II. ROLE OF NRC RESEARCH

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 existing knowledge and tools, or creating new knowledge and tools. It is expected that applicants will provide supporting arguments and documentation based on 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. The NRC also performs research to explore issues involving large uncertainties and to develop independent capabilities to enable the staff to review applicants' submittals. The duration of this research varies between short-term efforts 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 arise from the examination of industry trends and insights, insights that help the NRC foresee what information will be needed to respond to future regulatory issues.

While assessing challenges posed by the new reactor designs and technologies, the staff will need to consider, what research would be conducted by the applicants as part of their license application and the needs of the licensing office, then adjust accordingly. Research may be conducted by others with a vested interest (e.g., generic and technology-neutral research sponsored by DOE, or industry-supported organizations). Experience with the AP-600 certification, for example, indicates that the scope, schedule, and resources for such research programs are extensive, and that the staff could benefit from worldwide developmental research and experience. Mindful of our respective roles, and consistent with the NRC Strategic Plan, the NRC will continue to seek opportunities to interact with and, where appropriate, initiate cooperative programs with other agencies and organizations. These include U.S. universities and domestic organizations such as 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), and the European Commission (E.C.). 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 are an important part of the NRC's Strategic Plan.

In general, NRC research is focused on the development of expertise, tools, and methods that are needed to support the Agency's mission in understanding and resolving potential safety issues. The development of such expertise and methods contributes to the overall effectiveness and efficiency of the agency by helping to ensure high quality and timely reviews. Tools such as computer codes and experiments that generate data to validate these codes have also played an important role in that mission by providing the agency with the capability to independently assess plant safety and safety margins. Most of the existing NRC codes, however, were developed for LWR applications, and need to be modified to evaluate HTGR designs.

The NRC's statutory obligation demands that NRC institute a licensing process that will lead to decisions on significant safety issues that are of high-quality, technically sound, and supported by robust research. In planning research activities, the focus is primarily on areas where important gaps exist (e.g., in technological knowledge, 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 to be used to bridge the technological gaps. Another important facet of research involves materials testing and associated codes and standards development, which generally involves a consensus process. Such a process takes a long time and, as in the past, pre-application reviews are being used to identify the necessary new (or need to modify existing) codes and standards early in the process.

The general principle to be used for funding 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 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 the NRC is the beneficiary, research can be jointly funded by industry (or one segment of the industry) and the NRC. It is essential, however, that NRC's independence not be compromised in the process, that the quality and integrity of the data be maintained, and that all legal and administrative requirements be met. The process equally applies to relationships with other government agencies such as 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 be the applicant for new advanced reactor designs. Research needed to establish acceptance criteria associated with a new safety standard or requirement, or to address specific issues for a particular reactor design, can be funded independently by the NRC, in cooperation with the 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, and safety margins 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 understanding 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 and research findings, a new set of questions may be raised. The importance of answering the new set of questions by examining the question's pertinence to the 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, establishing a priority, and identifying the expected end-products. Routine peer reviews of the research products and anticipated schedules for specific research activities will be conducted to instill confidence in the scope and quality of the research; these reviews will include frequent interactions with the Advisory Committee on Reactor Safeguards (ACRS) and the Advisory Committee on Nuclear Waste (ACNW) to obtain feedback and guidance, as well as strong involvement of NRR and NMSS.

## III. OBJECTIVES AND STRUCTURE

The advanced reactor research plan will be used to generate and implement a research infrastructure to support licensing of advanced reactors. Within this context, the plan will be used 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 and flow of information among the various technical disciplines,
- Key research output results and links to the regulatory process,
- Priorities used to allocate resources,
- Key milestones and resources over a 5-year period (FY 02-FY 06).

The plan will provide a platform for communicating program objectives and goals and receiving feedback from internal and external stakeholders. The research activities within the scope of the current plan currently include PBMR, GT-MHR, IRIS, and AP-1000 designs.

Two types of research were considered essential: (1) research to establish the technical basis for regulatory decision making, and (2) research necessary to address uncertainties and gain insight into safety margins and failure points. In many ways, the first depends on the second, since building a sound technical basis requires a deep understanding of the technology, its application, and the inherent uncertainties. In general, research results would be used to support safety evaluation reports, or used to establish guidance in the form of regulatory guides, standard review plans (SRPs), or NUREG reports.

The plan also integrates ongoing research initiatives in both the domestic and international arenas. 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 activities that will require NRC funding consistent with FY2003-2005 budget projections.

In drafting this plan, the staff benefitted from the ACRS Advanced Reactors Workshop (June 2001), 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. Workshop participants assigned relative priorities to research areas and identified several opportunities for international cooperative research that draw upon existing domestic and international experience. Additional insights were gleaned from the June 4, 2001, ACRS Subcommittee on Advanced Reactors meeting that focused on regulatory challenges for future nuclear power plants. NRC staff also participated in and capitalized on feedback from the "Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs," held February 18–20, 2002, by the Organization for Economic Co-Operation and Development, and Committee on the Safety of Nuclear Installations (OECD/CSNI). Insights from NUREG-1802, "Role and Direction of Nuclear Regulatory Research," were used. Interactions with the ACRS and ACNW are planned for the second half of 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, as well as the ongoing PBMR pre-application review. The MHTGR review was 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.

To facilitate the identification of research areas important to the development of an infrastructure, a top-down approach was used as shown in Figure 1. The approach utilized the NRC strategic plan, and categorized research programs by three of the four strategic arenas: Nuclear Reactor Safety, Nuclear Materials Safety, and Nuclear Waste Safety. The fourth strategic arena, International Nuclear Safety Support, is intrinsic to the planning process and not separated out. Safety and safeguards was also considered in the plan. As shown in Figure 1, key research outputs were identified and linked to key research areas. It should be noted, however, that at the activity level, the figure does not contain all the research activities considered in the plan.

**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, which use defense-in-depth principles and conservative practices, provide a degree of margin that might not be applicable to PBMR or GT-MHR advanced reactor designs in all areas. To support the Nuclear Reactor Safety arena for these advanced designs, the current regulations will need to be applied and expanded as appropriate. To support these regulatory activities, research areas and activities were aligned to the four cornerstones of reactor safety:

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

Figure 1 shows the alignment and identifies the key research areas. In addition, some of the associated activities as shown includes the following:

| | 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 discussions of these activities are provided in the plan. In general, research products resulting from these activities either establish a technical basis for resolving specific safety issues or support another research area. Identified technical or safety issues associated with the research areas helped define the research infrastructure needs, schedules, and resource projections.

Information flow among the technical groups and framework is illustrated 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 provides key input to 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.

- Insights and data generated by the reactor/plant analysis (e.g., success criteria), together with performance information involving human factors considerations, I&C, and modeling assumptions, enter 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 enters into and becomes critical to the regulatory framework and associated decision-making activities.

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

It is important to note that the process does not generate a system of discrete and isolated technical disciplines working independently, but an integrated system that is both risk-informed and performance-based.

Identification of key accident scenarios is an important aspect of a licensing process. These events typically will 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. When 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 assess margins and 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 the research that is needed to support the framework.

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

**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 – storage, transportation, and disposal of spent-fuel and low-level waste

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

**Safeguards and Security**

Advanced reactor research efforts for the arena of Safeguards and Security will support the regulatory offices, principally NSIR, in the assessment of proliferation potential and the evaluation of security measures and the 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.

As requested by or through NSIR, RES will support NSIR and other offices and agencies with information needed for assessing, comparing, 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

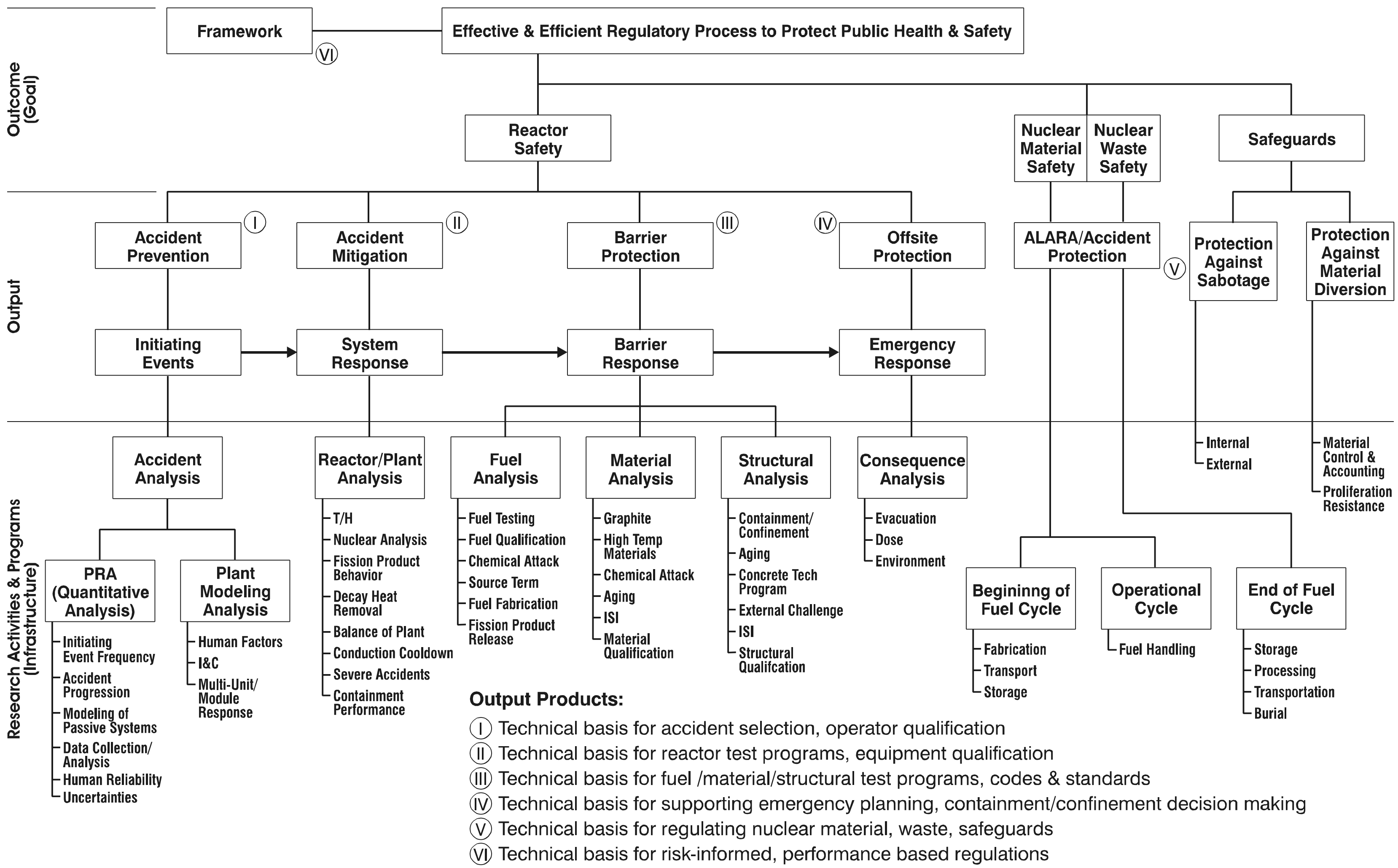


Fig 1. Key research areas for examination.

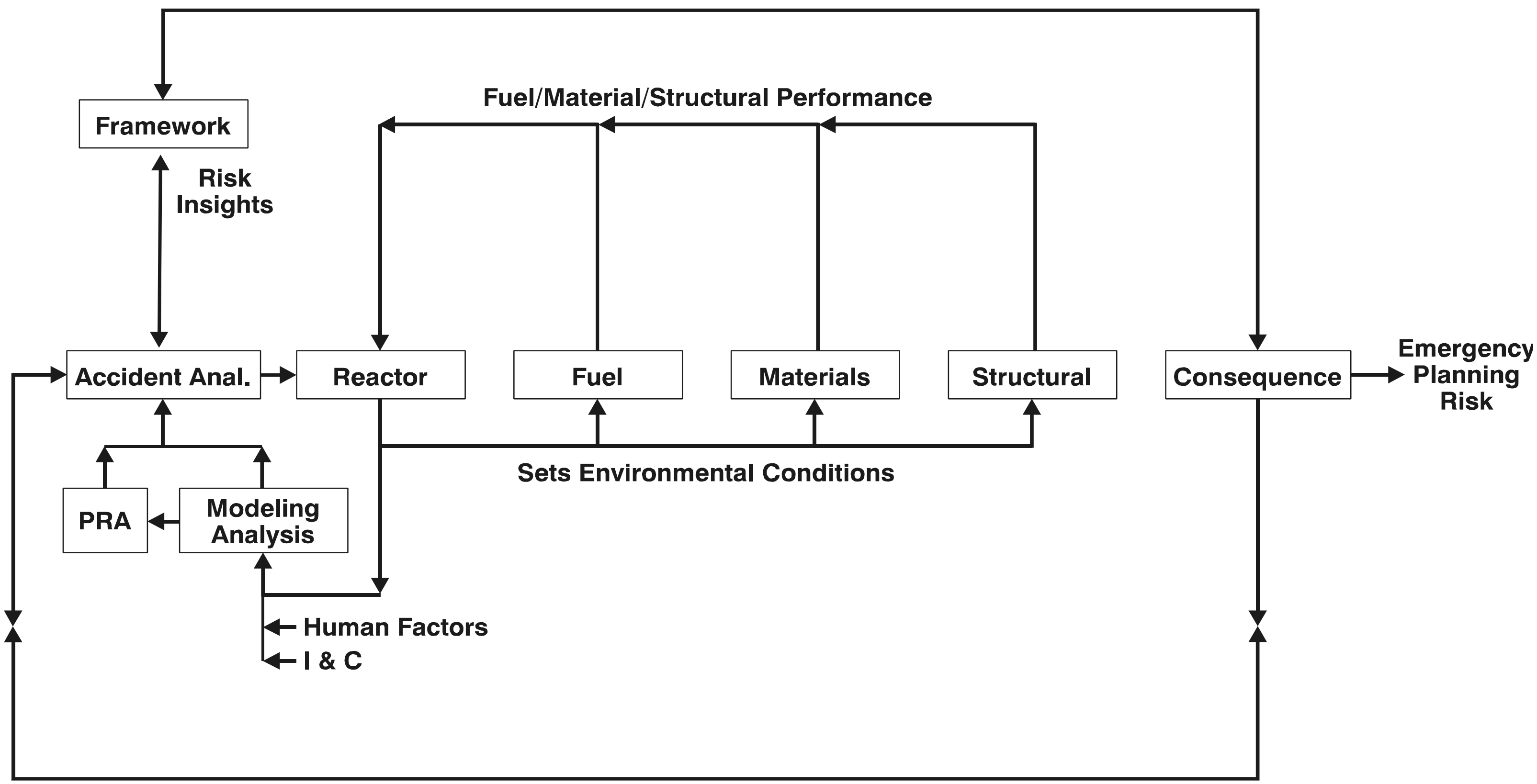


Fig 2. Information flow between technical groups.

