ADVANCED REACTOR RESEARCH

INFRASTRUCTURE ASSESSMENT

Revision 3
Office of Nuclear Regulatory Research
January 2003
Attachment 2

EXECUTIVE SUMMARY

In a Staff Requirements Memorandum (SRM) dated February 13, 2001, the Nuclear Regulatory Commission (NRC) 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 10 *Code of Federal Regulations* (CFR) Part 50 and Part 52, and other applicable regulations that may require updating. 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 and infrastructure assessment. This assessment is the subject of this report. As described within, the assessment is essentially a gap analysis of the NRC's advanced reactor research capabilities. At this point, the research identified and described does not delineate the activities that will be conducted solely by the NRC. Rather, it is intended to identify information gaps that exist at the NRC in terms of needed expertise, analytic tools, and methods. Within this context, it should be recognized that an applicant has the primary responsibility to demonstrate the safety case, and to a large extent this will impact the extent to which NRC research is necessary.

The scope of NRC advanced reactor research includes both confirmatory and anticipatory research activities as they apply principally to six reactors identified in the FLIRA report: the Pebble Bed Modular Reactor (PBMR), Gas Turbine-Modular Helium Reactor (GT-MHR), Westinghouse advanced pressurized water reactor AP-1000, and Westinghouse International Reactor Innovative and Secure (IRIS), General Electric's ESBWR, Atomic Energy Limited Advanced Canadian Deuterium-Natural Uranium Reactor (CANDU) Reactor (ACR-700). Generation IV (Gen IV) reactors have not been included at this time because of their preliminary stage of development. The staff plans to perform periodic assessments and maintain this as a living document to reflect any new issues and technologies not previously considered. In addition, future updates will capture new advanced light-water reactor designs now undergoing NRC pre-application review.

The infrastructure assessment originated from a technology-neutral perspective. Technical topics and activities were identified and linked to nine key research areas: (1) framework (including the development of regulatory decision-making tools based on the risk-informed, performance-based principles); (2) accident analysis (including probabilistic risk assessment (PRA) methods and assessments, human factors, and instrumentation and control); (3) reactor/plant systems analysis (including thermal-fluid dynamics, nuclear analysis, and severe accident and source term analysis); (4) fuels analysis and testing; (5) materials analysis (including graphite behavior and high-temperature metal performance); (6) structural analysis (including containment/confinement performance and external challenges); (7) consequence analysis (including dose calculations, and environmental impact studies); (8) nuclear materials safety (including enrichment, fabrication, and transport) and waste safety (including storage, transport, and disposal), and (9) nuclear safeguards and security.

It should be emphasized that not all the research described within this document will be done by the NRC. Information can and will be obtained through domestic and international cooperation, as well as through research and development conducted by promoters of the designs. Accordingly, prioritization and budgeted resources will take into consideration information obtained from others, with due consideration of NRC responsibility as an independent regulatory agency.

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ABBREVIATIONS

ABWR	advanced boiling-water reactors
ACNW	Advisory Committee on Nuclear Waste
ACRS	Advisory Committee on Reactor Safeguards
ADS	automatic depressurization system
AGR	advanced gas-cooled reactor
ALARA	as low as reasonably achievable
ALWR	advanced light-water reactor
AMPX	A Modular Code System for Processing Xsections
APEX	Advanced Plant Experiment
ASME	American Society of Mechanical Engineers
ASTRA	Advanced Gas Reactor in Kurchatov Institute, Russia
ATWS	anticipated transient without scram
AVR	Arbeitsgemeinschaft Versuchsreaktor
BICEP	a critical experiment facility in the United Kingdom
BISO	bi-isotropic
BWR	boiling-water reactor
CANDU	Canadian Deuterium-Natural Uranium Reactor
CDF	core damage frequency
CESAR	a critical experiment facility in Grenoble, France
CFP	coated fuel particle
CFR	*Code of Federal Regulations*
CHF	critical heat flux
CNSC	Canadian Nuclear Safety Commission
CONTAIN	containment and computational code
CREIPI	Central Research Institute of Electric Power Industry
CRP	coordinated research project
CY	calendar year
DBT	ductile-brittle transition
DBTT	ductile-brittle transition temperature
DHC	delayed hydride cracking
DOE	Department of Energy
E.C.	European Commission
ECC	emergency core coolant
ECOLE	a critical experiment facility in Cadarache, France
ENDF/B	Evaluated Nuclear Data File, Volume B
EPRI	Electric Power Research Institute
EPZ	Emergency Planning Zone
ERVC	external reactor vessel cooling
ESP	early site permit
FLIRA	Future Licensing and Inspection Readiness Assessment
FLUENT	computational fluid dynamic code made by FLUENT, Inc.
FP	fission product
FRG	Federal Republic of Germany
GA	General Atomics
GCR	gas-cooled reactor
GDCS	Gravity-Driven Cooling System
Gen IV	Generation IV
GIRAFFE	Japanese test facility for ESBWR
GROG	a critical experiments facility in Kurchatov Institute, Russia
GRSAC	Graphite Reactor Severe Accident Code
GT-MHR	Gas Turbine-Modular Helium Reactor
GUI	graphical user interface
GWd/t	gigawatt days per ton
HFE	human factors engineering
HLW	high-level waste
HRA	human reliability analysis
HTGR	high-temperature gas-cooled reactor
HTR	high-temperature reactor

ABBREVIATIONS

HTR-10	10-megawatt electric (MWe) High-Temperature Reactor
HTR-F	high-temperature gas-cooled reactor fuel technology
HTR-M	high-temperature materials research
HTR-PROTEUS	High Temperature Reactor Configuration of the Proteus Critical Experimental Facility in Switzerland
HTTR	High-Temperature Engineering Test Reactor
I&C	instrumentation and control
ICS	isolation condenser system
IAEA	International Atomic Energy Agency
IEEE	Institute of Electrical and Electronics Engineers
IFCI	integrated fuel-coolant interaction
INEEL	Idaho National Engineering and Environmental Laboratory
INET	Institute for Nuclear Energy and Technology
IRIS	International Reactor Innovative and Secure
IRWST	in-containment refueling water storage tank
ISI	inservice inspection
JAERI	Japan Atomic Energy Research Institute
JEF	Joint European File
JENDL	Japanese Evaluated Nuclear Data Library
KAHTR	a critical experiment facility in Jülich Research Center
KFA	Jülich Research Center
LBB	leak-before-break
LEU	low-enrichment uranium
LLNL	Lawrence Livermore National Laboratory
LLW	low-level waste
LOCA	loss-of-coolant accident
LWR	light-water reactor

ABBREVIATIONS

MAAP	modular accident analysis program
MACCS	MELCOR Accident Consequence Code System
MC&A	material control and accounting
MCNP	Monte Carlo "N" Particle
MELCOR	a severe accident computer code
MHTGR	Modular High-Temperature Gas-Cooled Reactor
MINERVA	a critical experiment facility in Cadarache, France
MORECA	code that simulates MHTGR developed by ORNL
MTR	materials testing reactor
MWd/t	megawatt days per ton
N4	France's 1450 megawatt electric PWR design
NACOK	test facility at FZ-Jülich
NDE	nondestructive examination
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NERI	Nuclear Energy Research Initiatives
NEWT	ORNL computer code
NII	Nuclear Installations Inspectorate
NJOY	Nuclear Data Processing System from Los Alamos National Laboratory
NMSS	Office of Nuclear Material Safety and Safeguards
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSIR	Office of Nuclear Security and Incident Response
OECD	Organization for Economic Co-Operation and Development
ORECA	code developed by ORNL
ORNL	Oak Ridge National Laboratory
PANDA	a Swiss facility
PANTHERS	Component test for safety systems in containment
PARCS	Purdue Advanced Reactor Core Simulator
PBMR	Pebble Bed Modular Reactor
PCCS	passive containment cooling system
PHEBUS-2K	program to exmine severe accident phenomena and accident mitigation phenomena
PIE	post irradiation examination
PIRT	phenomena identification and ranking table
ppb	parts per billion
ppm	parts per million
PRA	probabilistic risk assessment
PSI	Paul Sherer Institute in Switzerland
PUMA	Purdue University Multi-Dimensional Integral Test Assembly
PVRC	Pressure Vessel Research Council
PWR	pressurized-water reactor
R&D	research and development
RD-14M	experimental facility run by Atomic Energt of Canada, Limited
REBUS	a critical experiment facility in Belgium
RES	Office of Nuclear Regulatory Research
RIA	Reactivity Insertion Accident
RI-ISI	risk-informed inservice inspection
RPV	reactor pressure vessel
SA	severe-accident
SANA	test facillity at FZ-Jülich
SAPHIRE	Systems Analysis Program for Hands-on Integrated Reliability Evaluation
SCALE	Shielding and Criticality Analysis for Licensing Evaluation
SCC	stress corrosion cracking
SCDAP/RELAP5	Integral Severe Accident Analysis Code
SG	steam generator
SiC	silicon carbide
SNAP	Symbolic Nuclear Analysis Package
SQA	software qualification assurance

ABBREVIATIONS

SRM	Staff Requirements Memorandum
SRP	standard review plan
SSE	safe shutdown earthquake
SSHAC	Senior Seismic Hazard Analysis Committee
SSI	soil structure interaction
TBD	to be determined
TECHDOC	technical document
T/H	thermal hydraulic
THATCH	code developed by Brookhaven National Laboratory
THTR	Thorium Hochtemperaturreaktor
TRAC-M	NRC transient system analysis code
TRISO	tri-isotropic
UCSB	University of California at Santa Barbara
UK	United Kingdom
U.S.	United States
VERCORS	a French facility
VHTRC	Japanese test facility
VICTORIA	Radionuclide Transport and Decommissioning Codes
WIMS/ MONK	a lattice physics code from the United Kingdom/Monte Carlo criticality code from the United Kingdom
wt%	weight percent

I. INTRODUCTION

On February 13, 2001, the Nuclear Regulatory Commission (NRC) 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 10 CFR 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 high-temperature gas-cooled reactors (HTGRs) and advanced light-water reactors (ALWRs). In the FLIRA report, the staff also committed to the development of an advanced reactor research plan and associated infrastructure assessment that would provide a sound basis for budgeting research activities. It was envisioned that an assessment of the research infrastructure (i.e., methods, tools, experimental facilities, and expertise) would help set the direction for future advanced reactor research programs that would be needed to support new reactor licensing. To fulfill the FLIRA commitment to the Commission, the staff performed an infrastructure assessment, or gap analysis. Implementation of the assessment findings within the context of the advanced reactor planning process 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).