# IV. KEY RESEARCH AREAS AND ACTIVITIES

## IV.1 GENERIC REGULATORY FRAMEWORK DEVELOPMENT

### IV.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 technology 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 value of a licensing framework applicable to reactor designs that are different from currently operating plants. This framework would help to ensure that a structured and systematic approach ensures uniformity and consistency in the licensing and regulation of advanced reactors, particularly when addressing the unique design and operational aspects of these reactors.

In addition, 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 "technology-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.

### IV.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 both risk-informed and performance-based. Both deterministic and probabilistic results and insights will be used to identify what regulations are needed to govern 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 principles, such as defense-in-depth and safety margin.

### IV.1.3 Objectives and Planned Activities

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 3. 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. These will also be used to assure that the framework is appropriately performanced-based. 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.

Utilizing the above approach, a reactor and technology neutral 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 technology-neutral or global set of regulations for advanced reactors. Figure 2 is a general illustration of the development of the technology-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 technology-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.

The above licensing framework will be used to identify and formulate what regulations are needed. Potential regulations will be technology-neutral or globally applicable to all reactor types currently under consideration.

#### IV.1.3.1 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 technology-neutral regulatory requirements. The reactor-specific regulatory guides will not provide the detailed guidance for implementation of specific technical requirements, but will provide the proposed guidelines for expanding the technology-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 technology-neutral licensing framework will be used to identify and formulate both potential reactor-specific regulations and regulatory guides as needed.  These products will be developed for each of the advanced reactor designs under consideration.

#### IV.1.3.2 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.

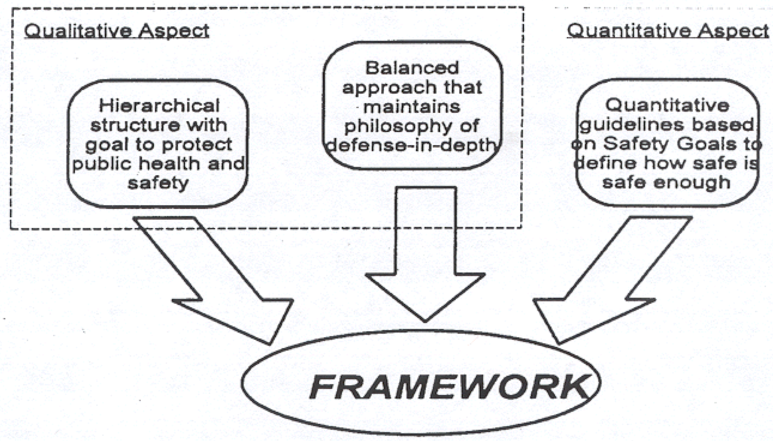


Figure 3 - Aspects of the Framework

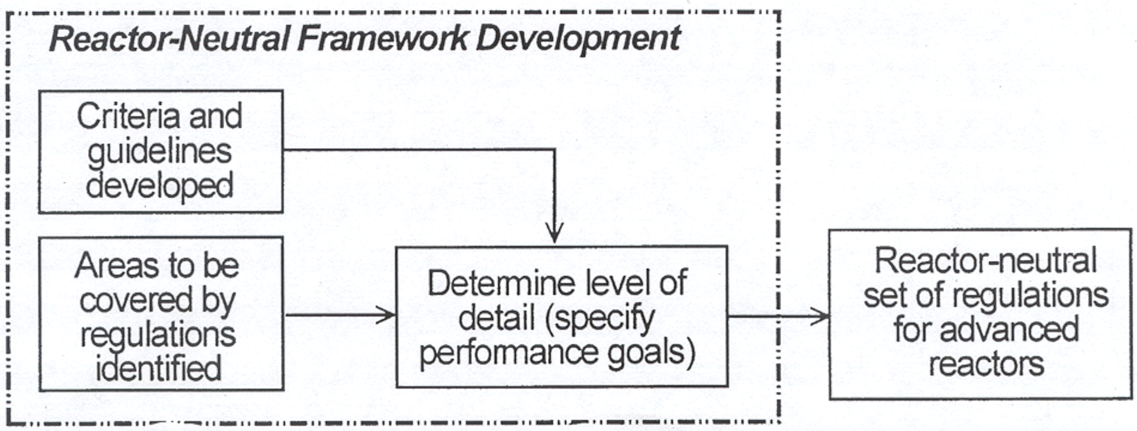


Figure 4- Development of a Reactor-Neutral Licensing Framework

## IV.2 REACTOR SAFETY

### IV.2.1 Accident Analysis

#### IV.2.1.1 Probabilistic Risk Assessment (PRA)

IV.2.1.1.1 PRA – Background

Future licensees have indicated that PRAs will be an integral part of their applications and NRC expects to 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.

During the past 30 years, the NRC has performed several PRAs, and has promoted PRA 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 for 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 existing PRAs. 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.

IV.2.1.1.2 PRA – 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 provide adequate safety margins and defense-in-depth. This approach to licensing can lead to unnecessary regulatory burden. 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 tools, methods, and 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.

PRA will be used in the licensing of advanced reactors, which is an application not as well developed as the use of PRA in risk-informed regulation of current LWRs. Applicants will provide arguments for the acceptability of their proposed advanced reactor design, on the basis of PRA results. Safety margins and defense-in-depth will be retained to protect the health and safety of the public, and PRA results and insights will be used to enhance the traditional approach. This dual process should bring all the technical information to bear in a structured fashion and keep to a minimum the prudent conservatism that must be applied to account for uncertainties. 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).

#### IV.2.1.1.3 PRA – Objectives and Planned Activities

The objective of the advanced reactor PRA work is to develop review guidance for NRC reviewers, explaining how to independently review advanced reactor PRAs. To develop this guidance, it is necessary to obtain:

- the data for the PRA,
- an understanding of the uncertainties,
- the methods necessary to model advanced reactor designs in PRAs, and
- the expertise to evaluate advanced reactor PRAs.

In the process of developing the review guidance, we will gain:

- an understanding of regulations needed as part of the licensing process, and
- identification of additional research needed.

This plan is comprised of three tasks. The first task is to develop the methods, data, and tools needed for evaluating the design and operational characteristics of advanced reactors that are different from those of current reactors. The second task is to use the results of the first task to: (1) gain expertise, (2) provide risk perspectives on other important areas of research in this plan, and (3) evaluate advanced reactor designs. Use will be made of existing PRAs to develop preliminary PRAs which will be revised as plant-specific information becomes available. Using this process will identify areas where additional research is needed and provide an ability to prioritize the needed research. The third task is documentation will provide guidance for the review of applicant's advanced reactor PRAs.

**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.

- **Initiating event identification and quantification**

The events that challenge advanced reactors will include some events common with the current generation of LWRs, (e.g., loss of offsite power and seismic events), and will include events specific to advanced reactors. It is necessary to 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 comprises the first step in assessing the challenges associated with a given reactor design. Extensive use of existing PRA information will be used, as appropriate. This quantification will provide the necessary initial data on initiating event frequencies for use in the PRA. As the PRA is developed to be plant-specific, the significant initiating event challenges will be reevaluated.

- **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 are likely to 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, the loss of helium and the effects of air (and potentially water) ingress on the accident progression need 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 ability of a reactor containment or confinement with a filtered venting system to provide protection against release of fission products. (The confinement concept has been successfully modeled in past PRAs, although it has not yet been applied to 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 reviews 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.

- **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 need to be determined for incorporation into the PRA. Passive systems have been treated in PRAs, such as in the AP-600 PRA, as either initiators (e.g., loss of coolant accidents (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. Other conditions could include a degraded or intermediate failure states. Therefore, the modeling approach needs to be reconsidered for potential modifications for advanced reactors and could proceed using the information from the AP-600 and AP-1000 designs until advanced reactor plant-specific information becomes available.

Digital systems typically have not been considered in past PRAs. In advanced reactors, however, I&C systems will normally be digital, which could include touch displays, fiber optic cables, computers and microprocessors. 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 or the timing of the failure modes could be different. For example, digital I&C could be more susceptible to what would previously have been considered low voltage spikes (because the digital components typically operate on 5 VDC instead of 120 volts), fail sooner under fire or loss of cabinet cooling conditions, or radiation damage for fiber optic cables. 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 and quantified to the extent practical.

- **Data collection and analyses**

Advanced reactors may 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 PRAs will be used, as applicable.

This task includes addressing the data uncertainties. Understanding the uncertainties is a very important aspect for any PRA; the uncertainties are likely to be much larger for advanced reactors given limited or lack of applicable data and operating experience, and the expected significant use of the PRA in the licensing process.

- **Human reliability analysis**

The operators' role and staffing levels in the new reactors is likely to be different than in current generation plants. The advanced reactor designs are proposed with strong reliance 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 operator performance (e.g., staffing and training) are part of a different activity of this plan. Human reliability methods 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 human error 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, both from a common mode failure and as the result of operator workload from monitoring multiple modules. Further, the extensive use of digital I&C (e.g., touch screens and different control designs) could impact the probability of human error and needs to be investigated. The likelihood of errors of commission or omission need to be understood under these conditions.

- **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 from that of electro-mechanical 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 need to be considered for advanced reactors from a scoping perspective to identify unique vulnerabilities.

- **Quantification**

The information gathered from the aforementioned areas needs to be integrated into a code to develop insights and provide guidance into other areas, such as thermal/hydraulics analyses. The SAPHIRE code could be used in the performance of an independent PRA but 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 SAPHIRE can reasonably handle 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 evaluated as part of the severe accident and consequence work of this plan, need to be incorporated into a PRA tool. A full scope PRA tool that integrates Level 1 (CDF) analyses with Level 2 and Level 3 analyses, and dynamic modeling is needed to provide the insights for developing review guidance.

- **Uncertainties**

Identification of uncertainties will help the decision-making process for deciding either to reduce the uncertainties by more research or to strengthen 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. Processes will be developed to identify and understand the significance of the modeling and completeness uncertainties.

- **Other operational states**

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

- **Multiple modules**

Current PRAs are usually performed for a single unit or sometimes for two sister units operating independently, but considering cross-ties. Some advanced reactor designs have identified that up to 10 modular units will operate 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) need to be considered.

- **Risk metrics**

The concepts of core damage frequency (CDF) and large early release frequency (LERF) may not be the best figures of merit for some advanced reactor designs. 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 a full scope PRA. Therefore, for advanced reactors, either the current subsidiary figures of merit need to be verified or more appropriate figures of merit need to be identified, consistent with NRC top level safety goals. Appropriate figures of merit will be developed for a policy paper for action by the Commission. After Commission approval, these figures of merit will be incorporated into the review guidance documents.

- **Safeguards and security**

As mentioned above, 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.

**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 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 license applications.

**Task 3. Documentation**

The documentation will document the insights which include identification of research needs, and provides information for developing regulatory guides and SRP sections. A wealth of information will be generated by performing Tasks 1 and 2. The PRA and review guidance should be sufficient for a reviewer to be able to determine the probabilistic implications of different design configurations and operation conditions. The documentation will provide insights for developing 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:

- assist the staff in independently reviewing advanced reactor PRAs,
- help identify research needs, and
- develop regulatory guides and SRP sections.

#### IV.2.1.1.4 PRA – Application of Research Results

The application of this work will be to:

- use 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.

### IV.2.1.2 Instrumentation and Control (I&C)

#### IV.2.1.2.1 I&C – Background

The new generation of advance reactors, 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 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 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.

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 improved 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 develop the knowledge and tools needed to support the review of these new reactor technologies. In some cases the research described in this section will be similar to the ongoing research in support of digital upgrades to exiting plants. Where appropriate, these activities will be coordination to ensure duplication of effort is minimized.

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, has developed integrated advanced control rooms and performed more research in the areas of automation of plant operations and advance plant monitoring and diagnosis than has 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 both by the vendors and through joint efforts with other organizations, such as 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.

The Department of Energy (DOE) research program to support development of future use of nuclear energy in the United States (US) currently includes six Nuclear Energy Research Initiatives (NERI) grants in the I&C area. These include research in the areas of automatic generation of software, control architectures, self diagnostic monitoring systems, smart sensors, and advanced instrumentation to support HTGRs. In addition to the current NERI grants in the I&C area, DOE's Long-term Nuclear Technology Research and Development Plan calls for additional research to support implementation of new technologies such as robust communications and wireless sensors, condition monitoring, distributed computing, advanced control algorithms, and on-line monitoring. All of these technologies could be used to support implementation of the advanced instrumentation and control systems for HTGRs.

#### IV.2.1.2.2 I&C – 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, shutdowns, and changes of operating modes. There will be fewer operators compared with the current generation nuclear power plants, (i.e., there may be 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. This will require a level of automation and coordination that is more complex than needed for current generation plants.

Because of the longer fuel cycles and much longer time between maintenance outages, the plants may 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.

Highly automated control rooms in other industries have used modern control theory controllers to increase plant availability and decrease workload on operators. It is likely that the new HTGRs will use some of these advanced 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 be investigated.

To 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 limitations in the models and data to support risk analysis, the uncertainties in this area are relatively high. Additionally, the reliability and security of digital instrumentation and control systems will be an ever increasing issue as advanced deigns perform more and more sophisticated safety and control functions.

#### IV.2.1.2.3 I&C – 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 :

- **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.

- **New risk models for I&C systems in advanced reactors**

This effort will complement the work that is currently being done at the University of Maryland and the University on Virginia but will focus on the development of 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).

- **Analysis of the requirements and potential issues involved with HTGR instruments**

This effort will include review of the existing requirements for 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, and others. This effort will also support the review of needed prototype plant instruments. This research will focus on generic issues and not on viability or preferability of various technologies.

- **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.

- **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 investigated.

- **Analysis of on-line monitoring systems and methods and advanced diagnostic methods needed to support HTGR**

This effort will include the review of current methods and 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 inservice inspection (ISI) intervals will also be evaluated.

- **Review of advanced control algorithms for application to advanced reactors**

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

- **Analysis of the requirements and potential issues involved with advanced light water cooled reactor instruments**

This effort will review the requirements for 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.

- **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 the current availability and maintenance schedules. How the limits and new capabilities of these systems will affect other issues, such as ISI intervals, will also be evaluated.

IV.2.1.2.4 I&C – Application of Research Results

The results from the first effort will provide insights and guidance 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 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. These programs will provide information for revisions to Chapter 7 of the SRP and in the supporting Regulatory Guides.

### IV.2.1.3 Human Factors Considerations

IV.2.1.3.1 Human Factors Considerations – Background

Nuclear power plant 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 that human error is 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 of and recovery from failures.

IV.2.1.3.2 Human Factors Considerations – 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 concurrently control multiple modules, which may be in different operating states, from a common control room. Operators will be required to monitor online refueling in one module, while other modules are in normal operating states and while another module 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 automatic systems rather than the active control that LWR operators now take. Higher-levels of knowledge and training may be needed to respond to situations when automatic systems fail. Any of these changes can pose new and challenging situations for operators and maintainers. RES can provide the regulatory staff with tools, developed from the best available technical bases, to support 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.

In accordance with 10 CFR 52, the staff of the NRC reviews the human factors engineering (HFE) programs of applicants for construction permits, operating licenses, standard design certifications, and combined operating licenses. Under 10 CFR 50, the staff also reviews licence amendments. The purpose of these reviews is to help ensure safety by verifying that acceptable HFE practices and guidelines are incorporated into the applicant's HFE program. The review methodology in NUREG-0711, "Human Factors Engineering Program Review Model," and SRP Chapters 13 and 18 is 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 (V&V); 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 ensure that human factors activities are risk-informed, there needs to be a close synergism with the human reliability analysis (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.

Currently there is no facility in the U.S. 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. There is a plan to build a PBMR simulator in South Africa with a completion date in late 2003. The French have reactor simulators that they operate or are developing for the N-4 reactor and for other concepts they are considering. There are also research simulators in Japan and Korea. The OECD Halden Reactor Project operates three reconfigurable research simulators (pressurized-water reactor (PWR), boiling-water reactor (BWR) and water-cooled water-moderated power reactor (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.

IV.2.1.3.3 Human Factors Considerations – Objectives and Planned Activities

Since much is still unknown about the human factors aspects of advanced reactors, only the initial task below and the task on staffing will be performed. The first three tasks and work done on human reliability analysis area will help direct future work.

- **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 licensees. 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 research plan, the NRC must first determine from the 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:

– 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. There are many advanced automated systems in transportation, aerospace, and petrochemical industries that may have operational similarities to advanced reactors. Research and experience related to such systems would 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 designers deal with the automation and potential failure of automation? How will operators be expected to control multiple modules? What will the operators' 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 plant staff 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 must 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 that are developed must have a sound technical basis derived from original research or information that can be adapted to NRC guidance without need for further research.

– 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. Rules, regulatory guides, NUREGs, the SRP, and consensus standards from IEEE and ANS/ANSI and proposed guidance from industry organizations (e.g., NEI, EPRI) would all be reviewed. Topics such as staffing, procedures, training, human-system interface, and fitness-for-duty would be included. 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 the plant, and others.

- **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 for conventional reactors. Since each of the existing conventional reactors is unique, each plant has a plant-specific simulator. However, it is anticipated that advanced reactors will be more standardized and thus generic simulators will be more practical. Such simulators would be the means for conducting procedure and design verification and validation called for by Chapter 18 of NUREG-0800 and possibly for conducting 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. 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 U.S., 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. It could be used to perform confirmatory studies of applicant submittals relative to issues such as staffing, control station design, procedures, other human factors, HRA, and digital I&C issues.

A 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 advanced 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 to collect data for HRA quantification will also be considered. Depending on the outcome of the study of existing facilities, additional resources may be needed to acquire simulator time or to develop a facility.

- **Analyzing Functions and Tasks**

Since the HFE Program Review Model described in Section IV.2.1.2.2 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 U.S. 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 in the licensing of advanced reactors. The use of such analytical models could enhance the efficiency and effectiveness of the licensing reviews and provide assurance of safe operations. The models would be used in a manner similar to thermal-fluid dynamics, fuel, and accident analysis codes and models. Data from human performance studies would be used to validate, populate, and maintain the code and would be used to assess applicant submittals.

- **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 for the staff. Not only does the humans' workload capacity have to be sufficient to fulfill their requirements during periods of normal operation, but also human capacity must 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 above could be used to develop and review staffing needs using a performance based approach, rather than developing prescriptive requirements. Such an approach would be consistent with the finding in NUREG/IA-0137, "A Study of Control Room Staffing Levels for Advanced Reactors," which states that "...decisions about control room staffing should be based on design features including function allocation, automation, integration, and plant-specific characteristics." This could result in a change to 10 CFR 50.54.

- **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 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). Innovative training concepts, such as embedded training and the use of virtual reality may also 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 on 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 be the requirements for simulation?  Can training and simulation be embedded into the operational setting?  The review of training and qualifications issues 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.

- **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.  The guidance needs to be assessed against advanced reactor systems, since advanced reactors will have computer-based or glass cockpit control rooms, and the procedures are likely to be computerized.  Guidance for the review of these systems should be developed to modify NUREG-0899 and SRP Chapter 13.

- **Human-system Interface**

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 was no validated criteria available, and there was not sufficient technical basis on which to develop the criteria.  Of special importance is guidance for high-level displays that is based on processed information with different types of processing, (e.g., functional decomposition and new display types, such as, flat panels and large screens).  This work could result in changes to or new review guidance.

#### IV.2.1.3.4 Human Factors Considerations – 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 function and task analysis is basic to staffing, training, human-system interface, 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 interface 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 or error forcing contexts.

### IV.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 margins, 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 for 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.

In advanced HTGR designs, the integrity of the coated particle fuel in its function as primary fission product barrier depends strongly on the maximum fuel temperatures reached during irradiation and in accidents. These fuel temperatures are predicted by reactor system calculations using a combination of codes and models for core neutronics, decay heat power, and system thermal hydraulics. So-called melt-wire experiments performed in Germany’s AVR reactor showed the unexpected presence of in-core hot spots, where maximum local operating temperatures were much higher than predicted with codes like those now being used by the PBMR developers. Moreover, the AVR’s true maximum local operating temperatures remain unknown due to measurement inadequacies in those experiments. For all advanced HTGR designs, significant uncertainties also exist in predicting the maximum fuel temperatures and vessel temperatures during heatup accidents. Such uncertainties relate to basic data like irradiation- and temperature-dependent thermal conductivities as well as the integral effects of variable local power densities with conductive, radiative, and convective heat transfer through the core and surrounding structures. Appropriate data measurements and system analysis tools will therefore be needed to support the staff's understanding and assessment of factors that govern fuel temperatures and uncertainties in relation to fuel integrity and HTGR safety margins.

Related research activities with analysis codes and data will also be needed for assessing the safety-related technical and policy issues associated with severe accidents and fission product release phenomena that differ dramatically from those in current and advanced LWRs. To meet research needs on all aspects of advanced reactor system analysis (i.e., nuclear analysis, thermal hydraulics, severe accidents and mechanistic release of fission products), the staff will seek to minimize costs and maximize benefits to the agency through active engagement in the planning and performance of domestic and international cooperative research efforts.

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 (a) conduct confirmatory analyses in the review of applicants' reactor safety analyses, (b) support development of the regulatory framework by assisting, for example, in the identification of safety-significant design basis and licensing basis events, and (c) 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 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.

### IV.2.2.1 Thermal-Hydraulic Analysis

#### IV.2.2.1.1 Thermal-Hydraulic Analysis – 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 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. Thermal hydraulic analysis is also used in the context of PRA to determine the best estimate of system states and thereby support analyses of the mechanisms and probabilities for system failures.

#### IV.2.2.1.2 Thermal-Hydraulic Analysis – Purpose

T/H analyses are typically used to assess what safety limits are needed and whether limits and margins such as fuel design limits are met, 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 T/Hs is necessary in order to confirm and quantify the expected safety margins of the proposed plants and to audit the applicant's calculations.

**(1) High-Temperature Gas-Cooled Reactors**

NRC staff has completed a preliminary survey of the analysis capabilities needed to model HTGR fluid flow and heat transfer in support of the staff's independent review of an HTGR safety analysis. Given the nature of HTGR transients, the preliminary findings indicate that 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. 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 transients in pebble bed reactors. The NRC analysis tools should be able to model all the turbo-machinery and passive decay heat removal systems, and 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 be 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 in this plan (see section on Severe Accident Codes and Source Term Analysis), the MELCOR code will be used in conjunction with TRAC-M and 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 use 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 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-HTGRs, such as Windscale, Magnox, and advanced gas-cooled reactors (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. 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, 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, adapting the necessary HTGR code features from GRSAC for use in 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.

Data will be 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. Past and ongoing HTGR research has been conducted at such reactor facilities as the Arbeitsgemeinschaft Versuchsreaktor (AVR), Thorium Hochtemperaturreaktor (THTR) in Germany, the High-Temperature Engineering Test Reactor (HTTR) in Japan, and the 10-MWe High-Temperature Reactor (HTR-10) in China. These and other experimental programs, such as the air-ingress tests done in the NACOK facility at FZ-Jülich and in a similar facility at JAERI as well as the pebble-bed fluid-flow and heat-transfer tests performed in the SANA facility at FZ-Jülich, provide significant sources of measured thermal hydraulic data. However, additional data is needed to investigate issues including the 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 with oxidation, and reactor cavity cooling. NRC 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 US 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 constitutive relationships will have to be 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 graphical user interface (GUI) to work with HTGR designs.

- Use 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.

**(2) Advanced Light-Water Reactors**

The T/Hs of ALWRs is relatively well understood because of the experimental and analytical efforts made to investigate the performance of conventional light-water reactor (LWR) systems. Advanced reactors, however, still pose significant challenges to engineering analyses 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.

Two advanced LWR systems are discussed: the AP-1000, and IRIS. Both designs rely on passive safety systems to ensure 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 T/H processes which they use.

The AP-1000 relies 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 AP-600 remain valid for many of the T/H processes that are important to the AP-1000. There are some T/H phenomena that are not well represented by previous tests for conditions expected during a hypothetical accident in an AP-1000. The major T/H issues for AP-1000 are primarily those T/H processes that are strongly dependent on the higher core steam production rate expected during an accident.

The major T/H issues for the AP-1000 include:

- **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 LOCA. 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 in-containment refueling water storage tank (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.

- **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 may also be needed in developing a database suitable for correlation and model development.

- **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 AP-600 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 this confirmatory data.

- **Direct vessel injection.** Flows from the core makeup tank and IRWST are injected directly into the downcomer in the AP-1000. This design feature is intended to reduce emergency core coolant (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 LWR with a power of up to 335 MWe. It makes use of passive safety systems to ensure 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 LOCA 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 T/H 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. This 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 – reactor coolant system 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 T/H codes for system response and containment response. Experimental data is needed to validate the codes used for the T/H 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.

### IV.2.2.1.3 Thermal-Hydraulic Analysis – Objectives and Planned Activities

**(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. Such development and assessment support will include: (a) adapting or building upon, where appropriate, selected GRSAC methods and data for use by TRAC-M (i.e., as an alternative to reinventing them for TRAC-M), and (b) comparing detailed GRSAC and TRAC-M results on reference HTGR transients and resolving the causes of any major discrepancies. An effort to modify TRAC-M to add the currently identified capabilities is being initiated at Los Alamos National Laboratory.

**(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.

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 T/H 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 simplified boiling-water reactor 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 to 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.

### (3) Planned NRC Research Activities

NRC needs an independent capability for HTGR T/H 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.

**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 spherical conduction models. An improved radiation model is also needed. Confirm and modify, as needed, the capability to model HTGR turbo-machinery. Confirm and modify, as needed, the capability to 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 T/H data and modeling needs for the HTGRs. The analysis will include issues and sequences raised in early analysis for the workshop. The deliverables will be an identification and ranking by, safety significance, of NRC data and modeling needs in the area of T/H for HTGRs.

- Develop Database: Development of needed 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.

**Advanced Light-Water Reactors.** 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 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 in order to support the regulatory decision-making process with safety-related information beyond that provided in 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.

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 T/H tools so that they can be extended to AP-1000 plant analysis. 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 T/H 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 T/H 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 reports describing the data, and a technical evaluation report describing T/H models and correlations developed from the data and needed to represent important AP-1000 processes. This work is ongoing at Oregon State University.

- AP-1000 code development and assessment: Assess TRAC-M for large and small break LOCA analysis in AP-1000. Ensure 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 large-break loss-of-coolant accident, small-break loss-of-coolant accident, 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 T/H 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. Will assume IRIS submittal in late 2003. Main objective is development of T/H tools to perform independent assessment and to confirm safety margin, verify success criteria, and provide input to the fuel and material analysis areas. Deliverables include IRIS plant input deck, workable TRAC-M code version for IRIS application.

- IRIS Helical steam generator (SG): 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 codes. 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 T/H conditions beyond those of interest to the applicant.