In addition to the SRM on FLIRA, the Commission issued an SRM that approved the staff's approach (SECY-01-0070) to pre-application review of the Pebble Bed Modular Reactor (PBMR). Pre-application reviews provide a forum for early interaction between the NRC and the applicant. The PBMR pre-application review, for example, provided valuable insights into policy, technical, and safety issues, and associated infrastructure needs for HTGRs in general. Implementation of research activities to address infrastructure needs are prioritized through the Planning, Budgeting, and Performance Management process and resources are assigned accordingly. In many cases, however, future budget estimates need to be determined in the absence of detailed information on the role of the applicant or industry in addressing the needs. As more information becomes available, resource requirements will be updated to reflect only those activities that require NRC funding, consistent with Fiscal Year 2003–2005 budget projections.

In addition to the PBMR, the nuclear industry has been exploring new, 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 AP-1000, and the Westinghouse International Reactor Innovative and Secure (IRIS). More recently, advanced LWR designs have entered into pre-application review; these include the ACR-700 and ESBWR. These two designs have also been captured by this assessment. An additional advanced boiling-water reactor (Framatome SWR-1000) will also be added once more detail is known. Generation IV (Gen IV) reactor concepts have not been included because of their preliminary stage of development. As discussed below, the infrastructure assessment is expected to be maintained as a living document and will be modified to accommodate new reactor designs.

The infrastructure assessment process focused on critical research areas and information that would be needed to technically support an advanced reactor license submittal review. It is important to note that the approach does not delineate what research would be conducted by the NRC versus the applicant or developer, but rather focuses on the ability to perform safety assessments of advanced designs. This includes identification of information gaps and the tools, data, and expertise needed to fill the gaps. To maintain maximum flexibility, the approach also started from a technology-neutral perspective; however, at some point consideration had to be given to design-specific technical and safety issues.

Most NRC regulations and associated regulatory infrastructure that are currently in place support the licensing of light-water reactors (LWRs). In certain cases, some of these regulations may not apply to future non-LWR licensing applications. The need to (1) develop new safety limits, (2) upgrade databases to assess safety margins or issues not previously considered for current reactors, or (3) address severe accidents is captured as a potential research activity. Although there are several areas in which the research infrastructure will need to be improved to address ALWRs, most research infrastructure gaps relate to HTGRs and ACR-700. These reactors present new challenges to the NRC from both a technical and safety perspective. To effectively and efficiently address these challenges, modifications to the existing regulatory framework will likely be necessary. Therefore, development of a new risk-informed, performance-based regulatory foundation to support an advanced framework is considered to be a key research activity.

It is envisioned that this document would be maintained as a living document, and it will be updated as appropriate to accommodate any new designs and issues. Future updates will (1) identify new information from applicants and potential applicants, international research activities, and the Department of Energy (DOE), and (2) reflect plans and activities to independently confirm an applicant's findings. Common to both is the resolution of safety and design technical issues and the tools, methods, data, and expertise required to resolve them.

II. ROLE OF NRC RESEARCH

While it is the responsibility of the applicant and designer to demonstrate the safety level of proposed reactor designs and technologies, the NRC will conduct, as necessary, research to help support the technical basis for licensing. 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 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 safety issues exist. The NRC also performs research to understand failure thresholds and 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 and 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. These concerns usually arise from the examination of industry trends and insights that help the NRC foresee what information will be needed to respond to future regulatory issues. These examinations, for example, brought about a number of design modifications and safety enhancements during the licensing process for the AP600 design.

While assessing challenges posed by 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, as well as what additional research would be needed to support the licensing office. The general principle to be used for funding a specific research activity is that if data is needed to support regulatory decisions on safety cases for a particular reactor design, the applicant would be responsible for the data. If the NRC believes it is important to explore issues involving uncertainties, or if it is necessary to develop capabilities to independently check licensee results, 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 the 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 by the applicant. 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.

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, the Nuclear Energy Institute (NEI), the Electric Power Research Institute (EPRI), and international nuclear organizations such as the Nuclear Installations Inspectorate (NII), the International Atomic Energy Agency (IAEA), the 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 infrastructure needs are focused on the development of expertise, tools, and methods that support the Agency's mission by identifying, 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 play 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 will need to be modified in order to evaluate HTGR and ACR-700 designs and unique aspects of new LWR designs.

The NRC requires a licensing process that will lead to decisions on significant safety issues that are high quality, technically sound, and supported by robust research. In planning research activities, the focus is primarily on areas in which 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 bridge technological gaps. Another important facet of research relates to materials testing and associated codes and standards development, which generally involve a consensus process. As in the past, pre-application reviews are being used to identify the necessary new (or modifications of existing) codes and standards early in the process.

Two types of research are essential in support of the regulatory process: (1) research to support 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 (i.e., building a sound technical basis will require a deep understanding of the technology, its application, and the inherent uncertainties). The products support safety evaluation reports or guidance in the form of regulatory guides and standard review plan (SRP) sections or NUREG reports.

It should be recognized that even a well-funded and 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 other complex technologies, advanced reactor regulation will be a complex blend of applying technical knowledge within the context of Commission policy and prudent regulatory decisions. Therefore, priorities set within the program will consider the relative importance of the activity to understanding safety issues and the risk significance of these issues. 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 novel set of questions may be raised. The importance of answering these questions by examining the question's pertinence to the safety issues being explored poses a challenge to the NRC. (For example, the performance of fuel particle coatings as a barrier to fission product release may require a new and different regulatory approach.) The benefit of this approach is that it provides a rationale for identifying the key research areas, establishing the basis for priorities and infrastructure needs, and identifying the users' needs and 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 NRR and NMSS feedback, guidance, and involvement.

III. OBJECTIVES AND STRUCTURE

The purpose of this study is to generate insights for implementation of an advanced reactor research infrastructure to support the regulatory process. Within this context, information 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 the flow of information between the various technical disciplines
- key research output results and links to the regulatory process
- priorities to allocate resources
- key milestones and resources over a 5-year period (FY 02-FY 06)

In assessing NRC's research infrastructure, the staff benefitted from numerous technical exchanges, including the ACRS Advanced Reactors Workshop (June 2001), a week-long DOE-sponsored HTGR training course (September–October 2001), and various international activities. These activities included interactions with worldwide experts on gas-cooled technology at the NRC Workshop on High-Temperature Gas-Cooled Reactor Safety and Research Issues held October 10–12, 2001. Workshop participants assigned relative priorities to research areas and identified several opportunities for international cooperative research that drew upon existing domestic and international experience. NRC staff 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 Cooperation and Development/Committee on the Safety of Nuclear Installations (OECD/CSNI). 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. NUREG-1802, "Role and Direction of Nuclear Regulatory Research," provided guidance.

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 more recent 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 were taken into consideration. Technical staff visited countries with HTGR experience, including Germany, Japan, China, South Africa, and the United Kingdom (UK). 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 research, high-temperature materials research, fuel performance research, and codes and standards for advanced designs.

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, was considered to be intrinsic to the planning process. As shown in Figure 1, 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 infrastructure assessment, but rather only those used to stimulate thought in the technical area.

Nuclear Reactor Safety Arena

The current regulations which use defense-in-depth principles and conservative practices, provide a margin. That margin might not be applicable to PBMR or GT-MHR advanced reactor designs in certain areas. In order to probe these margins from a generic perspective, research areas and activities were aligned to 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 associated key research areas. Some of the activities that link to these areas include the following:

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

The fire protection research infrastructure that is currently in place should be applicable to advanced designs, however, this issue will be revisited at a future date once conceptual design features and associated issues are better defined.

In general, research products resulting from these activities either support a technical basis for resolving specific safety issues or support another research area. Information flow among the technical groups and framework is illustrated in Figure 2. The process can be described as follows:

- Information in the form of data and analytic results generated by the fuels, materials, and structural technical groups provides key input to the reactor systems analysis. In turn, reactor/plant analysis provides key information on plant operating conditions and accident conditions that is needed by the fuels, materials, and structural analyses technical analysts.

- 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 into the PRA and are used in the 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 are 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 forms 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 drive the regulatory decision-making process, because they impact the safety system classifications. Thus, accident analysis, consequence analysis, and regulatory framework are directly linked to each other.

Once significant accident scenarios are identified for a plant design, reactor systems analysis can be performed and results used to place performance limits on the reactor fuel, reactor internals, and other structural materials. Additionally, reactor systems analysis and associated sensitivity studies can be used to assess margins, which are crucial to a robust accident analysis. As the process is implemented, risk perspectives will be used to support the regulatory framework decision-making activities.

Nuclear Materials Safety and Nuclear Waste Safety Arenas

Advanced reactor research activities for the Nuclear Materials Safety and Nuclear Waste Safety arenas 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.