- 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 T/H 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 T/H conditions beyond those of interest to the applicant. The expected 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 steam generator tube rupture) analysis in IRIS. Ensure 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.

#### IV.2.2.1.4 Thermal-Hydraulic Analysis – Application of Research Results

This research will be applied to develop and demonstrate 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 T/H 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-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, validation of success criteria, input into PRA, and understanding of safety margins.

### IV.2.2.2 Nuclear Analysis

#### IV.2.2.2.1 Nuclear Analysis – 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.

#### IV.2.2.2.2 Nuclear Analysis – Purpose

The purpose of the research activities described in this section is to provide the nuclear analysis tools, data, and knowledge bases that will be needed in supporting and 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., JEFF 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, their known limitations and shortcomings in relation to modern nuclear data evaluations and processing techniques would call for extensive reevaluation in the context of advanced reactors and their fuel cycles and would continue to limit the implementation of modern nuclear analysis methods.

In response to a 1996 user need memorandum from NMSS, 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 JEFF-3 and JENDL-3. With the recently completed AMPX upgrades and continued improvements to the NJOY nuclear data processing codes opportunity and motivation now exist to produce and test state-of-the-art nuclear data libraries for use in the analysis of reactor safety, nuclear material safety, waste 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 T/H and 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.

### (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, <20% $^{235}$U) instead of the high-enrichment uranium (HEU, >90% $^{235}$U) 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 Materials Analysis.

Physics analysis issues unique to the GT-MHR relate mainly to the effects of burnable poisons, the presence of both 19.9% enriched "fissile" coated particles and unenriched "fertile" coated particles 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, and related aspects of TRISO fuel performance, include the following:

- **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.

- **Reactivity control and shutdown absorbers**

Depending on design details, the reactivity worths of in-reflector control and shutdown absorbers may be sensitive to tolerances in the radial positioning of the absorbers within the reflector-block holes. 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.

- **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 (i.e., from adding hydrogenous moderator to the undermoderated core), 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).

- **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., anticipated transient without scram (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.

- **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.

- **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.

- **Pebble fission power densities and temperatures**

The computational models may need to account for pebble-to-pebble burnup and power variations within nodes or meshes. Computational studies with higher-order methods, such as exact geometry, continuous-energy Monte Carlo (MCNP), may be used to investigate the distribution of power among assumed clusterings of pebbles with various burnups located in the core interior, in the inner-reflector mixing region, and near the outer reflector wall. 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 pebble thermal conductivity with neutron fluence.

- **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. Further computational studies may, therefore, be needed to establish technical insights on acceptable modeling approximations (e.g., mesh 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.

- **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 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).

- **Radionuclide decay before accident testing of TRISO fuel**

In understanding how out-of-reactor heatup and power-transient tests can be used to demonstrate the performance of TRISO fuels in reactor accidents, one should consider the potential effects from physical changes that can occur in the fuel during the time intervals between fuel irradiation and testing. Such physical changes would include those arising from the decay of short-lived fission products and actinides and from other time- and/or temperature-dependent processes (e.g., chemical reactions, material cooling, creep, annealing, precipitation, condensation, diffusion, permeation, migration) that could affect the mechanical loading and effective strength of particle coatings under the respective simulated or actual accident conditions. Specific analyses of nuclide generation, depletion, and decay will therefore be needed for evaluating how radioactive decay changes the fuel's inventory of important actinides and fission products (e.g., those that potentially affect gas pressure and layer strength in the coated particles) during the time intervals between fuel irradiation and out-of-reactor accident testing. (Note: This nuclear analysis issue relates directly to fuel analysis issues described in Section IV.3.2.)

- **Physics of TRISO fuel irradiation in test reactors versus HTGRs**

The extensive use of various test reactors for the irradiation testing of HTGR TRISO fuels raises questions about the nonprototypicality of the neutron energy spectra, accelerated fuel burnup rates, and fuel temperature histories in the test reactors. Reactor-specific calculations of neutron fluxes and nuclide generation, depletion, and decay should therefore be performed to provide a basis for analyzing the sensitivity of computed fluences and fuel nuclide inventories to the neutronic differences between the test reactors and HTGRs. Of interest are the potential effects of such differences on TRISO fuel performance (i.e., fission product retention) under normal and accident conditions. Such differences therefore include the differences in irradiation temperature histories, burnup rates, and neutron energy spectra that result in different neutron fluences, different rates of plutonium production and plutonium fission versus uranium fission, and, thus, different yields of important fission products. It is known, for example, that $^{235}U$ and $^{239}Pu$ give substantially different yields of various fission products that potentially affect TRISO fuel performance. (Note: This nuclear analysis issue relates directly to fuel analysis issues described in Section IV.3.2.)

**(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:

- **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.

- **Fuel depletion validation.** The available experimental database for validating LWR fuel depletion analysis methods consists largely of destructive radio chemical 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.

- **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 neutronics significance in IRIS than in shuffled PWR cores. Neutronics 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.

- **Decay heat power.** Due to depletion modeling issues and the apparent shortage of available radio isotopic or calorimetric validation data applicable to the IRIS fuel designs at high burnup, specific technical guidance will likely be needed on accepted methods for computing decay heat sources with appropriate consideration of validation uncertainties.

Additional 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.

#### IV.2.2.2.3 Nuclear Analysis – 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.

**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 pre-application 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 use 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, and (4) MELCOR code model development for modular HTGRs.

- 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 conjunction with AMPX-processed nuclear data libraries in the performance of exploratory studies and preparing design-specific nodal physics data tables for input to the NRC's PARCS spatial kinetics code.

**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, models, and related code development and analysis tasks (e.g., involving the PEBBED and MCNP codes).

- 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 the GROG experiments and the ASTRA pebble-bed 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 Los Alamos National Laboratory, 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 coordinated research project (CRP) on safety performance of HTGRs. Such activities include code-to-code comparisons as well as experimental benchmarks taken from various sources such as recent and planned benchmark measurements at HTR-10 in China, HTTR in Japan, and ASTRA in Russia. In addition, there may be opportunities in this or other forums to pursue 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, and the new International Reactor Physics 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 NRR and NMSS (SFPO and DWM), 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.

To fill 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, code-to-data benchmarks, and code-to-code comparisons, conducted by the IAEA, the European Commission, and OECD/NEA.

**Planned NRC Research Activities**

Listed below are the planned research activities pertaining to the nuclear analysis issues described previously:

**(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.

**(2) Familiarization with pre-existing codes and methods for core neutronics and decay heat in (1) PBMR, (2) GT-MHR, and (3) IRIS.** In coordination with pre-application review activities, gain familiarity with pre-existing reactor neutronics codes or, where available, the reactor neutronics codes, decay heat algorithms, analysis assumptions, validation data, and uncertainty treatments that are being used or intended for use by pre-applicants in their 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.

**(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/Monte Burns, SCALE/NEWT/SAS2D, WIMS/MONK, Venture 2000, PEBBED), and available applicant codes where needed, to perform exploratory and scoping analyses on selected issues as described in this chapter. The general approach entails the initial use of higher-order methods, like MCNP/Monte Burns, followed by progressive approximations to understand basic and detailed phenomena and to explore effects of the necessary approximations and assumptions used with more practical analysis methods (e.g., few-group diffusion theory versus multi-group or continuous-energy transport theory) and models (e.g., smeared coarse-mesh or nodal material-geometry versus exact material-geometry). 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.

**(4) Preparation and testing of spatial kinetics models of (1) PBMR, (2) GT-MHR, (3) IRIS, and (d) AP-1000.** In collaboration with NRR staff: (a) develop PARCS input models that are compatible with the coupled TRAC-M models, and (b) use appropriate lattice physics and depletion analysis tools with state-of-the-art cross section libraries (see item 1) to prepare the design-specific nodal data tables needed for performing spatial kinetics analyses with the PARCS code, coupled with the TRAC-M code.

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

**(6) Validation for depletion and decay heat in (1) PBMR, (2) GT-MHR, and (3) IRIS.** Analyze available information on 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 at ORNL under RES sponsorship, to help assess their applicability to the respective fuels and operating parameters and to help prioritize further data needs and quantify remaining validation uncertainties. Participate in cooperative programs for new experimental data as well as code-to-data and code-to-code benchmarking activities.

**(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.

#### IV.2.2.2.4 Nuclear Analysis – 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 investigation and analysis 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 and the factors that govern them (e.g., absorber worths, reactivity feedback effects).  Especially important in this context is the development of state-of-the-art master cross section data libraries.  As noted above, the resulting data libraries will replace the currently used pre-1990s data libraries, whose known limitations and shortcomings would otherwise limit the implementation of modern nuclear analysis methods and require extensive reevaluation in the context of advanced reactors and their fuel cycles.  The new master cross section libraries will play a fundamental role in all nuclear analysis activities for reactor safety, nuclear materials safety, waste safety, and safeguards and will be generically applicable to all systems associated with conventional and advanced reactors.

### IV.2.2.3 Severe Accident and Source Term Analysis

#### IV.2.2.3.1 Severe Accident and Source Term Analysis – 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 substantial quantities of radioactive fission products (FPs) into the environment.  The ability to model progression of severe accidents and estimate releases of FPs into the environment is required to quantify risk and to address severe accident issues.  NRC has developed several codes to model severe accidents.  These codes have been used to develop and improve the NRC regulations dealing with severe accident issues, such as 10 CFR 50.67, "Accident Source Term."

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.  MELCOR is chosen as the NRC consolidated severe accident code which can model most aspects of a severe accident including thermal hydraulics, core melt progression, FP transport in the reactor system and containment.  For LWRs, many experiments (U.S. and international) have also been carried out in support of development of a fundamental understanding of the phenomena of severe accident and FP transport.  The recent NRC focus in severe accident has included upgrading MELCOR and benchmarking it against the more specialized severe accident codes and experimental results.

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

#### IV.2.2.3.2 Severe Accident and Source Term Analysis – Purpose

Accidents that lead to FP 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 FP release to the containment and subsequently to the environment.

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, data and accident progression analysis codes 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 HTGRs and advanced designs such as IRIS, both the types of sequences and the process by which FPs may be released from the fuel will be different than current generation LWRs. As a result of diffusion during normal operation, in HTGRs for example, rupture of coated fuel particles as a result of accidents, and vaporization during high-temperature degradation of the fuel, FPs may be released.

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 and technical capability in the area of FP transport and behavior during high temperature fuel degradation is needed in order to assess the risk from HTGR operation. Because FPs released from the fuel are transported through the primary system and containment predominantly as aerosols, the offsite releases and offsite radiological consequences may be significantly reduced by FP deposition in the primary system and containment. Aerosol deposition occurs through a variety of mechanisms such as gravitational settling, thermophoresis, and diffusiophoresis. Therefore, research activities will focus on FP transport and behavior in the primary system and containment or other structural buildings.

#### IV.2.2.3.3 Severe Accident and Source Term Analysis – Objectives and Planned Activities

MELCOR has most of the capabilities needed to analyze beyond design-basis accident issues for HTGRs. However, modifications to MELCOR are needed to model these reactors, because of the different fuel design and the different reactor internal structure design. Proposed modifications are described below, together with an activity to assess MELCOR against available experimental data and other codes.

**Modify Code to Incorporate Available Models/Data**

Modify code to incorporate available models/data, and enable application to HTGRs, which is needed to model the FP release from the core and deposition in the reactor coolant system and containment.

- **Extend FP release models**

  Extend FP 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 burnup on FP release and transport should be included.

- **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.

- **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.

- **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.

**Evaluate the Need for Additional FP deposition/transport experiments and models.**

When model implementation in the MELCOR code is completed, assess the code against available experiments. Also, prepare input decks for selected advanced reactor designs, and demonstrate code capabilities for selected performance scenarios.

**Assess the Code Against Available Experimental Data and Other Codes.**

To achieve this objective, a literature review will be performed of HTGR experiments on FP release during high temperature fuel degradation and deposition in the primary system and containment under accident conditions. Because FP 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.

**Apply Code to Specific Advanced Reactor Design.**

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 FP release to the containment and subsequently to the environment.

The results of the database work will be used to develop and assess FP 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.

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. For advanced light water reactors, the evolution of severe accidents and source terms will be similar to the current generation of plants. However, a major issue for AP-600 design certification is related to in-vessel retention of melt, and this issue will be addressed for AP-1000 as well.

- **In-vessel melt retention**

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 to ensure the capability to assess sequences that include this phenomena.

For AP-1000, if in-vessel melt retention cannot be assured and in the event of reactor vessel breach, ex-vessel severe accident phenomenological loads on reactor containment resulting from ex-vessel steam explosions, direct containment heating, core concrete interactions, and hydrogen combustion have to be assessed.

For IRIS, the affect of high burnup fuel on the evolution of severe accidents and source terms should be examined. It is envisaged that the MELCOR modeling of FP transport through the reactor system has to account for unique features (e.g., helical tubes) of the design.

**Related NRC and International Research Activities**

As discussed under the "Fuel Analysis" section, in the early 1980s, the Federal Republic of Germany (FRG) has performed irradiation experiments for TRISO coated particle design with low enriched $UO_2$. The experiments included aspects such as accident simulation testing up to 1600$^{O}$C. The FRG fuel irradiation testing research included FP transport in the fuel kernel, 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. However, these experiments did not cover the FP behavior for high burnup fuel (e.g., 80 GWd/t for PBMR) envisaged for current gas cooled reactors.

The International Atomic Energy Agency (IAEA) had also published many reports of meetings of technical specialists working in the area of HTGR fuels HTGR fuels utilizing CFPs. Meeting topics have included, FP 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).

Since 1985, the Japanese Atomic Energy Research Institute (JAERI) conducted an HTGR research and development program in cooperation with the DOE under a DOE-JAERI memorandum of agreement. 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. In Japan, the reference HTGR fuel involves hexagonal prismatic graphite blocks utilizing graphite fuel rods containing fuel compacts with TRISO CFPs. 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.