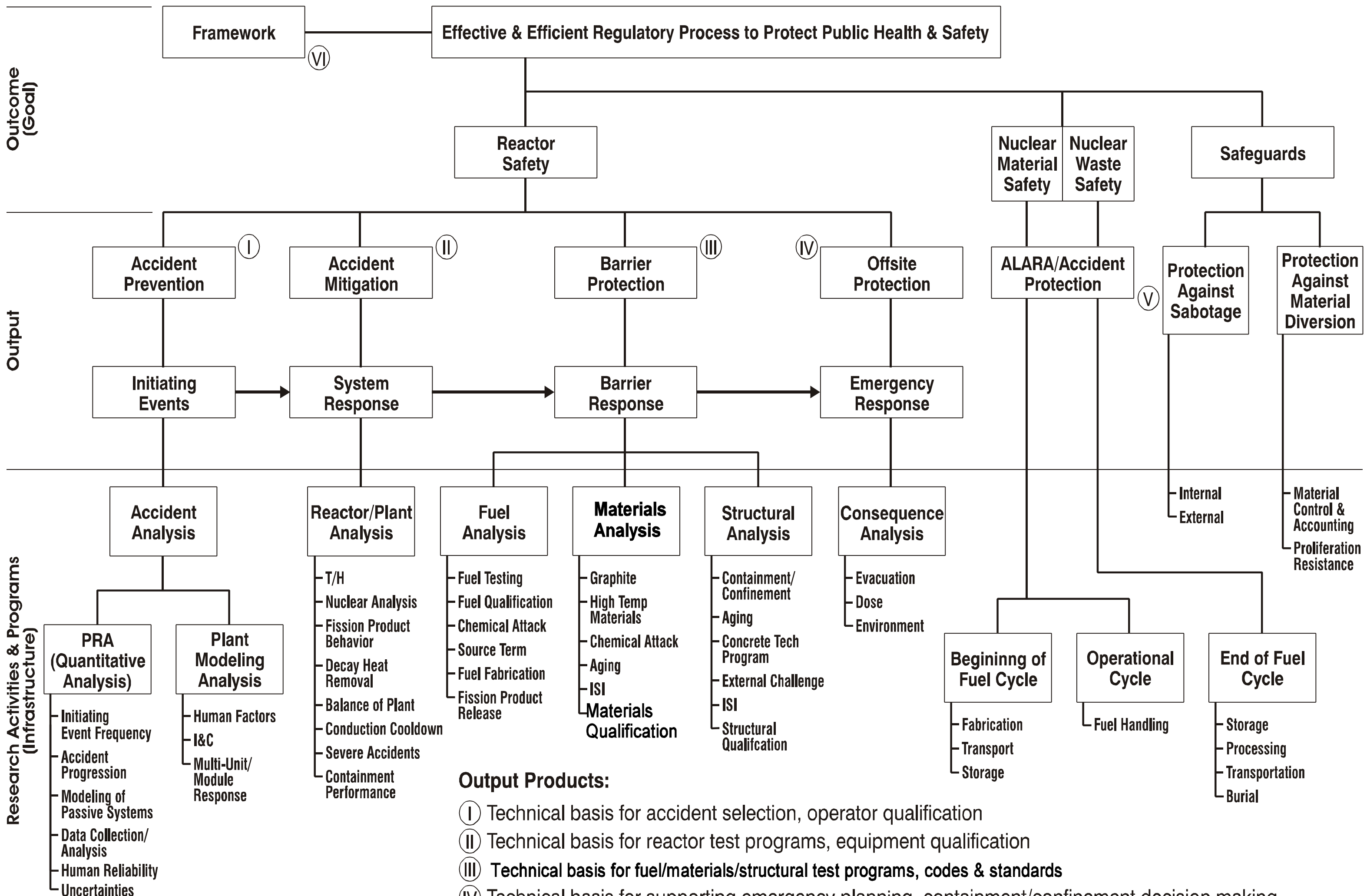
Discussions of anticipated NRC research activities and infrastructure needs associated with these arenas are provided in Section IV.3.

Safeguards and Security

Advanced reactor research efforts in safeguards and security will generally support other regulatory offices, principally NSIR. Research areas include proliferation potential and the evaluation of security measures, as well as the material control and accounting (MC&A) systems needed for preventing and detecting nuclear material diversion throughout the proposed advanced reactor fuel cycles. Brief discussions of anticipated research activities to support these regulatory domains are included in Section IV.4.

As requested by or through NSIR, RES will support NSIR and other offices and agencies with information needed for their assessments. This coordinated research support will be responsive to any new issues emerging from government-wide initiatives on Homeland Security.

Advanced Reactor Research Infrastructure
Key Research Areas and Areas for Examination



Output Products:

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

Figure 1 Key Research Areas for Examination

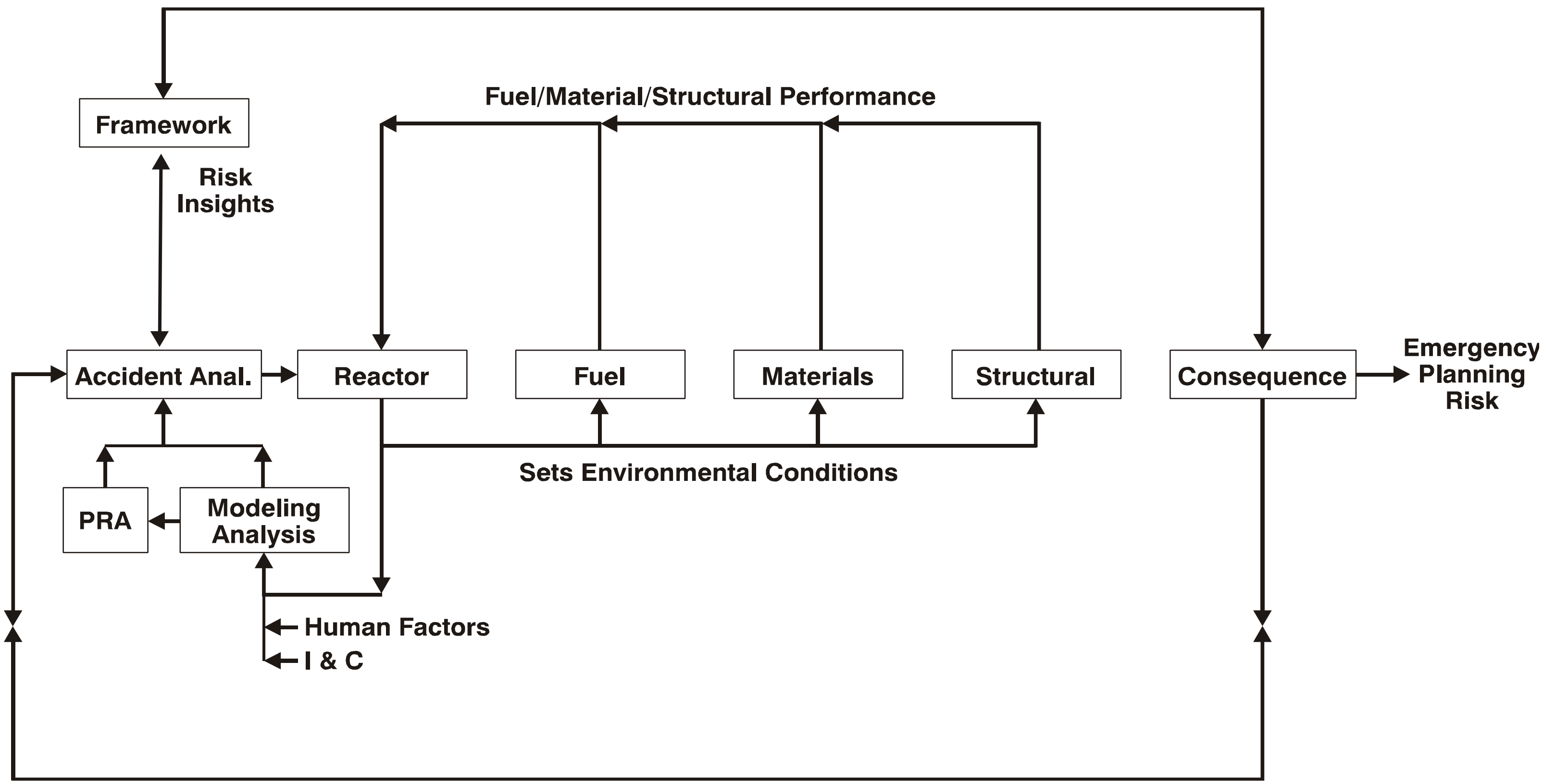


Figure 2 Information Flow Between Technical Groups

IV. KEY RESEARCH AREAS AND ACTIVITIES

IV.1 Generic Regulatory Framework Development

IV.1.1 Description of Issues

The NRC has over 40 years of nuclear power plant licensing and regulating experience, and this experience (e.g., regulations, regulatory guidance, policies and practices) has been focused primarily on LWRs with limited application to gas-cooled and advanced reactors. Advanced reactors will have design and operational issues associated with them that are technologically different from current LWR issues. However, NRC LWR experience can contribute and provide insights or “lessons learned.”

The most important insight from this experience is the recognition of the 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 will instill 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 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 designs currently under consideration. This approach, referred to as “technology-neutral,” would take 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

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 governing licensing these advanced reactors will be both risk-informed and performance-based. Both deterministic and probabilistic results and insights will be used to identify applicable regulations 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 the most risk-significant areas while maintaining basic principles, such as defense-in-depth and safety margin.

IV.1.3 Objectives and Associated 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, by focusing on the goal of protecting public health and safety and including the strategic performance goals of the NRC's Strategic Plan. These will also be used to assure that the framework is appropriately performance-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. This statement also expresses 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 the 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 4 is a general illustration of the development of the technology-neutral framework. The process starts by using safety criteria and regulatory guidelines determined to be applicable to advanced reactors, as well as 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. 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 essential. 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 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 licensing requirements. However, it is envisioned that certain reactor-specific regulatory areas may need to be addressed 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.