The European Commission (E.C.) 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. Additional irradiation experiments on German archive fuel and GA compacts fabricated using a new manufacturing process are expected to begin in CY 2002. The irradiation experiments will be followed by accident heat up simulations with FP release measurements and post irradiation examinations.

The Massachusetts Institute of Technology (MIT) has established a high temperature pebble bed reactor research project and will study migration of FPs (silver) through coatings, and chemical attack on silicon carbide by palladium.

For AP-1000, the larger debris mass and power density for the 1000MW core will impact the effectiveness of external reactor vessel cooling (ERVC) concept. The ERVC analysis was performed for AP-600 (documented in DOE/ID-10460). Since the AP-600 design certification, the OECD-RASPLAV and OECD-MASCA projects have performed experiments on in-vessel melt behavior, and the findings from these experiments will have to be accounted for in the analysis of external reactor vessel cooling.  Also for AP-600, experiments were performed at the Pennsylvania State University and at the University of California at Santa Barbara (UCSB) on critical heat flux (CHF) to study the heat removal in the reactor lower head under flooded conditions.  Recently, UCSB has performed additional CHF testing pertinent to AP-1000 (higher heat flux due to higher power density melt) for the industry.

For HTGRs, NRC has initiated:  a review of past experiments and studies performed; MELCOR development and assessment for HTGRs, including the use of GRSAC to support the development and assessment effort, and a TRISO Fuel Particle PIRT .  For AP-1000 design certification, data and findings from new experiments performed since AP-600 design certification are being used to assess the in-vessel retention strategy for AP-1000.  In addition, evaluation of the applicability of conclusions from the AP-600 severe accident phenomena review to AP-1000 will be performed.

#### IV.2.2.3.4 Severe Accident and Source Term Analysis – Application of Research Results

This research will be applied to develop and validate analysis tools needed to evaluate the behavior of the new reactor designs under postulated accident conditions and any resulting releases and transport of radioactive fission products within and outside the reactor system boundaries.  This information will be critical in supporting resolution pathways to policy and safety issues, specifically with respect to containment versus confinement issues, and emergency preparedness.

## IV.2.3 Fuel Analysis

This section addresses research activities for both HTGRs and ALWRs.

### IV.2.3.1 Fuel Analysis – Background

HTGRs, such as the PBMR and GT-MHR have unique safety features and 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 (i.e., fuel pebbles, fuel compacts) that comprise the fueled core.  The TRISO CFPs provide the principal safety barrier and primary containment function against release of FPs to the environment during normal operation, design basis accidents and accidents beyond the design basis.  FP release is the sum of initial CFPs defects and heavy metal contamination from manufacture; CFP failures that occur during normal plant operations, including anticipated operational transients; and CFP failures that occur during design basis accidents or accidents beyond the design basis (i.e., “severe” accidents).

HTGR applicants are expected to propose that the accident source-term be based on models and methods that mechanistically predict FP release from the fuel.  Should this be the case, 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, applicants will also likely propose that HTGR plants utilize a 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, largely, on the applicant’s capability to confirm, as well as the NRC’s capability to confirm, fuel FP 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 FP 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, design-basis accident conditions, and postulated accident conditions beyond design basis.  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 fuel residence time in the core.  The key factor affecting fuel particle performance during an accident (following the prior degrading effects of the operating conditions) is the peak particle temperature during the accident.  Temperature increases can occur due to heatup events, which are 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) the CFPs.

To predict CFP performance, and a deterministic approach to the source term, capabilities in a number of interfacing technical areas will be needed.  These include: (1) nuclear analysis for 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; and (2) T/H analysis of normal operating core temperature distributions, accident core temperature distributions, and core temperature and flow distributions (for fuel oxidation during postulated air intrusion events).  The FP release rates from the fuel during normal operation and postulated accidents are key inputs to the accident source term calculation which is addressed in another part of the plan.

Additionally, it will be essential to qualitatively and quantitatively understand the design margins and safety margins to large increases in CFP failure rates and large increases in FP release. These margins will need to be known for normal operation, design-basis accidents and, potential accidents beyond the design basis.  The design margin should be demonstrated by the applicant.  The fuel safety margin is on top of the fuel design margin.  The safety margin involves the margins to failure for conditions that exceed the fuel design conditions (e.g., fuel design specifications, fuel manufacturing specifications, fuel maximum operating temperature, fuel maximum burnup, fuel maximum fast fluence limits, fuel maximum particle power and residence time).  It is expected that safety margin aspects will be a developed by an applicant. However a complete assessment of the safety margin would likely require NRC research since HTGR designers and applicants generally oppose testing fuel to conditions that go substantially beyond the licensing basis.

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 as discussed later in this section.

Additional insights that bear on the extent to which additional NRC regulatory research is needed in the area of HTGR fuel performance analysis is provided below. These paragraphs recognize the considerable worldwide research on HTGR fuels with TRISO CFPs that has been conducted over the last 30 years or is currently ongoing. NRC HTGR fuel performance analysis research should capitalize on this body of work to establish the infrastructure of knowledge, data, and tools needed to support HTGR fuel-related policy decisions and license application reviews. The existing research provides a base and context for deciding which NRC research should be pursued to fill infrastructure gaps without duplicating previous applicable reference work.

With respect to the ALWRs, there is a need to conduct confirmatory research to establish an information base related to the long-term performance and behavior of the fuel assemblies and control rods under temperatures and flux levels expected under normal operating and accident conditions in the IRIS design proposed by Westinghouse and its partners. Furthermore, various performance parameters which would be affected by temperature, radiation (e.g., burnup, maximum fluence), and oxidation in the event of transients or accidents, need to be examined. NRC should consider conducting studies to obtain confirmatory data to assess behavior at burnups greater than 90 MWd/t.

The first IRIS core will employ standard <5% $UO_2$ fuel and standard PWR fuel assembly design. This represents current, proven and licensed fuel technology, therefore no licensing issues related to fuel are foreseen by Westinghouse and its partners. A path forward for future fuel cycle enhancement (extending the core life to 8–10 years by increasing the fissile content to about 8% enrichment) has been envisioned, but it will not be part of the initial IRIS design for licensing.

- **Past Research**

The design of HTGR fuels with TRISO CFPs has evolved empirically over the last 4 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 used in cylindrical fuel compacts at Peach Bottom Unit 1, 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 CFP irradiation performance and accident performance. The fuel and materials development efforts included testing of a variety of CFPs in prismatic fuel elements which were involved in 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 and fuel elements with BISO were 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$^{O}$C limit for pebble fuel elements with TRISO CFPs. The concept was that TRISO CFP failures would not occur until well above1600$^{O}$C, while the peak transient fuel temperature for a modular HTGR design would not exceed 1600$^{O}$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), 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 in irradiation testing programs.

Until recently, the 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 the1990s. Meeting topics have included HTGR fuel development (1983), FP 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. Recently, the IAEA has taken steps to establish a new international coordinated research project (CRP-6) on HTGR fuel. The areas identified for the CRP include: fuel performance data; fuel performance modeling and data characterization; fuel operating experience, fuel irradiation and accident condition testing, and fuel licensing issues. Fuel fabrication technology for quality and performance may also be included in CRP-6.

Since 1985 the Japan Atomic Energy Agency Research Institute (JAERI) conducted HTGR research and development 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 exchanges 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 U.S. 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.

- **Current Research**

The U.S. DOE has established an Advanced Gas Reactor Fuel Development and Qualification Program. The program includes fabrication technology development, irradiation testing, and accident condition testing. The qualification of TRISO CFP fuel types will be based on the German fuel fabrication process. The irradiation testing program in the near-term involves German archive fuel with German manufactured TRISO CFPs. The near-term irradiations would develop performance data for the reference German fuel for test conditions exceeding those previously examined. The tests will determine TRISO particle fuel performance for the more demanding design conditions (e.g., higher operating temperatures) of modern modular HTGR designs relative to the earlier German pebble bed reactor designs. The test program will explore the failure margins of TRISO particle fuels with test conditions that are well beyond the conditions associated with the fuel design basis.

The E.C. is currently sponsoring an approximately $16M, 4-year research program on HTGRs. The EC HTR program includes a project on fuel technology (HTR-F). The objectives of the HTR-F are to: re-establish the know-how that existed in the past in the areas of fuel design and fuel fabrication, assess the performance of fuels with TRISO CFP at very high burnups, develop an analytical code for predicting TRISI CFP behavior under irradiation, and retrieve and evaluate data from past fuel irradiation experiments with the aim of constructing a searchable fuel database. Irradiation experiments on German archive fuel and GA compacts fabricated using a new manufacturing procedure are expected to begin in CY 2002. The irradiation experiments will be followed by accident heat up simulations with FP release measurements and post irradiation examinations. The purpose of the German archive fuel experiments is to develop fuel performance data for reference TRISO fuel for conditions which significantly exceed the previous test conditions associated with the earlier German fuel qualification testing programs. The conditions involved are far more demanding and go beyond the design conditions expected for a modern modular pebble bed reactor. The E.C. 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 an analysis tool for assessing particle behavior under irradiation and accident conditions. 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. As of early CY 2002, power escalation of the HTR-10 had not yet been authorized.

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 FPs through coatings and chemical attack of the SiC coating by palladium. 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 and 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 will 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 elements made with the German fuel manufacturing process perform satisfactorily in conditions simulating 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 after CY 2005.

#### IV.2.3.2 Fuel Analysis – Purpose

The purpose of 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 and IRIS fuels.  This infrastructure is needed to support the staff's review of a PBMR, GT-MHR, or IRIS application.  The plan for establishing the infrastructure capitalizes on worldwide research that has been conducted on CFPs over the last 30 years.

Fuel vendors and applicants are expected to demonstrate that significant fuel failures do not occur even for operating and accident conditions that exceed the design basis.  However, NRC research will be needed to fully understand and quantify the margins to significant increases in TRISO particle fuel failures for normal operation, design-basis accidents, and accidents beyond the design basis.  The research plan will provide the staff with the requisite level of knowledge in the areas of fuel design, manufacture, operational performance, and accident performance, necessary to independently and authoritatively assess the applicant's technical and safety basis for fuel quality and safety performance.  Analytical tools will be developed and validated to enable the staff to independently predict fuel performance (including CFP failure and FP release) during normal operation, design-basis accidents, and potential severe accidents.  Research will provide the staff with an independent capability to calculate TRISO particle fuel source term for normal operation, design-basis accidents, and potential severe accidents.

With respect to the IRIS design, the IRIS reactor "safety by design" approach attempts to first eliminate the possibility of accident sequences from occurring, and second, to reduce the severity of consequences and/or the probability of occurrence.  The integral reactor vessel configuration is a beneficial  layout for implementing this approach.  Because the integral reactor vessel contains the steam generators, reactor coolant pumps, and the pressurizer, there is no external large loop piping and therefore eliminates the possibility of a large LOCA.  In addition, the IRIS integral RV configuration results in a tall vessel with elevated steam generators and a low pressure drop flow path, which provides increased natural circulation capability and intrinsic mitigation of Loss of Flow Accidents (LOFAs).  The integral RV also provides a large inventory of water above the reactor core, which slows the reactor response to transients and postulated small LOCAs.

However, main steamline break or ATWS may be more severe (in terms of return to power) than in current PWRs.

#### IV.2.3.3 Fuel Analysis – Objectives and Planned Activities

The overarching objective of the NRC research in the 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 and ALWR fuels.  The specific objectives are as follows:

- **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 FP release;

- Provide the data which explores the limits (i.e., margins) of fuel performance and FP release for conditions that are well beyond the design basis for parameters which are important to the fuel safety margins. These conditions involve fuel operating temperature, maximum fuel accident temperature, fuel oxidizing environment, fuel burnup, energy deposition and deposition rate in the fuel (due to reactivity accidents), beyond those that are expected to be examined by the fuel vendor or applicant.

- 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, quality assurance program, data analysis methods), and

- Provide data for use in developing/validating NRC analytical models and methods.

• **NRC fuel analytical model and methods development**

The purpose would be to:

- Independently evaluate HTGR fuel behavior, including CFP failure, FP release and margins of safety, and

- Evaluate the effects of variations in irradiation service conditions, and uncertainties (i.e., sensitivity studies).

• **NRC fuel fabrication technology Expertise**

The purpose would be to:

- Provide NRC staff with in-depth knowledge of contemporary HTGR fuel fabrication, including the critical process parameters, critical product parameters and quality control measures that are vital to achieving the required fuel quality and fuel characteristics that will provide the required fuel performance over the life of the plant's fuel supply.

**(1) HTGR Fuel Irradiation Testing Plan**

**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 FP 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$^{O}$C/hr up to a set temperature (e.g., 1600$^{O}$C, 1700$^{O}$C or 1800$^{O}$C) and then holding the fuel at the set temperature for several hundred hours while FP release measurements are taken. The results of ramp-and-hold tests up to 1600$^{O}$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$^{O}$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 and moisture intrusion for a postulated heat exchanger tube failure with the reactor helium pressure falling below the heat exchanger tube pressure. There have been experiments on unirradiated HTGR fuel in air and water at HTGR accident temperatures. These experiments have involved measurements of fuel oxidation due to air or moisture impurities in helium during fuel experimental irradiations. However, few experiments have been conducted 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. The staff has been told that the PBMR design does not have a potential for such large rapid reactivity events. Further, the GT-MHR control rods, which are located in the central core (fueled) region, are expected to incorporate engineered safety features to prevent a failed drive housing from rapidly and fully ejecting a control rod from the core. For these reasons, PBMR and GT-MHR applicants are expected to claim that large and rapid reactivity insertion events are not within the licensing basis and that design specific fuel testing is not needed. Accordingly, 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 by the NRC.

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 FP release for fuel operations beyond the maximum allowed operating temperature (e.g., 1250 $^{O}$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.

**Plans**

It is assumed that HTGR applicants/vendors will conduct all fuel testing necessary to support their license applications. Such fuel testing would be expected to address all significant aspects of the licensing basis. These aspects are: a sufficient range of parameters to cover uncertainties and variations; the plant-specific service conditions (e.g., core maximum operating temperature, fuel design burnup, fast fluence, particle power) of the PBMR and GT-MHR, a sufficient quantity of fuel elements and CFPs to establish a sufficient statistical database, and the range of potential CFP failure mechanisms and performance factors (e.g., FP release) applicable to or potentially applicable to the licensing basis. It is also expected that such testing will use fuel 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 may be met with German or U.S. archive fuel or pre-production fuel.

It is important that the NRC staff and contractors have expertise on the proper conduct of HTGR fuel irradiation experiments, including a thorough understanding of sound 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 HTGR fuel irradiation and accident simulation program methods, quality assurance practices, etc.

The proposed 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 HTGR production fuel for demonstration or prototype HTGR plants that may be built in the U.S. Table 1 at the end of this section summarizes a proposed irradiation testing plan for German archive pebble fuel. Table 2 summarizes a proposed testing plan for PBMR production fuel, and Table 3 summarizes a proposed testing plan for GT-MHR production fuel compacts. Testing of the German archive fuel will provide information on the acceptability of traditional testing methods, insights into adequacy of vendor testing programs and information on operational and accident condition safety margins for reference TRISO particle fuel types. These test plans could be implemented in connection with the following cooperative agreements, and any proposed testing would not duplicate but will capitalize on testing performed by DOE.

This plan assumes NRC participation in a cooperative HTGR fuel test program with the 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 and data that can be used to confirm an applicant's fuel performance analysis. Further, staff knowledge of fuel testing will be increased for later use in the review of an applicant's fuel qualification program documents.

This plan also assumes NRC will participate in the E.C. research 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 FP release measurements and post irradiation examinations. The NRC will also support the retrieval of data from past HTGR experiments with the aim of constructing a searchable fuel database.

### (2) HTGR Fuel Analytical Model and Methods Development

**Issues**

The body of irradiation and accident simulation (heating) tests have enabled the development of analytical tools for evaluating 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 FPs with the SiC layer leading to SiC degradation, 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 designers to help quantify margins and by safety analysts in calculating mechanistic source terms.

**Plans**

The NRC, as a first step, will review ongoing research aimed at developing tools for performing mechanistic analyses of HTGR fuel performance and existing HTGR fuel performance analysis models and methods. 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 FP 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 tests. 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 that may not be simulated in the fuel irradiation testing programs.

### (3) HTGR Fuel Fabrication Process Expertise

**Issues**

A comparison of the irradiation data for German made TRISO coated particle fuel with U.S made TRISO coated particle fuel shows that the gas release rate (i.e., particle failure rates) during irradiation for U.S. fuel is three orders of magnitude higher than the German fuel. A recent critical INEEL study of causes for these differences confirmed the long held view that differences in the process parameters used for applying the individual coating layers of the TRISO coated particle is a major factor in irradiation performance and accident condition performance. The U.S. fuel met the established specifications for the measurable fuel particle layer physical, material and chemical characteristics (e.g., thickness, density, strength, impurities) which were consistent with design and safety requirements and were equivalent to the German fuel. However, key differences in fabrication processes were found to result in critical differences in the layer characteristics such as micro structure, layer bonding and layer anisotropy. The differences in particle characteristics resulted in significant differences in the in-reactor (irradiation and accident) behavior of the two fuel types. The importance of fabrication process was recognized in Germany and was included in the fuel manufacturing specification along with the product specifications.

The regulatory oversight measures to ensure the requisite characteristics and consistent quality of the fuel (supply) over the term of a license of an HTGR is a significant safety issue and a potential Commission policy issue. Measures might include fuel fabrication technical specifications and fabrication facility inspections. Other additional or alternative measures might involve reactor coolant activity monitoring and periodic end-of-life fuel accident simulation testing. These alternatives can have significant technical and regulatory advantages and disadvantages, however. An additional policy issue is whether a plant can be licensed before fuel testing is complete, which links to whether the correct and complete fuel manufacturing process and product specifications have been identified and specified. Research activities are included in the advanced reactor research plan to establish the staff's knowledge and expertise of the critical fuel fabrication product, process parameters, and quality control measures that are vital to ensuring the requisite fuel characteristics, quality, and performance.

**Plans**

A major research element of the EU HTR-F is to re-establish know-how on TRISO coated particle fuel fabrication. The research includes both fabrication of fuel kernels and coatings.

The NRC is seeking to enter into a cooperative agreement with the HTR-F. The project includes fuel fabrication element aimed at re-establishing expertise in the fabrication of fuel kernels and particle coatings. It is expected that the HTR-F project will identify the critical process and product attributes and the necessary quality controls to fabricate HTGR fuels with the required quality and characteristics needed to maintain consistently good fuel performance over the life of the plant. An NRC cooperative agreement with the HTR-F would be expected to provide the NRC with access to the information on fuel fabrication technology developed by the HTR-F project. The NRC should also endeavor to utilize technical information exchanges with foreign organizations having expertise in TRISO particle fuel fabrication (e.g., China, Japan) to obtain information and to develop expertise on the fabrication of TRISO particle fuels.

### (4) IV.2.3.3.4 IRIS Fuel Analytical Model and Methods Development

**Issues**

For the IRIS fuel design, the research will provide independent data and code analyses to support regulatory decision-making for cladding performance and fuel response to licensing basis accidents.
To be able to achieve these objectives, research related to the following issues needs to be considered:

- Higher projected cladding temperatures at full power (for maximum power rod and average power rod). For example, the average assembly outlet temperature is 626 F, which could impact corrosion, creep, and axial growth of cladding.

- Erbium as an integral burnable absorber in the UO2 fuel pellets.

- Excessive cladding corrosion when core life becomes longer than 4 years.

**Related NRC and International Research**

Work should be extended at Argonne National Laboratory to measure mechanical properties for advanced Zirconium-Niobium alloys to be used in IRIS.

Work on FRAPCON and FRAPTRAN is being carried out at Pacific Northwest National Laboratory, but additional effort would be needed to extend code assessment to burnups greater than 75 GWd/t and incorporate new cladding properties.

Additional work at Halden is needed on behavior of high burnup fuel, including fuel thermal conductivity fission gas release, absorber materials, and cladding corrosion for the extended burnups and new alloys.

#### IV.2.3.4 Fuel Analysis – Application of Research Results

The intended safety characteristic of the TRISO coated fuel particles (CFP) within fuel elements is to provide the principal barrier and the primary containment function against the release of FPs to the environment during normal accident conditions. Given the significance of the fuel barrier for the HTGR designs, the fuels research program will be used to provide insights on FP source term for normal operation and accident conditions. The source term information is needed for systems analysis, accident analysis, and consequence analysis and will play a significant role in supporting regulatory decisions in a number of areas, including containment/confinement and evacuation planning. The fuels analysis will also provide technical basis and criteria for HTGR fuel qualification testing, and support regulatory decision-making on fuel performance, including the acceptability of an applicant's fuel irradiation program.

# Table 1. German Archive 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 Pebble | | | | | | N/A |
| 2 | Archive Pebble | | | | | | N/A |
| 3 | Design Max Fuel Temp+Ramp Hold | ------Accel----- | -------Accel------ | ------Accel-------- | ------Accel-------- | **Δ** | 1600$^{O}$C Ramp Heatup |
| 4 | Design Max Fuel Temp + Acc Temp | ------Accel----- | -------Accel------ | ------Accel-------- | ------Accel-------- | **Δ** | 1600$^{O}$C Accid Simulation |
| 5 | Design Max Fuel Temp+Ramp Hold | -----Accel------ | -------Accel------ | ------Accel-------- | ------Accel-------- | **Δ** | 1800$^{O}$C Ramp Heatup |
| 6 | Design Max Fuel Temp+Real Time | -- Real-Time-- | ----Real-Time-- | ---- Real-Time--- | --- Real-Time---- | **Δ** | 1800$^{O}$C Ramp Heatup |
| 7 | Design Max Fuel Temp+50$^{O}$ C | -----Accel------ | ------Accel------- | ------Accel-------- | ------Accel-------- | **Δ** | 1800$^{O}$C Ramp Heatup |
| 8 | Design Max Fuel Temp+Air Ingress | -----Accel------ | ------Accel------- | ------Accel-------- | ------Accel-------- | **Δ** | 1600$^{O}$C+ Air Ingress |

**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** |
|---|---|---|---|---|---|---|---|
| | | 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$^{O}$C Ramp Heatup |
| 4 | Design Max Fuel Temp+50$^{O}$ C | -----Accel------ | ------Accel------ | -----Accel------- | -----Accel------ | **Δ** | 1800$^{O}$C Ramp Heatup |
| 5 | Design Max Fuel Temp+20K BU | -----Accel------ | ------Accel------ | ------Accel------ | ------Accel----- | -----Accel------**Δ** | 1600$^{O}$C Ramp Heatup |
| 6 | Design Max Fuel Temp+Real Time | -- Real-Time-- | -- Real-Time--- | ---Real-Time--- | -- Real-Time-- | **Δ** | 1800$^{O}$C Ramp Heatup |
| 7 | Design Max Fuel Temp+Air Ingress | -----Accel------ | -----Accel------- | -----Accel------- | -----Accel------ | **Δ** | 1600$^{O}$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$^{O}$C Ramp Heatup |
| 12 | Design Max Fuel Temp +Acc Temp | -----Accel------ | ------Accel------ | -----Accel------- | -----Accel------ | **Δ** | 1600$^{O}$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).

## Table 3. GT-MHR Production Fuel Irradiation Tests

| # | Irradiation Purpose | Burnup Increment (GWd/t) 0 to 25 | 25 to 50 | 50 to 75 | 75 to 100 | 100 to 125 | Safety Test Δ |
|---|---|---|---|---|---|---|---|
| 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).

## IV.2.4 Materials Analysis

### IV.2.4.1 Materials Analysis – 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. The licensing approach for HTGRs used by NRC to independently confirm design and support safety evaluations relies more heavily on the use of PRA. Information from the materials research area is needed for conducting PRAs. 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, through its on-going code activities, is addressing 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, 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 inservice inspection (ISI) program and for continuous monitoring. First, more components are enclosed in pressure vessels making access for inspection difficult. Second, there are longer operating cycles between scheduled, short-duration, refueling outages during which ISIs can take place. These two suggest a 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.

If periodic ISIs are found to be ineffective for maintaining safety, the NRC may have to require the use of continuous online monitoring techniques for structural integrity and leakage detection.

#### IV.2.4.2 Materials Analysis – Purpose

The NRC staff needs to develop independent research and expertise in the high temperature materials area for HTGRS to evaluate and establish a technical basis regarding the safety of advanced reactors. 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 PRAs 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 alloys; (6) treatment of pipe as a vessel; (7) degradation by carburization, decarburization, and oxidation of metals in HTGRs; (8) issues related to inspection of HTGR and ALWR reactor components; (9) performance and degradation of graphite under high levels of irradiation; (10) lack of knowledge for prediction of irradiated graphite properties from as-received virgin graphite properties; (11) lack of data on oxidation kinetics of reflector grade graphite, fuel pebble matrix graphite, and graphite dust; (12) applicability of graphite sleeve properties to large block graphite properties; and (13) 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.  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 and to gain confidence and understanding of defense-in-depth.

### (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 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 can be 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.

In HTGR designs the connecting pipe which carries hot helium from the core to the power conversion system (PCS) is treated as a vessel because this pipe is designed, fabricated, and inspected to the same rules as a reactor pressure vessel. The consequence of this assumption is that a design basis double ended break is not considered for the connecting pipe and therefore, no mitigating systems are incorporated in the design. Considering this pipe as a vessel will require further investigation, because the pipe is of much smaller diameter and therefore, much thinner wall than a reactor pressure vessel designed to the same working pressure. If an unexpected degradation mechanism should initiate in the pipe, because of the thin wall, it can propagate through the wall in a relatively short time and possibly not be detected by ISI. Conversely, if an unexpected degradation mechanism were to initiate in a pressure vessel, it would require long times to propagate through the greater wall thickness, allowing enough time to be detected by ISI.

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 will need 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 to affect the progression of the above phenomena and to reduce fatigue life, creep life, and resistance to stress corrosion cracking, oxygen needs to be included in new experimental studies.

**(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 or refueling, ISI intervals may be long and the amount of inspection conducted 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.

### (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, reflectors (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 of 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 makes the volume 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 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 from 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, and thermal conductivity 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 be released 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.

#### IV.2.4.3 Materials Analysis – 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 develop an understanding of the mechanics of pebble flow. A description of this research for metallic components, ISI, and graphite components follows.

**(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 an impure helium environment, 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 address reductions in life due to the operating environment. If the codes and procedures do not consider this phenomena, then the data base developed can be used to update the codes and procedures to provide design procedures and rules that 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 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 either to 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, quantify 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 the welded conditions. 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, including 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.

With respect to international agreements, there is considerable research that has been performed or is ongoing in the European Community (EC) and Japan on high temperature metals for HTGRs. Through interactions with technical staff in the EC and Japan, the NRC staff identified several areas that address NRC research objectives. Work of interest in the EC is a) review of RPV materials, focusing on previous HTRs, in order to set up a materials property database on design properties, b) compilation of existing data on materials for reactor internals and selection of the most promising alloys for further development and testing, and c) compilation of existing data on turbine disk and blade materials and selection of the most promising alloys for further development and testing. Experimental work in these areas includes a) research on a pressure vessel steel containing 9% Cr (irradiation testing, fatigue, creep-fatigue, tensile, fracture toughness); both heavy-section base metal and weldments are included in the studies; b) mechanical and creep tests of candidate alloys for reactor internals at temperatures up to 1100$^{o}$C with focus on the control rod cladding; and c) tensile, fatigue, and creep tests from 850$^{o}$C up to 1300$^{o}$C for two different turbine blade materials, one forming an aluminum oxide protective layer, the other a chromium oxide layer.

Work of interest that has been conducted by Japan Atomic Energy Research Institute (JAERI) includes development of a high temperature metallic component design guide, research on high temperature metal corrosion, and irradiation effects on a 2 1/4 Cr-1Mo reactor pressure vessel steel.