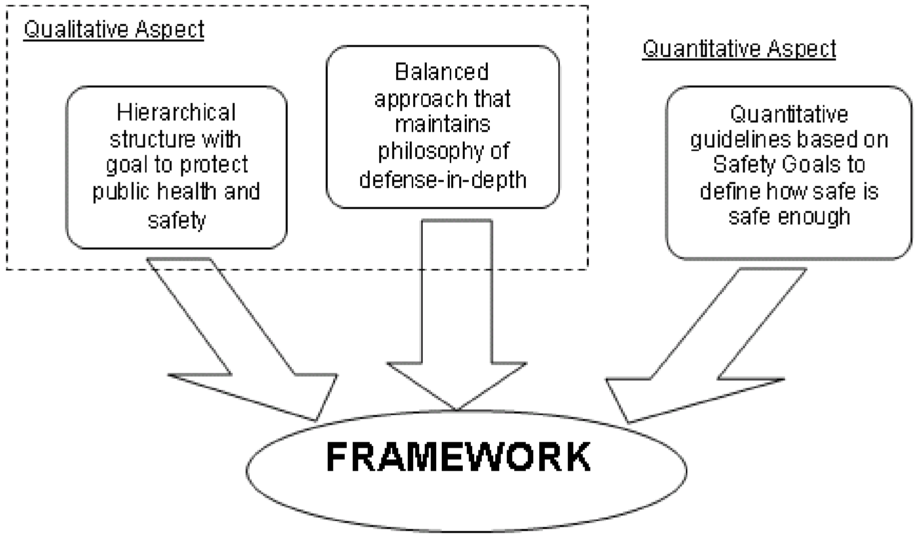


Figure 3 – Aspects of the Framework

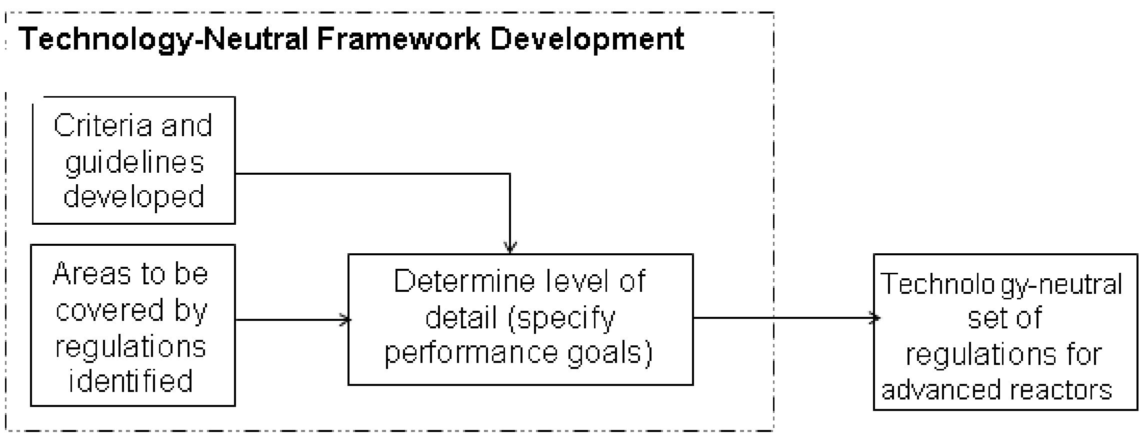


Figure 4 – Development of a Reactor-Neutral Licensing Framework

IV.2 Reactor Safety

IV.2.1 Accident Analysis

IV.2.1.1 Probabilistic Risk Assessment (PRA)

IV.2.1.1.1 Background

Future licensees have indicated that PRAs will be an integral part of their applications. 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 27 years, the NRC/Atomic Energy Commission has performed PRAs, and has promoted the use of PRAs 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 the PBMR, GT-MHR, and IRIS) are new designs and, therefore, the current PRA experience is limited. The limitations of current PRA experience applies to (1) system modeling approaches and associated underlying hypotheses (e.g., treatment of passive systems); (2) the 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); (3) failure data, and most importantly, (4) the design, materials, systems, and safety approach. These limitations need to be addressed as part of this work. Extensive use will be made of existing PRAs. The tools, expertise, and data (including information related to uncertainties) need to be developed to enable the staff to evaluate advanced reactor PRAs.

This work interfaces with virtually every other area of this infrastructure assessment. Given that PRA is an iterative process, knowledge of reactor systems, fuels, materials, human performance, and 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 infrastructure assessment.

IV.2.1.1.2 Purpose

The purpose of this work is to develop the methods, expertise, and technical basis needed for an independent staff review of a PRA submitted as part of an advanced reactor 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 based on sound engineering judgment; the approach to licensing was to provide safety margins and defense-in-depth. This approach to licensing can lead to unnecessary conservatism and may not have identified some sequences that could be important from a risk perspective. Experience 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 PRA tools, methods, and expertise, areas for which there is insufficient information (e.g., due to insufficient operating experience) will be identified. These areas need to be the subject of expert judgment or sensitivity studies to gain an understanding of the uncertainties.

The use of PRA is expected to increase in the licensing of advanced reactors. Applicants will provide arguments for the acceptability of their proposed advanced reactor design based on PRA results. Safety margins and defense-in-depth will be retained to protect the health and safety of the public. 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 reduce the conservatism traditionally provided. Therefore, developing the PRA tools, methods, and expertise is important for the review and licensing of these reactors. Having this capability enables the staff to make 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 MELCOR Accident Consequence Code System [MACCS2]).

IV.2.1.1.3 Objectives and Associated Activities

The objectives of the advanced reactor PRA work are to develop review guidance for NRC reviewers, explaining how to independently review advanced reactor PRAs and to support the development of a risk-informed regulatory framework. 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 infrastructure assessment is comprised of three tasks which will be undertaken concurrently. The first task is to develop the methods, data, and tools needed for evaluating the design and operational characteristics of advanced reactors that differ 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 infrastructure assessment, and (3) evaluate advanced reactor designs. Existing PRAs will be used to develop limited scope PRAs that can then be revised as plant-specific information becomes available. This process will identify areas in which additional research is needed and will provide an ability to prioritize needed research. The third task is to document the guidance necessary for the review of advanced reactor PRAs.

Task 1. PRA Development for Advanced Reactors

There are fundamental tasks that need to be performed to support either performing a 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 to the current generation of LWRs (e.g., loss of offsite power and seismic events) and some that are 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 need to be re-evaluated.

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, needs to be performed as part of thermal hydraulics (T/Hs) and severe accident work and should be fed into the PRA as part of the data necessary 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 (FPs). (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 is part of T/Hs and severe accident work, 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.

The source term issue is part of the T/Hs and severe accident work of this infrastructure assessment. The knowledge of fuel performance is a prerequisite to performing an independent review of the PRA. Core behavior in accidents, such as overheating or immersion in media (for example, in gas-cooled designs air, or if possible, water), needs to be understood. This behavior should be understood not only for fresh fuel but also for end-of-life fuel to evaluate the impact, if any, of burnup.

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 that 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 should be reconsidered to determine the need for potential modifications based on advanced reactor designs. This determination 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 infrastructure assessment. PRA modeling should 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 volts direct current instead of 120 volts) or radiation damage for fiber optic cables. Digital I&C could also fail sooner under fire or loss of cabinet cooling conditions. 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 needs to be addressed and quantified to the extent practical.

Data collection and analyses: Advanced reactors may introduce different systems and components, hence, LWR data may not be applicable to these new systems. 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 an important activity. 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.

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 proposed have a strong reliance on the premise that they will be free from human-error, 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 infrastructure assessment. 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. It is important to 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. There is also the potential of a reactivity insertion accident during, or as the result of, a seismic event, particularly for the PBMR and ACR-700.

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 T/H analyses. The Systems Analysis Program for Hands-on Integrated Reliability Evaluation (SAPHIRE) code could be used in the performance of an independent PRA but needs modifications for a full scope PRA (e.g., external and internal events, full and low power). A full scope PRA generates 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) that need to be evaluated as part of the severe accident and consequence work of this infrastructure assessment, should also be incorporated into a PRA tool. A full scope PRA tool that integrates Level 1 core damage frequency (CDF) analyses with Level 2 and Level 3 analyses, as well as dynamic modeling, is needed to provide the insights necessary 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 an approach for identifying the uncertainties associated with modeling and estimating risk. Three types of uncertainty exist: modeling, data, and completeness. Processes need to be developed to identify and understand the significance of the modeling and completeness uncertainties.

Other operational states: The 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. In some advanced reactor designs identified, 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 possible 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 CDF and large early release frequency may not be the best figures of merit for some advanced reactor designs. However, Level 3 PRA results (offsite consequences is part of the severe accident and consequence work of this infrastructure assessment) 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. A parallel effort with industry with an exchange of ideas can be useful. 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 in which 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 the PRA

The results developed in Task 1 will be used to (1) gain expertise, (2) provide guidance for assessments in other areas of this infrastructure assessment, and (3) develop an improved independent capability to evaluate advanced reactor PRAs. The level of detail is determined, in part, by the PRA information needed for supporting the licensing process and the timeliness of new information. The results could provide a basis for performing comparisons with advanced reactor PRAs submitted by the applicant.

Task 3. Documentation

The documentation include the identification of research needs, and will provide 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 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.
- Develop regulatory guides and SRP sections.

IV.2.1.1.4 Application of Research Results

This work will aid the following areas.

- 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 infrastructure assessment to help identify where there is inadequate information, and, thus, support staff decision making for research.
- 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 Background

The new generation of advanced reactors, both for HTGRs and ALWRs, will provide 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 U.S. 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 (ABWRs) 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 switchyard.

I&C systems play an important role both in reactor control and in providing information on the balance of the plant. 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, online monitoring methods, and enhanced cyber security. The NRC has recently started new research programs in the areas of wireless communications and online monitoring. This research will support the development of NRR review guidance 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 activities described in this section focus on knowledge and tools to support the review of new reactor technologies. In some cases, the research described in this section will be similar to ongoing research in support of digital upgrades to existing plants. Where appropriate, these activities will be coordinated to ensure that duplication of effort is minimized.

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

General Atomics (GA) is using plant simulators to help optimize control system designs. PBMR Corporation is also developing 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). An opportunity to collaborate on some of these research programs may exist, particularly in the areas of advanced control algorithms and control of multiple plant modules.

The DOE research program to support development and future use of nuclear energy in the U.S. 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 I&C systems for HTGRs.

IV.2.1.2.2 Purpose

Advanced reactors 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. Fewer operators will be needed compared to the current generation nuclear power plants (i.e., there may be as few as 3 operators for 10 modules). This will require that not only normal operations but off-normal operations and recovery be more highly automated. This will also require a level of automation and coordination that is more complex than that found in current generation plants.

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

To understand the more complicated digital I&C systems within a risk-informed licensing framework, additional risk modeling is necessary. These activities could support the research on 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 I&C systems will become more important as advanced designs perform more sophisticated safety and control functions.

IV.2.1.2.3 Objectives and Associated Activities

To expand NRC's knowledge and understanding in the I&C area, the following research areas have the highest importance.

Review of operating experience and lessons learned from ABWR and N4 control system development and regulatory review: Operating experience and design lessons learned will be reviewed to identify issues associated with digital systems and technology in current use in advanced reactors in other countries. Additionally, the review will identify the regulatory analysis methods and tools that have been used by foreign regulators. This information can be used to focus the remaining I&C research efforts based on the experience of others.

New risk models for I&C systems in advanced reactors: This effort would complement the work that is currently being done at the University of Maryland and the University of Virginia. Focus should be on the development of risk models for advanced reactor I&C systems that address the possible safety issues of the systems and that can be integrated into advanced reactor risk models. This research would permit incorporation of detailed I&C reliability models into scoping PRAs used for development of design basis accident scenarios. Research could also be used in the analysis of how these digital systems will affect plant responses to accidents.