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 (October 2001, US NRC, Rockville). 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 unacceptably high temperatures.

As mentioned above, considerable research on high temperature materials for HTGRs of interest to NRC has been conducted, is ongoing, or planned in the EC and Japan. To leverage NRC resources and obtain data in a timely manner, the staff has visited facilities and met with members of the international community to initiate a dialogue on cooperation. Descriptions of research on high temperature materials described in this plan have been shared with the international community, in particular with Japan and the EC. NRC staff has met with technical staff and officials of the EC and JAERI to discuss potential cooperation. The EC has agreed with the importance and need for the research outlined in our plan and welcomes the NRC to participate in their high temperature materials research (HTR-M) program. Similarly JAERI has agreed in principle to cooperate with the NRC. Participation is through the exchange of research results, and not funds, from the parties' research programs. Some of the work described in the advanced reactor research plan will be addressed in the EC current program and their future program to initiate in 2003. Some of the key work possibly not fully addressed in the EC programs is in the areas of a) effects of the helium environment with impurities on degradation of materials, and b) aging and sensitization. Exchange of NRC research results in these areas could be used for cooperation with the EC HTR-M programs.

**(2) Inservice Inspection (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 appropriate to the reactor design and operating environment, along with the inspection variables, will be considered 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 and crack growth and for leak detection. Acoustic emission techniques will be used for laboratory testing of specimens under simulated HTGR and ALWR conditions (respective temperature, noise sources, coolant flow, etc.) to evaluate 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 results 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.

Areas of international cooperation and exchange would involve work planned by the EC on evaluation of ISI methods, and work on risk-informed inspection program evaluation by NRC. Of additional interest would be potential international cooperation on evaluations of on-line continuous monitoring techniques for structural integrity and leak detection using HTGR test reactors.

**(3) Graphite**

Research will be conducted to evaluate graphite for HTGR application. This will involve studies 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 unpublished data from ORNL taken under the DOE NP-MHTGR program. 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 principal 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 the effect of irradiation and to 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 to establish in-block variation and variability between graphite batches.  Large graphite blocks to be used for reflector material will be sectioned, tested, and evaluated to quantify through-thickness properties and determine if properties developed from thin graphite components can be extrapolated to large blocks.  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 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 for which a large body of research and experience is available.

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 will be developed for any necessary follow-on studies.

The EC research effort is currently reviewing the state of the art on graphite properties in order to set up a suitable database.  The EC is planning to perform oxidation tests at high temperatures on fuel matrix graphite and on advanced carbon-based materials to obtain oxidation resistance in steam and in air.  Recently, the EC begun extensive characterization and irradiation testing of five different graphites (2 from UCAR, 2 from SGL, 1 from a Japanese source) that are currently produced and could be used in future HTGRs.  The properties of these graphites as a function of temperature and irradiation exposure will be studied.  As mentioned above for the high temperature metallic components, the EC plans to address much work described in this research plan, however a key area possibly not fully addressed in the EC programs is the correlations of virgin graphite properties and manufacturing parameters to post-irradiation graphite properties. Exchange of NRC research results in this area could be used for cooperation with the EC HTR-M programs.

The UK is conducting ongoing research on graphite properties and has had experience with operating gas cooled reactors that 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 and 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 to 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 to evaluate the relevance of this information for application to currently proposed HTGRs.

#### IV.2.4.5 Materials Analysis – Application Of Research Results

Research results will provide input on component probability of failure for NRC Probabilistic Risk Assessments (PRAs) to independently confirm and support safety evaluations. Since failure probability data for components of advanced reactors is not available from operating experience, very large uncertainties are inherent in the values selected and in the results of the PRAs. To reduce the uncertainties, information on failure probabilities would be derived from research results of potential degradation mechanisms (fatigue, creep, creep-fatigue, oxidation, thermal aging, stress corrosion cracking, crevice corrosion cracking, irradiation damage, and dimensional changes) of components in the operating environment of advanced reactors and with quantitative information of the initiation times and growth rates.

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 degradation 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 to use continuous online monitoring of structural integrity for structures and components of advanced reactors.

### IV.2.5 Structural Analysis

#### IV.2.5.1 Structural Analysis – 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 LWRs and as such may not adequately address analysis, design, construction, and licensing issues of the ALWRs such as AP-1000 and IRIS and other types of HTRGs such as PBMR and GT-MHR. 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, inspection, material aspects, and challenge of external events for the HTGRs, 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 make recommendations related to the use of performance based and/or risk-informed criteria to evaluate the acceptability of proposed advanced reactor designs.

In 1996 and 1997, the NRC updated the seismic and geological criteria for siting 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 U.S., 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. Additional data and recommendations will improve and facilitate the licensing process for advanced reactor designs.

In the proposed HTGR reactor vessel internal structure designs, the ceramic reflector structure consists of graphite blocks with holes for control rods. Therefore, it is necessary to retain alignment through vertically arranged blocks that are supported vertically by a dowel system, and circumferentially by a radial keying system. For the AP-1000, fuel tubes are taller than conventional designs and the seismic margin is controlled by fuel design. Confirmatory research is needed on these tall structures since they are 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. Due to the lack of licensing experience on seismic response of deeply buried structures, research insights are 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 U.S. and foreign journals, transactions, and proceedings as well as in earlier research by Sandia National Laboratory.

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 evaluation criteria and the need for verified design/analysis methodology for unique types of modules, such as the concrete-filled steel plate 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.

#### IV.2.5.2 Structural Analysis – Purpose

The purpose of this research activity is to develop the criteria for the structural and seismic evaluation of the new features of advanced reactor designs. The advanced 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. 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 assessment, (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; and reduction in the amount of piping required. 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-U.S. codes and standards for Safety Class design and construction applications. However, not withstanding these features, the majority of the analysis, design, fabrication, and construction methods are similar to those applicable to recent domestic commercial nuclear power plants.

The unique design features of the HTGRs include different 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 may result 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 utilize some structural design and construction processes similar to the ALWRs 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 U.S.

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.

A key area of analytical and experimental research for advanced reactors 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 the PBMRs.

Current seismic 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 determining SSI effects and passive earth pressures on deeply embedded or buried structures during earthquakes.

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 evaluation criteria due to the fact that existing U. S. codes and standards do not address composite structures (concrete-filled steel plate modules) and the need for verified design/analysis methodology for unique types of modules, such as the concrete-filled steel plate wall and floor modules.

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 a 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.

#### IV.2.5.3 Structural Analysis – Objectives and Planned Activities

The overall objective of this research is to assess 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 technical basis for 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:

- **Seismic Hazard Assessment**

The objective of this research activity is to update the two current seismic hazard assessments for the central and eastern U.S. 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 selection 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 of proposed advanced reactor designs.

- **Nonlinear 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. Research performed by foreign R&D organizations and regulators will be reviewed for applicability and to determine gaps where additional research is needed. Analytical and experimental 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 nonlinear configurations such as the HTGR reactor components consisting of nonductile graphite core reflectors and supports. There is special need to perform experimental verification of the seismic response of the first-of-a-kind design of HTGR internals.

Research needs to be conducted on the nonlinear static and dynamic structural analysis of advanced reactors with long fuel tubes and core support structures whose seismic margin might be controlled by the fuel design. For the PBMR reactor, fuel pebbles are piled into a considerably tall configuration resulting in nonlinear responses during horizontal and vertical components of earthquakes. Research should 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.

- **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 and passive earth pressures for these types of deeply embedded structures will have a significant influence on the analytically predicted seismic response. Research performed by foreign R&D organizations and regulators will also be reviewed for applicability and to determine gaps where additional research is needed. Research experience in the area of seismic analysis and design of tunnels and buried piping will be utilized to the extent applicable.

Current seismic SSI analysis computer codes used in the nuclear industry 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 and passive earth 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 and state-of-the-art analytical and experimental 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.

- **Effect of High Temperature on Concrete**

The objective of this research is to investigate the change in concrete properties when it is 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 normal use , and 300°F for abnormal conditions.

The operating temperatures of the primary reactor vessels for some of the advanced reactor designs being considered are greater than those for 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 additional shrinkage 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 the literature, including journals, conference transactions, and proceedings. Sandia National Laboratories earlier research on LWR severe accidents also accumulated significant data on the effects of high temperatures on properties of concrete, Oak Ridge National Laboratory has also assembled information on concrete subjected to high temperature. Lessons learned from facilities where concrete was found to be subjected to high temperatures for long durations will also be investigated and utilized.

- **Modular Construction**

Modular construction has not been used in the U.S. for nuclear power plants but some techniques have been used in Japan. It has been proposed by the HTGR, GT-MHR, AP-1000 and the IRIS designers 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 the module, of the joints and connections as well as appropriate damping values for seismic analyses. Presently, U. S. codes and standards guidance is lacking or non-existent for the design of concrete-filled steel plate wall and foundation modules, and for the design of the connection of the concrete-filled module to a concrete-filled steel plated foundation module.

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. Calculation methods will be verified based in part on available test data on structural modules 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.

With respect to 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."

- **Risk-informed 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 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 indicated that inspection resources need to be focused on risk-significant areas and that inspection methods should be tailored to the potential degradation mechanisms. In some cases, existing inspection requirements are not focused on locations where cracks and leaks have been discovered.

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.

#### IV.2.5.4 Structural Analysis – Application Of Research Results

The end product of this work will be guidance in a NUREG for each task and updates of regulatory guides and SRPs, as necessary. In addition, the completion of an updated methodology for seismic hazard assessment 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 new seismic analysis techniques will be new or revised computer codes that may be utilized by the staff for the review of new reactor submittals. The results of the efforts on concrete performance and modular construction 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.

### IV.2.6 Consequence Analysis

#### IV.2.6.1 Consequence Analysis – 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.

#### IV.2.6.2 Consequence Analysis – 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.

#### IV.2.6.3 Consequence Analysis – 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.

#### IV.2.6.4 Consequence Analysis – 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."

## IV.3 MATERIALS SAFETY AND WASTE SAFETY

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

#### IV.3.1.1 Nuclear Analysis for Materials Safety and Waste Safety – 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 materials safety and waste safety. Nuclear analysis research efforts for the reactor safety arena and the safeguards arena are discussed in other sections of this document.

While nuclear analysis is by no means the only technical discipline of importance to the regulation of nuclear materials 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).

### IV.3.1.2 Nuclear Analysis for Materials Safety and Waste Safety – 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.

#### (1) Nuclear Criticality Safety at the Front End of the Fuel Cycle[1]

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 studies may be needed to support the staff's independent assessment of acceptable criticality modeling practices for HTGR particle fuel forms. It is noted, for example, that LEU pebbles and compacts are generally much more reactive than would be predicted by simplified computational models that smear the fuel particles and matrix carbon 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 studies 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.

#### (2) Safety Analyses for Spent Fuel Management[2]

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,

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

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

(b) radiation sources for shielding, and (c) spent-fuel reactivities (i.e., burnup credit) for criticality safety.  As has been the case with current LWRs, the technical safety issues for away-from-reactor management of spent fuel, as regulated by NMSS, will generally be encountered after those for the NRR-regulated at-reactor handling and storage of irradiated fuels.  Especially for at-reactor handling and storage, it is anticipated that extensive burnup credit will be requested in the 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.

#### IV.3.1.3 Nuclear Analysis for Materials Safety and Waste Safety – 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.

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

- **Nuclear Data Libraries**

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

- **Nuclear Criticality Safety at the Front End of the Fuel Cycle**

(2) 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 recommendations 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 benchmarking activities for code validation and code-to-code comparison activities for qualifying code users and modeling practices.

- **Safety Analyses for Spent Fuel Management**

(3) 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

recommendations 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 benchmarking and code-to-code comparison activities.

(4) Validation and Modeling Guidance on Predicting Decay Heat and Radiation Sources in Spent Fuel from (a) PBMR, (b) GT-MHR, and (c) IRIS: Building upon closely related work on burnup credit (previous item) and short-term decay heat sources for reactor safety (see section IV.2.2.2 on Nuclear Analysis for Reactor Safety), identify and review existing and planned databases of spent fuel radiation measurements, radionuclide assays, and calorimetry measurements and use sensitivity methods to assess their applicability for code validation in predicting the long-term (i.e., 10 days to 100 years and beyond) decay heat and radiation sources in spent fuel from the respective reactor types. Develop options and recommendations for the evaluation and treatment of remaining validation uncertainties. Develop modeling guidance to help ensure that accepted modeling approximations and assumptions will not lead to significant underpredictions of long-term decay heat or radiation sources. Participate in cooperative programs for new experimental data as well as code-to-data benchmarking and code-to-code comparison activities.

#### IV.3.1.4 Nuclear Analysis for Materials Safety and Waste Safety – 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.

### IV.3.2 Uranium Enrichment And Fuel Fabrication

#### IV.3.2.1 Uranium Enrichment and Fuel Fabrication – 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 (as low as reasonably achievable (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.

#### IV.3.2.2 Uranium Enrichment and Fuel Fabrication – Purpose

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

#### IV.3.2.3 Uranium Enrichment and Fuel Fabrication – 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 Integrated Safety Analysis summaries 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.

#### IV.3.2.4 Uranium Enrichment and Fuel Fabrication – 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.

### IV.3.3 Transportation and Storage

#### IV.3.3.1 Transportation and Storage – 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, particles in pebbles or compacts inside blocks 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[3] (as discussed in Section IV.3.1) 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.

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

#### IV.3.3.2 Transportation and Storage – 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.

#### IV.3.3.3 Transportation and Storage – 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 in Section IV.3.1.1.

#### IV.3.3.4 Transportation and Storage – 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.

### IV.3.4 Waste Disposal

#### IV.3.4.1 Waste Disposal – 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 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 (GR-) 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 GR- 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 GR- 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.

Releases of radionuclides to the environment from low-level waste (LW) disposal facilities containing waste streams from advanced reactors and radionuclide releases to the environment from the decommissioning materials of advanced reactors must be understood to predict the transport of radionuclides in soils, ground and surface water, the atmosphere, and the surrounding biosphere to estimate radiation exposure to the average member of the critical group and to ensure compliance with 10 CFR Part 61 regulatory requirements and the policies in the Decommissioning Standard Review Plan (NUREG 1727).

Areas of interest from a research perspective include issues related to waste streams generated from the operation of the advanced reactors. Long-lived radionuclides present in the waste streams may be different from the waste streams and distribution of radionuclides in current LWRs. Questions regarding LW generated by advanced reactors include: (1) Will the Part 61 waste classification system make some advanced reactor waste streams ineligible for disposal in LW disposal facilities? (2) How much LW is generated by advanced reactors and what are the important waste streams and types of advanced reactor LW? (3) Are there long-lived radionuclides present in advanced reactor LW that are not present in LWR LW? (4) Will activated metals be a significant component of advanced reactor LW? (5) How much transuranic waste is generated by the advanced reactors? (6) Will packaging and shipping requirements have to change?

#### IV.3.4.2 Waste Disposal – 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 GR- 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 GR-.

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.

With respect to LW, determining radionuclide releases from advanced reactor LW and decommissioning materials under varying chemical and physical conditions are important aspects in determining source terms and assessing the performance of LW disposal facilities and reactor decommissioning sites. To calculate radionuclide releases from advanced reactor LW disposed in LW disposal sites and from decommissioning materials at decommissioned advanced reactor sites, one needs to consider radionuclide inventories of both the LW and decommissioned materials, radionuclide releases and solubilities. For LW, additional data and information on waste types, forms, and waste containers is required.

For decommissioning advanced reactor sites, it is important to provide data and information about planned decommissioning activities that will be helpful in order to establish specific decommissioning plan requirements. For example, will certain radionuclides present in decommissioning materials present a unique decommissioning challenge?

#### IV.3.4.3 Waste Disposal – 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.[4] The other remaining issues will be addressed as part of the advanced reactor fuel program.

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

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.

The primary objective of the advanced reactor LW disposal research program is to provide experimental data and information to be used to determine realistic radionuclide inventories, calculate realistic radionuclide source terms releases from LW disposal facilities and decommissioned sites, and support the development of regulatory criteria (e.g., regulations, regulatory guides, policy guidance, standard review plans) for the disposal of LW generated by advanced reactors and the decommissioning of advanced reactor sites.

Research activities would include:

1. Characterize advanced reactor LW waste streams and decommissioning materials for radionuclide and chemical content.

2. For LW, determine radionuclide concentrations by waste stream, waste type, waste form, and waste classification.

3. Identify differences between advanced reactor LW streams and radionuclides and the LW waste generated by current LWRs.

4. Determine radionuclide releases, including the chemical and physical factors affecting releases, by performing laboratory and field leaching studies on LW, including activated metals, and decommissioning waste materials.

5. For important long-lived radionuclides which may be present only in advanced reactor waste and not in typical LWR waste, determine sorption coefficients using soils typical present at LW disposal facilities and decommissioning sites.

6. Proved probabilistic distributions and associated statistical parameters for radionuclide releases for use in risk-informed and performance based computer codes.

7. Determine if $^{14}C$, $^{110m}Ag$, and other radionuclides as well as carbon dust present a unique decommissioning challenge.

#### IV.3.4.4 Waste Disposal – 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 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 basis 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.

### IV.3.5 Personnel Exposure Control During Operation[5]

#### IV.3.5.1 Personnel Exposure Control During Operation – 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 PBMR and for the GT-MHR: 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.

#### IV.3.5.2 Personnel Exposure Control During Operation – 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.

#### IV.3.5.3 Personnel Exposure Control During Operation – 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[6] and assess associated radiation streaming issues[7] in view of potential concerns over vessel fluence[8] as well as radiation

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

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

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

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

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.

#### IV.3.5.4 Personnel Exposure Control During Operation – Application Of Research Results

Provide reviewers with insights from analyses.

## IV.4 SAFEGUARDS

### IV.4.1 Safeguards – Background

The fuel elements for PBMR and IRIS will be enriched up to 8 wt% and for GT-MHR up to 19.9 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 material control and accounting potentially more difficult. This research area addresses material and reactor safeguards, including the 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.

### IV.4.2 Safeguards – Purpose

The purpose of advanced reactor research activities for the Safeguards arena is to support the establishment of a technical basis for the staff's assessment of advanced reactors and their fuel cycles in terms of:

(1) the potential consequences from internal and external threats to reactor facilities, fuel enrichment facilities, fuel fabrication facilities, shipments of fresh fuel materials, shipments of spent fuel and waste, storage facilities for spent fuel and waste, and waste disposal facilities;

(2) the adequacy of material control and accounting and security measures for preventing and detecting material diversion throughout the respective fuel cycles;

(3) the potential for overt and covert misuse of reactors to produce materials for fission weapons; and

(4) the technological barriers to extraction and processing of materials for use in fission weapons and radiological weapons (i.e., dirty bombs).

The safeguards activities should be commensurate with the relative ease and potential consequences of diverting the respective advanced reactor fuel materials. Work in these areas should be coordinated with the safeguards related activities of the IAEA, especially as they relate to international safeguards, and with the safeguards and homeland security efforts of other government agencies, as appropriate.

### IV.4.3 Safeguards – 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. In addition, the technological barriers to extracting plutonium and other radionuclides from irradiated fuel materials will be described for the respective advanced reactor technologies.

Specific activities include:

**(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.

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 conventional 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.

**(2)** **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 (i.e., dirty bombs).

**(3) 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.

### IV.4.4 Safeguards – 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.

## V. PHENOMENA IDENTIFICATION AND RANKING TABLE (PIRT) PROCESS

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.

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) AP-600 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:

## V.1 UMBRELLA PIRT FOR COMPREHENSIVE REACTOR SAFETY EVALUATION

### V.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, (ii) the June 4, 2001, ACRS Subcommittee on Advanced Reactors meeting, and (iii) relevant NRC pre-application 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 pre-applicant 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.

### V.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.

### V.1.3 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.

## VI. IMPLEMENTATION

Successful implementation of an effective advanced reactor research infrastructure will depend upon several factors including 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., development of regulatory framework) will have to be initiated and completed whether there be one or more license applications. As discussed in Section V, a systematic and logical Phenomenon Identification and Ranking Table (PIRT) process will be implemented to prioritize various research topics. Using the guidelines, needed research activities can be ranked in order of importance/priorities, available resources can be allocated, and schedules can be established.

Inevitably, 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 arenas. To alleviate the burden, some shared research with the industry is also expected. 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 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:

### VI.1 IMPACT ON DECISION-MAKING PROCESS (e.g., high, medium, low)

- How conservative will the decision have to be if the information is not obtained?
- Does the information have to be independent of the applicant's information?
- What are the implications if the desired information is not generated to the level desired or in the time frame required?

### VI.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?
- What part of the cost of generating/developing data/tools can be shared by the applicant i.e., beyond what the applicant is required to submit to make the safety case?
- What additional information is available from other sources (e.g., international partners or via domestic ventures, such as DOE- or industry-funded efforts)

- Do we have the necessary performance/acceptance criteria for the final product? What levels of uncertainties would we accept? How will uncertainty be treated?

- Would there be a need to do any sensitivity analysis?

- What means (e.g., experimental data, code-to-code validation, peer review) would we need for testing/validating/accepting the final product?

## VI.3 PLANNING

- When should the project be completed to support the licensing process?

- When does the NRC need to initiate the research efforts? This is especially important for long-lead time products (e.g, fuel irradiation, thermal fluid dynamics testing).

- 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?

- 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?

- How do the schedule constraints of other related key areas affect the outcome of this research project?

- 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?

## VI.4 LEVERAGING (Is the desired information/product (or part of it) available from domestic or international partners?)

- Can the applicant be asked to provide part or all of the supporting data?

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

- 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?
- What is the feasibility of a joint venture with the industry?
- Can the required information be purchased internationally or domestically for reasonable cost or by making a contribution in kind?

## VI.5 REQUIRED FISCAL AND HUMAN RESOURCES

- Can the applicant be asked to share the cost of generating/developing the information?
- 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 facilities to conduct and support the desired research, generate data, or develop the desired tool?
- How much time and resources are needed for quality checks or independent testing/validation of the end-product? Do we have peer reviewers identified?
- 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?

# VII. COOPERATIVE RESEARCH

Unlike proven LWR technology where extensive LWR-related operational worldwide 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 only be gross approximations. 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 experience, namely, the aviation and chemical industries, 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.

## VII.1 INTERNATIONAL COOPERATION

Inevitably, a great deal of HTGR-related data will have to be generated in laboratory settings under accelerated, simulated operational and post-accident conditions. 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 and international partners will be an integral part of future planning. NRC's active participation in ongoing research programs and new cooperative efforts with various international organizations needs to be designed so as to deliver optimum mutual benefits while off-setting costs.

## VII.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. The existing AVR operational experience and data provide significant insights in identifying the future research needs. 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 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 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 CRPs on code validation using data from HTR-10 and HTTR; a graphite database being developed under the sponsorship of IAEA.

## VII.3 WORKSHOPS AND MEETINGS

The staff hosted and participated in several conferences and workshops. In the year 2001, three advanced reactor workshops were hosted by the NRC. On June 4, 2001, the ACRS Subcommittee on Advanced Reactors sponsored a public workshop. On July 25, 2001, "Workshop on Future Licensing Activities" was sponsored by NRR. Both workshops were open to the stakeholders. From October 10-12, 2001, an HTGR Safety and Research Issues Workshop was hosted by RES. Participation at this workshop, however, was by invitation only. It was intentionally kept free of parties with vested interest, such as vendors, builders, and potential 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 venues for future internatuional cooperation were also identified. 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. This workshop provided a broad overview of the advance reactor designs being considered worldwide, and also served as a valuable forum for discussions on advanced reactor concepts being studied and for identification of safety issues and research needs. In April 2002, the staff participated at the ICONE-10 conference in Washington. Also in April, the staff attended the HTR conference held in Petten, the Netherlands. The staff continues additional dialogue with international partners to explore prospects for future cooperation on HTGR-related efforts.

## VII.4 PROSPECTS FOR FUTURE INTERNATIONAL COOPERATION

Since the beginning of the PBMR pre-application review process in 2001, US delegations have visited South Africa, UK, Germany, China and Japan. There is considerable potential for future cooperative efforts with various countries. Various invited national and international experts focused on identifying key HTGR safety issues and necessary research and development to support HTGR licensing reviews. Additionally, NRC-sponsored conferences Nuclear Safety Research Conference (NSRC), held in November 2001, and the Regulatory Information Conference (RIC), held in March 2002, each had sessions devoted to advanced reactor issues. These conferences were open to the public and were widely attended by potential applicants and vendor representatives as well as consultants and the members of public. 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. In April 2002, the staff met with the E.C. representatives in Washington and again in the Netherlands with their counterparts and other key researchers to further discuss the details of possible future cooperation. Additional dialogue is necessary to formulate formal agreements with the E.C. and other countries to develop HTGR-related new codes and experimental data, and to share the existing data. The staff continues its efforts to develop details of future cooperative research efforts with various international partners, including possible exchange of experts. The staff is in the process of developing a matrix identifying key research topics being examined by various international partners; the NRC priorities and the extent of interest in the ongoing programs; the scope of possible joint venture in the existing or the modified programs; the status of formal agreements with the key participants/sponsor; and the need for new or modified agreements. This matrix will be maintained as a “living” illustration and will be modified, as necessary.

## VII.5 PARTICIPATION IN THE IAEA-SPONSORED PROGRAMS

The IAEA’s documented data from various Coordinated Research Programs (CRPs), as well as international conference proceedings in various TECDOC, represent a significant information base. It is anticipated that the NRC will actively participate in the future HTGR-related CRPs. The NRC expects to participate 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 the UK. With support from Japan, South Africa, UK, and US, the IAEA has begun development of a database related to irradiated nuclear graphite properties. The objective of this effort is to preserve the existing worldwide knowledge on the physical and thermo-mechanical properties of the irradiated graphite, to provide a validated data source to member countries with interest in graphite-moderated reactors or development of HTGRs, and to support continued improvement of graphite technology applications. The database 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, intends to identify research needs and exchange information on advances in technology for selected topical areas of primary interest to HTGR development. The Group will establish, 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$^{o}$ 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$^{o}$ 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.

## VII.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. Earlier, 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 meeting, 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, the IVV-2M in particular. Integration of the

European Program (HTRTN) with the Japanese and Chinese programs was strongly recommended. Basic studies, such as core physics code qualification, fuel and material irradiation, and graphite behavior and characterization were suggested. It was also recommended that

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

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

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

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

(5) 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. Since irradiation experiments need extensive time and resources, it is important that information exchange on irradiation experiment details be done effectively.

## VII.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 may also be expanded to encompass future efforts in conducting HTGR fuel testing and experiments. Currently, under 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), and foreign R&D organizations (24). There are nine ongoing projects under NERI that relate to the GCR technology. GCR fuel irradiation program and 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 Westinghouse, various US universities, and the 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. In addition, DOE's Generation IV Technology Roadmap 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, and particle fuels); and innovative safety research (including advanced computational methods for seismic, T/H, and nuclear analysis).

## VII.8 DOMESTIC EFFORTS

In April 2002, Exelon announced its termination of the PBMR pre-application review activities. The staff is in the final stages of phasing out this effort. Initiation in the near-term of the pre-application review of the GT-MHR continues to maintain 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, and high-temperature materials performance issues. 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 R&D organizations.

In May 2002, the staff met with the MIT representatives to discuss their ongoing research and development programs which includes different advanced reactor concepts. Prospects for a cooperative agreement with MIT on HTGR-related research and other topics are being explored. In a separate venture, the staff is working on the development and application of a systematic decision making process for prioritizing research. This approach will be used as one of the tools for prioritization of future research efforts.

## VII.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, 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 HTGRs and ALWRs come 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, and 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.

## VII.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 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.

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 fluid dynamics, 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 to demonstrate 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.