Analysis of the requirements and potential issues involved with HTGR instruments: Existing requirements for design, construction, and operation of the HTGR will need to be compiled and evaluated. Information related to neutron detectors, particularly for PBMR, and temperature sensors would have a higher priority. This information would provide a better understanding of how the requirements were developed and what review methods are the most appropriate. Information gathering should focus on generic issues and not on viability or preferability of various technologies. This information would support the development of guidance for the regulatory review and qualification of these new instruments.

Development of models of autonomous control: Information and models are needed on autonomous control methods that could be used in advanced reactors. Information should be gathered on methods used in other industrial applications, such as natural gas power plants. This information would assist in the development of models for advanced reactor applications.

Analysis of control systems used to integrate the control of multiple module plants: The means and extent of integration of control systems in advanced reactors using multiple modules will need to be understood. The points at which control and safety systems are integrated, the nature and extent of automated actions, and the associated effects on plant response to transients and accidents will need to be evaluated to identify potential safety issues. This information would inform the review of these systems.

Analysis of online monitoring systems and methods and advanced diagnostic methods: Incorporation of online surveillance and maintenance capability may become an integral part of the system design for advanced reactors. For example, online monitoring systems for degradation, not just failures, could be used to continuously assess system performance to support longer inspection and maintenance intervals. Technologies for online monitoring and advanced diagnostic methods will need to be evaluated to identify critical attributes that should be addressed by reviewers. This information would be used in the review of advanced reactor technical specifications associated with surveillance requirements.

Review of advanced control algorithms for application to advanced reactors: Information is needed on current control algorithms likely to be used in advanced reactors and the potential issues associated with these algorithms when used in a reactor setting. This information would be used to support development of review guidance for evaluating the performance of these control systems in response to plant transients.

Analysis of the requirements and potential issues involved with advanced light-water cooled reactor instruments: Performance information will need to be developed for new neutron detectors expected to be used to support ultra-long life cores. Review guidance may need to be modified to support safety evaluations of these instruments. Design verification and validation for this equipment will present major challenges. This performance information would support the development of guidance for review and qualification of these new instruments.

IV.2.1.2.4 Application of Research Results

The lessons learned from the General Electric ABWR and the French N4 designs would provide insights and guidance to help identify those I&C systems and technologies that have been used in advanced reactors in other countries, as well as any operational issues related to those systems. The remaining work would provide independent tools and methods to assist in assessing new technology that will be an integral part of U.S. advanced reactors. These programs would provide information for revisions to Chapter 7 of the SRP and the supporting Regulatory Guides.

IV.2.1.3 Human Factors Considerations

IV.2.1.3.1 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 (1) human error is a significant contributor to CDF; (2) by improving human performance, licensees can substantially reduce their overall CDF; (3) a significant human contribution to risk is in failure to respond appropriately to accidents; and (4) human performance is important to the mitigation of and recovery from failures.

IV.2.1.3.2 Purpose

Advanced reactors are expected to present a concept of operations and maintenance to the staff that is different from what is currently the case at conventional reactors. Operators will be expected to concurrently control multiple modules, which may be in different operating states, from a common control room. Operators will need to monitor online refueling in one module, during normal operating states in other modules, while responding to a possible transient in yet another module. 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 qualifications may be required of the plant staff to maintain digital systems and to focus decision making on monitoring and bypassing automated 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. The 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 regulatory staff will be able to review applicants' 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, NRC staff reviews the human factors engineering (HFE) programs of applicants for construction permits, operating licenses, standard design certifications, and combined licenses. Under 10 CFR 50, the staff also reviews license amendments. These reviews help to 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; Design Implementation, and Human Performance Monitoring.

Current regulations and guidance (e.g.,: 10 CFR 26, 10 CFR 50, 10 CFR 52, and 10 CFR 55, Regulatory Guides 1.8, 1.134, and 1.149, NUREG-0700, NUREG-0899, and 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, new regulations and guidance may need to be created as newer reactor and control technology is developed and introduced, to address the new concept of operations. A sound technical basis should 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 in the system needs to be considered from the initial concept development stage so that the role is appropriate to the function eventually assigned, as specified in the Institute of Electrical and Electronics Engineers' (IEEE) Standard 1023.

To ensure that human factors activities are risk-informed, a close synergism with the human reliability analysis (HRA) aspects of this infrastructure assessment is necessary. 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 database necessary to adapt the HRA techniques to advanced reactors. HRA in turn can help prioritize the human factors efforts.

Currently no facility exists 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 N4 reactor and other concepts they are considering. Japan and Korea also have research simulators. 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) at their facility in Norway. These simulators can all be controlled through a common advanced design control room. The OECD cannot simulate any of the advanced plants (e.g., PBMR), but they do 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 Objectives and Associated 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 considered. Other tasks 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 the role humans play in advanced reactors may differ from their role in LWRs. Therefore, to develop a detailed human factors infrastructure assessment, the following must first be determined 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 should be performed. 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 of its impact on 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 online 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 as well as guidance to assist the regulatory staff in reviewing applicant submittals and developing a knowledge base for performing these 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, a systematic review of the existing licensing criteria to determine their applicability to advanced reactors would need to be performed. Rules, regulatory guides, NUREGs, the SRP, and consensus standards from IEEE and the American Nuclear Society/American National Standards Institute (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, etc.

Review existing human performance research facilities: It is important to understand the operator's role in the operation of advanced reactors, particularly because it is likely to 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 U.S. nuclear power sector, 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. Such a facility 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 earlier in Section IV.2.1.3.2 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 can be developed or adapted for use by the staff. The use of such analytical models could enhance the efficiency and effectiveness of 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 as well as to assess applicant submittals.

Staffing: Industry 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 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, consistent with the Systems Approach to Training. NUREG-1220 and inspection modules are used by the 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. If 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. Certain questions need to be considered: 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 available for paper-based emergency operating procedures only, 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 in the context of advanced reactor systems, because advanced reactors will have computer-based or glass cockpit control rooms, and their 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, guidance may need to be developed for certain issues not covered in NUREG-0700. These issues were not included in NUREG-0700, Rev. 2 because no validated criteria were available, and the technical basis on which to develop the criteria was not sufficient. 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 Application of Research Results

The result of the first effort listed will be an Insights Report to 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 analytic 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 analytic 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. 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 to 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 address infrastructure needs in the area of reactor systems analysis, which includes T/H analysis, nuclear analysis, and severe accident and source term analysis. For the T/H analysis of 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 reactor safety, materials safety, waste safety, and safeguards and security. Nuclear analysis research for reactor systems analysis will include the development and testing of: (1) 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 (2) neutron transport and shielding models as needed to analyze 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:

(1) evaluating the progression of credible severe accident scenarios involving core damage phenomena, such as fuel melting or high-temperature chemical attack and (2) modeling any resulting releases and transport of radioactive fission products (FPs) within and outside the reactor system boundaries.

In advanced HTGR designs, the integrity of the coated particle fuel in its function as primary FP 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 T/Hs. So-called melt-wire experiments performed in Germany's Arbeitsgemeinschaft Versuchsreaktor reactor (AVR) 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 FP 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, T/Hs, severe accidents, and mechanistic release of FPs), 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), thereby giving 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 (1) conduct confirmatory analyses in the review of applicants' reactor safety analyses, (2) support development of the regulatory framework by assisting, for example, in the identification of safety-significant design-basis and licensing-basis events, and (3) 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 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 needed, applicants typically apply 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. T/H analysis is also used in the context of PRA to determine the best estimate of system states, thereby supporting analyses of the mechanisms and probabilities for system failures.

IV.2.2.1.2 Purpose

T/H analyses are typically used to (1) assess what safety limits are needed and whether limits and margins such as fuel design limits are met, (2) predict transient effects on system components and materials, and (3) 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.

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 is typical in LWR analyses. Some design-basis transients are driven by radiative and conductive heat transfer through porous and solid structures, not convection. Although this parameter currently exists in all codes, it 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 for 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 MELCOR, and the general purpose computational fluid dynamics codes, such as FLUENT. The reactor system analysis code for HTGR applications will be built upon the existing MELCOR code. Also, as discussed in this infrastructure assessment (see Section IV.2.2.3 on Severe Accident and Source Term Analysis), the MELCOR code will be used in conjunction with FLUENT for analyzing events that cause core damage (e.g., air ingress with significant graphite oxidation).

Where appropriate, the development of new capabilities in MELCOR will use or build upon corresponding features in the two earlier HTGR accident analysis codes, Graphite Reactor Severe Accident Code (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). The 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, also 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 MELCOR will be the best use of agency resources. The MELCOR code already possesses many of the features discussed above, the staff owns and controls the MELCOR source code, and, given the code's modular structure, new capabilities can be added with relative ease. For example, MELCOR 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 MELCOR (for example, modeling helium turbines), adding this capability can be readily achieved by changing one or more of the MELCOR 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 provides the ability to more reliably predict parts of the fluid system when it is necessary to assess the capability of the reactor system code against some assumed known reference standard or when it is necessary 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 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 Japan Atomic Energy Research Institute (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 T/Hs data. However, additional data is needed to investigate issues including the pebble-bed hot spots inferred from the melt-wire test results at AVR, the 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. The NRC staff will initiate cooperative efforts with the international community to identify data needs and to 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 considered for research:

- Confirm and modify as needed the capability to model flow and heat transfer in packed beds. The solver in MELCOR 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, the turbine model will need to be changed 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, add the ability to track multiple noncondensable gas sources.

- Assess code speed 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 evaluated for partitioning the analysis into time periods in which 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 phenomena identification and ranking table (PIRT) process, the information developed as part of previous HTGR programs, and the IAEA review of data to develop data needs for code development and assessment.

- 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.

- 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.

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 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 T/Hs is necessary in order to confirm and quantify the margins to failure for proposed ALWRs. This section discusses those features and the T/H issues for advanced light-water reactors.

Three advanced LWR systems (the AP-1000, ESBWR, and IRIS) and a heavy water moderated reactor system (ACR-700) are discussed. All 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 the safety margin to core damage depends on assessing the performance of these passive systems and quantifying uncertainties associated with the T/H processes used.

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, thereby eliminating the possibility of a large LOCA. In addition, the IRIS integral reactor vessel 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. The integral reactor vessel also provides a large inventory of water above the reactor core, which slows the reactor response to transients and postulated small LOCAs. However, a main steamline break or ATWS scenario may need to be considered in more detail.

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:

- <u>Entrainment from horizontal stratified flow</u>. 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 automatic depressurization system (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 Advanced Plant Experiment (APEX) facility that have been planned by DOE will also provide data useful in evaluating this process.

- <u>Upper plenum pool entrainment and de-entrainment</u>. 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 in which the two-phase level drops below the bottom of the hot legs. Experimental data for prototypical upper plenum geometry is needed, as well as analytical models to account for entrainment and de-entrainment in the upper plenum. The DOE-planned 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.

- <u>Low pressure critical flow</u>. 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 for fourth stage ADS testing prior to operation. Currently, NRC is sponsoring experimental work at Purdue University using the Purdue University Multi-Dimensional Integral Test Assembly (PUMA) facility to obtain this confirmatory data.

- <u>Direct vessel injection</u>. 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 addressed 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 ESBWR include:

- <u>Multi-Dimensional Natural Circulation With or Without Boiling</u>. During steady-state, normal full-power operation, subcooled water in the reactor vessel downcomer drives the coolant upward in the core, where boiling takes place. The two-phase mixture continues the upward flow into the chimney region and then enters the stand pipes and steam separators, where steam and water are separated. Steam continues the upward flow into the dryer and exits the vessel via the main steam lines. The water separated in the steam separators drains downward into the downcomer annulus and mixes with the feedwater flow. The mixed flow then enters the lower plenum and the core to repeat the process.

During the blowdown phase of a loss-of-coolant accident (LOCA) or a transient that leads to the activation of the automatic depressurization system, the natural circulation flow is interrupted. After the vessel blowdown, the Gravity-Driven Cooling System (GDCS) water is intended to drain into the vessel downcomer and mix with the remaining water there before entering the lower plenum and the core. The vessel pressure will be reduced because of void collapse and less boiling in the core. In addition, water (if any) from the passive containment cooling system (PCCS) drain tanks is allowed to drain into the vessel downcomer. Meanwhile, the isolation condenser system (ICS) condensers also play a role to reduce the vessel pressure by condensing the steam (depending on the venting of the noncondensable gas) and returning the condensate to the downcomer. During the draining of the GDCS pools and the PCCS drain tanks, natural circulation with or without boiling in the core exists in the reactor vessel. Since the core is at decay power, the core flow rate is expected to be much smaller than in normal full-power operation. If the two-phase mixture level is below the inlet of the steam separators, the loop-like circulation described above for normal operation does not exist. Instead, the flow in the core may be either all upward flow or a combination of upward flow in certain regions and downward flow in the rest of the core. The natural circulation flow in the ESBWR during a LOCA (or during normal operation) is multi-dimensional. One-dimensional modeling would therefore be inadequate. The natural circulation flow determines the core inlet mass flow rate and subcooling, which are needed (along with other parameters) to predict the steam generation rate in the core or the steam flow rate from the vessel to the containment. As a result, the containment pressure and the vessel pressure are affected by the natural circulation in the vessel. The draining rate of the GDCS pools or the PCCS drain tanks can also be affected (through vessel pressure).

Adequate models, assessed against experimental data, are therefore needed to address the natural circulation flow during normal operation, LOCAs, and startup. Furthermore, startup stability involving flow and power oscillations may become an issue of the natural circulation reactors. This issue is discussed further in Section 3.2, Nuclear Analysis.

- Two-Phase Mixture Level. Two-phase mixture level in the reactor vessel is important because it determines whether the core remains covered during a LOCA or a transient leading to actuation of the automatic depressurization system (ADS). If the core remains covered, the peak clad temperature of fuel rods will be close to the liquid saturation temperature. Adequate models are therefore needed for the two-phase mixture level in the reactor vessel.

  Similar models are also needed to calculate the two-phase mixture level in a PCCS pool. During a LOCA, the water level in the PCCS pool will decrease because of the boiling in the pool (on the outer surface of the condenser tubes). This can eventually lead to the uncovery of the condenser tubes. As a result, the PCCS heat removal rate can be reduced, and the drywell pressure will be adversely affected.

Similar models are also needed to calculate the two-phase mixture level in an ICS pool during either a LOCA or a non-LOCA transient for which the decay heat removal is carried out by the isolation condensers. If the condenser tubes become uncovered because of the boiling in the pool, the heat removal rate of the isolation condensers will decrease and the vessel pressure will be adversely affected.

- <u>Steam Condensation with the Presence of Noncondensable Gas</u>. Condensation of steam with the presence of noncondensable gas (nitrogen or hydrogen) occurs on the inside walls of the vertical PCCS condenser tubes during a LOCA or a transient with ADS actuation. It can also occur on the drywell walls. The presence of the noncondensable gas may significantly reduce the heat removal rate of the PCCS condensers or the steam condensation rate on the drywell walls. An adequate model for this phenomena is therefore required in order to determine the drywell pressure.

  Condensation of steam with the presence of noncondensable gas can occur in the suppression pool, when the PCCS condensers vent a mixture of steam and noncondensable gas into the pool. This process is different from the condensation on a wall, and it involves the condensation of a submerged plume or jet of the gas mixture in a pool. If the steam is completely condensed before reaching the pool surface, its contribution to the wetwell pressure rise is reduced.

- <u>Gas Stratification and Mixing in the Drywell</u>. Since the presence of noncondensable gas may significantly reduce the heat removal rate of the PCCS condensers for which the inlet is located near the top of the drywell, the code needs to adequately predict the spatial distribution of the noncondensable gas in the drywell. This phenomenon depends on gas stratification (from density or temperature difference) and mixing in the drywell. Adequate models are therefore needed to handle drywell gas stratification and mixing, which could affect the PCCS heat removal rate and the containment pressure.

- <u>Thermal Stratification and Mixing in the Suppression Pool</u>. Thermal stratification in the suppression pool affects the wetwell pressure because the temperature of the top layer of the pool determines the steam saturation temperature in the gas space above the pool surface. The wetwell pressure is the sum of the steam pressure and the noncondensable gas pressure in the gas space above the pool. Adequate models for this phenomena are therefore needed to determine the wetwell pressure.

  Thermal stratification in the pool water depends on the mixing process that is produced either during the clearing of the main vents (between the drywell and the wetwell), or during the venting of the PCCS or ICS condensers. The phenomena is therefore related to Item 3 above (steam condensation with the presence of noncondensable gas).

- <u>Boron Mixing in the Core</u>. When the control rods fail to be inserted into the core during an anticipated transient without scram (ATWS), boron solution will be injected into the core bypass region. The effective shutdown of the reactor depends on how well the boron solution is mixed with the coolant in the core. An adequate model for this phenomenon is needed to address the ATWS. A computational fluid dynamics code can be used to analyze this process, if necessary.

- Dynamic Response and System Interactions. Adequate models are needed to analyze the dynamic response of the passive safety systems and system interactions. Examples include the GDCS injection rate to the vessel (depending on water level in the GDCS pools, the wetwell pressure, the reactor vessel pressure, and flow resistance of the injection line); the PCCS drain tank injection rate to the vessel; adverse impact of a leaking vacuum breaker (between the drywell and wetwell) upon PCCS performance; and the impact of intermittent venting or containment heat loss upon PCCS performance.

The major T/H issues for ACR-700 include:

- Quenching of Horizontal Fuel Bundles. An assessment of the TRAC-M heat transfer models will be conducted to determine their applicability to the ACR-700. Some postulated accidents lead to fuel heatup where emergency core cooling water is relied on to return the fuel temperatures to a subcooled state. Those horizontal fuel pins that are well above the horizontal water level are surrounded by steam and radiate much of their heat to the pressure tube wall and onto the calandria tube which is cooled by the moderator water in the calandria tank. Just above the water level is a two-phase region caused by the rod quenching process as the water level rises following emergency core coolant (ECC) injection.

- Flow Rates from Headers to Feeders. In the ACR-700 design, the pumps supply water to horizontal inlet headers. Feeder pipes are welded at various elevations on the lower half of each header to supply the fuel channels. An outlet feeder pipe from each channel connects to one of the outlet headers. When the two-phases in a header are stratified, cooling of a fuel channel is influenced by the elevation of its feeder connection on the header. When flow is out of the header, feeders connected near the bottom of the header receive water while those connected higher up on the header may receive steam. Special models are needed to treat this phenomenon.

- Energy Transfer from Pressure Tube to Calandria Tube. During normal reactor operation, the gas gap between the pressure tube and the calandria tube insulates the hot primary fluid from the cold, low-pressure water in the moderator (calandria) tank. A heat exchanger is connected to the calandria vessel to provide moderator cooling. During some accidents, the moderator can act as an important heat sink if the pressure tube gets hot enough to sag and press against the calandria tube. The sagging and the increase in thermal conductance must be modeled in the codes.

- Thermal-hydraulic Phenomena in the Calandria Vessel. During steady-state operation, a detailed model of the calandria tank is probably not necessary because the energy transfer process is slow enough that the moderator heavy water can stay fairly well mixed. However, if a pressure tube should rupture, an accurate model is needed to determine the course of the accident. Complete condensation of the break effluent steam will occur if the pressure tube is sufficiently submerged. If a tube near the top of the moderator tank ruptures, thermal stratification could lead to incomplete condensation and over-pressurization of the tank. Rupture discs will then break, allowing the tank to blow down.

- Natural Circulation Flow and Heat Transfer. Natural circulation flow rates and heat transfer around the primary loop and on the secondary side of the steam generator can be difficult to model. Flow between an inlet header and an outlet header has dozens of parallel flow paths to take. When the reactor is in a cool-down mode, with the primary pumps off and ECC on, the flow may be forward in one fuel channel and reversed in an adjacent fuel channel. This has been observed in the RD-14M experiment and is determined by gravity head and steam generation or condensation rate differences between channels.

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 coil has an outer diameter of approximately 1.6 meters (m). During LOCAs, heat transfer by the steam generators is 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 needed 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 percent 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 IRIS' 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 evaluation of 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 system 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 Objectives and Associated Activities

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 MELCOR development and assessment efforts. Such development and assessment support will include: (1) adapting or building upon, where appropriate, selected GRSAC methods and data for use by MELCOR (i.e., as an alternative to reinventing them for MELCOR), and (2) comparing detailed GRSAC and MELCOR results on reference HTGR transients and resolving the causes of any major discrepancies. An effort to modify MELCOR to add currently identified capabilities is being initiated at Sandia National Laboratory.

Related international research: The IAEA sponsored an international standard problem modeling the conduction cooldown of an HTGR. Specifically, this effort was directed at modeling passive heat removal systems and highlighted the importance of accurate modeling of heat sources and the difficulties in modeling these passive systems. The results of this study are documented in IAEA's technical document, TECHDOC-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 to focus the review of previous HTGR programs and the IAEA review of data on developing data needs for code development and assessment. This effort will also include collaborative efforts with the international community.

The NRC has maintained an active, confirmatory T/Hs research program to better understand phenomena that are important to advanced passive plants such as the AP-1000. The experimental program conducted at Oregon State University using the APEX facility has been central to this effort. 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 a 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 State University to investigate entrainment from the hot leg to branch lines. Both of these facilities are expected to yield experimental data important to 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.

Identified NRC research activities: NRC needs an independent capability for HTGR T/Hs analyses that has been thoroughly assessed and peer reviewed. The first priority for this effort will be focused on adding the necessary capability for HTGR analysis to MELCOR. 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. The following issues should be considered as part of the MELCOR development effort:

  - 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 noncondensable 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 software qualification assurance (SQA) documentation for HTGR analysis.

  PIRT analysis. An analysis using PIRT methodology on T/Hs data and modeling needs for HTGRs needs to be conducted. The analysis will include issues and sequences raised in early analysis for the workshop. The deliverables for this task will include an identification and ranking by safety significance of the NRC data and modeling needs in the area of T/Hs for HTGRs.

  Database development. Needed data, based on the analysis of the HTGR's designs and analysis methods, should be developed including the development of test facilities to collect information required to complete code validations. Appropriate data will also be collected for input deck preparation. Task deliverables will include reports describing the facilities and the relevant data.

- Advanced Light-Water Reactors. The NRC research objectives are to perform the experimentation and code development necessary to validate the success criteria for conditions or accident scenarios that are risk-significant. 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 fulfill 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.

  For ESBWR, data from the basic and separate-effects tests are needed to assess and improve TRAC-M models for these important phenomena. Full-size component test data from PANTHERS for the PCCS condensers and ICS condensers currently exists. The existing database including that from GE[1] will be reviewed and selected for TRAC-M model assessment and improvements. There are integral-systems test data from PANDA and GIRAFFE, which include the late blowdown phase, GDCS water injection into the vessel, and PCCS and ICS operation for long-term cooling. The code needs to be assessed against these integral test data to demonstrate its adequacy in predicting the operation of the passive safety systems (GDCS, PCCS, and ICS) under a small driving force. Furthermore, natural circulation stability test data exists from CRIEPI and PUMA. TRAC-M needs to be assessed against these data.

  PUMA Integral Experiments. Integral tests at PUMA will complement the existing GE's integral test data from PANDA and GIRAFFE. These PUMA tests are listed below.

  - PUMA counterpart tests (under test conditions similar to those in PANDA or GIRAFFE) to provide data from a different scaling facility (quarter-height in PUMA vs. full-height in PANDA and GIRAFFE).

  - PUMA system interaction tests to investigate potential interactions between passive safety systems or between a passive safety system and an active non-safety system. The purpose is to determine whether the performance of a passive safety system is adversely impacted from the operation of either an active non-safety system or other passive safety systems and components. These tests will be identified after the PIRTs have been reviewed by the staff.

  - PUMA multiple failure tests (e.g., concurrent failure of a vacuum breaker and a PCCS condenser) to explore the safety margin of the design and the conditions that may lead to core damage.

To meet experimentation and code development objectives for validating ACR-700 and IRIS success criteria, 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 the safety case, 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. These tests will support the regulatory

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[1]GE Nuclear Energy, "ESBWR Reports," August 30, 2002.

decision-making process with safety-related information beyond that provided in an applicant's submittal. Improved models for two-phase flow and heat transfer in helical coils for the IRIS design and T/H phenomena for ACR-700 horizontal cores will 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 noncondensable distribution would need to be assessed and improved.

APEX-AP-1000 Confirmatory Integral Testing: Data for code validation and confirmation of safety margins needs to be provided. The APEX facility (currently being upgraded to represent AP-1000) will be used to develop an independent set of experimental data that NRC can use 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 outside the scope normally addressed by the applicant. The tests, currently planned by DOE, are designed to confirm the safety margin that is expected in the AP-1000 design, and will help identify any new processes or concerns not adequately addressed by T/H codes. The deliverables for this task include experimental data and evaluation reports describing the tests themselves.

> ***AP-1000 model development and separate effects testing.*** Experimental data needs to be obtained to develop T/H models for phase separation in the hot leg–branch line connection. These models are necessary to benchmark analyses in support of AP-1000. Separate effects test data, technical reports describing the data, and a technical evaluation report describing T/H models and the correlations developed from the data needed to represent important AP-1000 processes will be developed. This work is ongoing at Oregon State University.
>
> ***AP-1000 code development and assessment*****.** The TRAC-M code for large and small break LOCA analysis in AP-1000 will be assessed to ensure that TRAC-M can produce reliable results for the AP-1000. These results must be suitable to confirm licensing calculations and to explore beyond-design-basis behavior of the plant. The main objective of this task is to qualify TRAC-M for independent assessment of AP-1000 behavior during a large-break LOCA, a small-break LOCA, and long-term cooling. Task 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*****.** Special models (or an initial model if data is insufficient), will be developed and an initial independent assessment of IRIS behavior will need to be performed using a 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. The 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 for this task include IRIS plant input deck and a 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 Steam Generator (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 to be insufficient 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 pounds per square inch [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 analyses 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 initially constructed by industry and later by NRC staff.

IRIS code and model development. The TRAC-M code for LOCA (and possibly steam generator tube rupture) analysis in IRIS must be assessed to ensure that TRAC-M can produce reliable results for IRIS. These results must be suitable to evaluate licensing calculations and to explore beyond-design-basis behavior of the proposed design. The main objective of this task 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. Task deliverables include code validation reports, a code version validated for the IRIS plant design, and several re-calculations of the IRIS plant using a more refined code version.

IV.2.2.1.4 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.

As outlined in the preceding sections, the T/H 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 Background

The term "nuclear analysis" describes all analyses that address the interactions of nuclear radiation with matter. Nuclear analysis encompasses: (1) fission reactor neutronics, both static and dynamic; (2) 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; (3) radiation transport and attenuation as applied to the evaluation of material damage fluence, material dosimetry, material activation, and radiation protection, and (4) nuclear criticality safety (i.e., the prevention and mitigation of critical fission chain reactions ($k_{eff} \geq 1$) outside reactors).

This section addresses nuclear analysis infrastructure needs encountered in the evaluation of reactor safety. These nuclear analysis needs 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 document.

IV.2.2.2.2 Purpose

The purpose of the research activities described in this section is to provide the nuclear analysis tools, data, and knowledge bases that may be needed to support the staff's safety evaluations for the respective advanced designs. In identifying the 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 Evaluated Nuclear Data File, Volume B (ENDF/B) in the United States; Joint European File (JEF) in Europe, or Japanese Evaluated Nuclear Data Library (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 FP 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. Such techniques, developed in recent years under RES sponsorship, can 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 Very

Superior Old Programs 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, Purdue Advanced Reactor Core Simulator (PARCS), and for the criticality, depletion, and shielding analysis sequences in the NRC's Shielding and Criticality Analysis for Licensing Evaluation (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 re-evaluation 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 A Modular Code System for Processing Xsections (AMPX) code suite. This upgrade will 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, JEF-3 and JENDL-3. With the recently completed AMPX upgrades and continued improvements to the Los Alamos National Laboratory's Nuclear Data Processing System codes (NJOY), 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.

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)(less than 20 percent ^{235}U) instead of the high-enrichment uranium (HEU)(more than 90 percent ^{235}U) and thorium used in earlier HTGRs. Both designs 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 online fueling; its pebble-bed annular core with statistical packings of fuel pebbles of varying burnups; the intermixing of graphite pebbles and fuel pebbles near the boundaries between the fueled core region and the central graphite region; and the potential for seismic compaction events, misloading events, anomalous local packing and clustering of pebbles, and anomalous flow patterns of pebbles through the core such as might be caused by localized pebble bridging; jamming of chipped or fractured pebbles; unanticipated funneling effects near the core exit, or unanticipated radial gradients of pebble flow velocity resulting from the strong temperature dependence of pebble-to-pebble friction (i.e., as seen in the THTR-300 pebble bed reactor). Related research activities on the mechanics of pebble beds, including pebble flow and intermixing, statistical packing, bridging, and seismic pebble-bed compaction, are included in the Materials Analysis section (Section IV.2.4).

Physics analysis issues unique to the GT-MHR relate mainly to the effects of burnable poisons, the presence of both 19.9 percent 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 infrastructure development that could be useful in evaluations of PBMR and GT-MHR reactor safety, and related aspects of tri-isotropic (TRISO) fuel performance, include the following:

- Temperature coefficients of reactivity. The ability in the from of expertise and tools may be 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 may need to 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, 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 need 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 may be needed to assess 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 licensing review 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 IV.2.3, Fuel Analysis). 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 nominally stated as 80 gigawatt days per ton (GWd/t). Therefore, determining a suitable value for discharge/recycle burnup criterion (i.e., <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 percent-enrichment fuel pebbles as to the 8 percent-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 percent-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. Such hot spots can be used to 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, data on the effect of decay-power hot spots, in particular, may be needed in evaluating the maximum fuel temperatures arising in pressurized or depressurized LOCAs.

- 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 "N" Particle (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 LOCAs (e.g., conduction cooldown events). This high-temperature annealing heat source may need to 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 FPs 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 may be needed to evaluate how radioactive decay changes the fuel's inventory of important actinides and FPs (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 that this nuclear analysis issue relates directly to fuel analysis issues described in Section IV.2.3.)

- 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 non-prototypicality 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 may need to 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., FP retention) under normal and accident conditions. Such differences include the variations 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 FPs. It is known, for example, that ^{235}U and ^{239}Pu give substantially different yields of various FPs that potentially affect TRISO fuel performance. (Note that this nuclear analysis issue relates directly to fuel analysis issues described in Section IV.2.3.)

ALWR core neutronics and decay heat generation: Reactor neutronics and decay heat analysis issues for the AP-1000 are essentially identical to those for the AP-600 and the current generation of PWRs with respect to, for example, their gradual evolution to the higher initial enrichments and new burnable poison designs needed for higher burnups and longer cycles.

ESBWR uses shorter (by 2 feet) fuel assemblies and larger control rods (for reducing the number of control rod drives), compared to the operating BWRs. The current reference fuel for ESBWR is the GE12 fuel assembly of a 10x10 fuel rod array, which consists of 78 full-length fuel rods, 14 part-length fuel rods, and 2 large water rods (with each water rod occupying the space of 4 fuel rods) near the center. The control rods are cruciform blades located between a group of 16 fuel assemblies. The existing physics database is extensive. Some code work (e.g., generation of cross sections) is needed using TRAC-M that includes the PARCS reactor kinetics code to analyze the ESBWR transients in which neutronics becomes important.

During the reactor startup, flow and power oscillations may occur when the vessel is at low pressure with low natural circulation flow. Since reactor kinetics are coupled with thermal hydraulics, this stability issue and other issues such as ATWS will be analyzed by TRAC-M, including the PARCS reactor kinetics code.

Neutronics and decay heat analysis issues specific to the IRIS design include the following:

- Fuel depletion modeling. Initially, IRIS designers indicated that initial enrichment could be greater than 5 percent. If this is the case, then greater than 5 percent depletion analysis of the IRIS fuel designs with their 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) may be needed to assess such depletion modeling issues to develop appropriate technical guidance.

- Fuel depletion validation. The available experimental database for validating LWR fuel depletion analysis methods consists largely of destructive radiochemical assays performed in the 1970s and 1980s on rod segments from about a dozen discharged PWR and BWR fuel assemblies. The database includes very limited data from fuel rods with integral burnable poisons, initial enrichments above 4 percent, or burnups beyond 40 GWd/t. Sensitivity analyses, based on methods developed in recent years under RES sponsorship, may be needed to help assess the applicability of the existing databases to validate IRIS fuel designs and to assist in the prioritization of further data needs and the estimation of remaining validation uncertainties.

- Neutronics of high-burnup cores. The initial 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, may have greater neutronic 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 such as effective-delayed neutron fraction and prompt neutron lifetime.

- Decay heat power. Due to depletion modeling issues and the apparent shortage of available radioisotopic or calorimetric validation data applicable to the IRIS fuel designs at high burnup, 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 concerns may include, 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 Objectives and Associated Activities

The NRC research objectives are to establish and qualify the independent nuclear analysis capabilities that may be 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 associated NRC research efforts include the following:

- RES in-house analysis and contractor projects conducted in the late 1980s and early 1990s 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) developing sensitivity and uncertainty analysis methods that use cross-section covariance data; (3) developing modeling and validation guidance for computing radionuclide inventories in high-burnup LWR fuels; and (4) developing guidance on modeling and validation uncertainties in computing the reactivity of spent PWR fuel.

- Ongoing RES projects and tasks on (1) Modular HTGR Accident Analysis (ORNL); (2) MELCOR 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 (1) develop two-dimensional-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 (2) prepare design-specific nodal physics data tables for input to the NRC's PARCS spatial kinetics code.

For ACR-700, several major design changes factor into meeting a stated design goal to eliminate positive coolant void reactivity. These will require additions to the databases needed for validating the nuclear analysis codes and methods.

During the pre-application review, the vendor's existing and planned benchmark databases for the ACR-700 nuclear analysis will be assessed. Depending on the prototypicality and ranges of parameters covered by the vendor's benchmark experiments and tests, the need to develop additional data will be identified.

Anticipated analytical needs in the reactor physics area for ACR-700 safety analysis include utilization of three codes include (1) a lattice physics depletion code to calculate few-group nodal data for the ACR lattice; (2) a code to analyze the range of expected core operating configurations resulting from on-line fueling of the initial, transitional, and equilibrium ACR cores; and (3) a code to solve the steady-state and transient neutron flux distributions in the core. A three-dimensional neutron kinetics capability can address the need to properly analyze the behavior of the ACR-700 reactor core during events that result in large changes in the spatial neutron flux distribution. This would necessitate code modifications and model development and testing with the existing NRC spatial kinetics code, PARCS, as well as coupling of the resulting kinetics models to the corresponding ACR-700 thermal hydraulics models in TRAC-M. The necessary PARCS modifications and models will depend on the method selected for lattice-averaged cross-section generation.

The ACR-700 lattice differs significantly in design from LWR lattices. LWR fuel pins are laid out in square-grid assemblies with small water gaps between assemblies. ACR-700 fuel pins are laid out in rings within a cylindrical fuel element that is contained in a pressure tube. Some of the ACR-700 fuel pins contain dysprosia, an integral absorber not used in LWRs. The pressure tubes pass through a calandria vessel filled with a heavy-water moderator at low temperature and pressure. As for LWRs, the ACR-700 lattice-averaged cross sections would require a two-dimensional treatment of the geometry.

Most existing lattice physics codes developed for LWRs, including the NEWT-based code now being developed under NRC sponsorship rely on two-dimensional geometric approximations to model the transverse orientations of the absorber rods and the poison injection lines and three-dimensional effects from the axial water gaps between fuel bundles in a channel. The use of such cross sections in the spatial statics and kinetics computations must be validated against experimental data to determine the need for further development.

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

- Establishment of a cooperative research agreement with the Massachusetts Institute of Technology (MIT) that includes sharing of pebble-bed reactor physics codes, models, and related code development and analysis tasks (e.g., involving the Pebble Bed [PEBBED] and MCNP codes).

- Acquisition of HTGR physics benchmark data from the international High Temperature Reactor Configuration of the Proteus Critical Experimental Facility in Switzerland (HTR-PROTEUS) program conducted in the early 1990s at the Paul Sherer Institute (PSI), Switzerland. (Parameters included room temperature only, ordered and random pebble beds, 15–20 percent-enriched LEU fuel, Plutonium sample worths, moisture ingress worths, in-reflector absorber worths.)

- Acquisition of HTGR physics benchmark data from Russia, including the GROG experiments and the Advanced Gas Reactor in Kurchatov Institute, Russia (ASTRA) pebble-bed experiments, as well as any newer physics experiments supporting the design and safety analysis for the plutonium-burning GT-MHR in Russia. (Also acquire pulsed test data on fresh high-temperature reactor (HTR) fuel.)

- Evaluation of feasibility and technical merits of acquiring existing benchmark data from British Magnox, AGR, and early HTR programs, including a critical experiment facility in the United Kingdom (BICEP), Dungeness B, and various HTGR-related experiments done in the 1970s by Winfrith and British Energy.

- Where relevant, acquisition of existing HTGR physics benchmark and test data from Fort Saint Vrain testing and operations, the Compact Nuclear Power Source experiments at Los Alamos National Laboratory, the THTR-300 testing and operations, AVR testing and operations, the KAHTR experiments in Germany, and the CESAR experiments in France.

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

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

- Collaborate on and addition of new physics benchmarking activities to the IAEA's ongoing coordinated research project (CRP) on HTGR safety performance. 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 technical document (TECHDOC) and its references.

- Participation 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.)

- Participation in selected existing and planned HTR-N activities of the E.C.

- Participation 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, consolidation of 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 could be useful. RES staff could seek interoffice cooperation with staff in NRR and NMSS (Spent Fuel Project Office and Division of Waste Management), 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 Commissariaat à l'Energie

Atomique/Cadarache in France. 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 E.C., and OECD/NEA.

Identified research activities: Listed below are the potential research and infrastructure developmental activities pertaining to the nuclear analysis issues described previously.

- Preparation of modern cross-section libraries. The upgraded AMPX code system, supplemented by NJOY could be used to prepare state-of-the-art master cross-section libraries for use in performing exploratory and confirmatory analyses on reactor safety and material safety issues. The resulting cross-section libraries would need to be tested and verified via selected benchmark calculations pertaining to reactor neutronics, criticality, depletion, and radiation shielding. The resulting cross-section libraries would be generically applicable for nuclear analyses involving all conventional and advanced reactor technologies.

- Familiarization with pre-existing codes and methods for core neutronics and decay heat in PBMR, GT-MHR, and IRIS. Pre-application review activities, could be used to gain familiarity with pre-existing reactor neutronics codes and, if available, the reactor neutronics codes, decay heat algorithms, analysis assumptions, validation data, and uncertainty treatments that would be used by pre-applicants in their licensing-basis safety analyses. Insights and questions arising from this familiarization process could be incorporated into the prioritization, planning, and execution of the NRC's overall research efforts in this and related technical areas.

- Initial exploratory and scoping studies for core neutronics and decay heat in PBMR, GT-MHR, and IRIS. Using available independent codes (e.g., GRSAC, MCNP/Monte Burns, SCALE/NEWT/SAS2D, WIMS/MONK, Venture 2000, PEBBED), and available applicant codes when needed, the staff could 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 the 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). Insights and questions arising from these exploratory and scoping studies could be incorporated into the prioritization, planning, and execution of the NRC's overall research efforts in this and related technical areas.

- Preparation and testing of spatial kinetics models of PBMR, GT-MHR, and IRIS. As needed, the staff could (1) develop PARCS input models that are compatible with the coupled TRAC-M models, and (2) use appropriate lattice physics and depletion analysis tools with state-of-the-art cross-section libraries (see first item) to prepare the design-specific nodal data tables needed for performing spatial kinetics analyses with the PARCS code coupled with the TRAC-M code.

- Validation and testing for core neutronics in PBMR, GT-MHR, and IRIS. Information on the planned reactor startup and operational tests and measurements related to reactor neutronics could be used to validate core neutronics. Information on existing and planned validation databases (e.g., critical experiments, worth measurements, reactor tests) and sensitivity studies, based on analysis methods developed in recent years at ORNL under RES sponsorship, could help assess their applicability to design-specific reactor neutronics phenomena and help prioritize further data needs. Participation in cooperative programs for acquiring new experimental data and conducting relevant code-to-data and code-to-code benchmarking activities could fill remaining infrastructure gaps.

- Validation for depletion and decay heat in PBMR, GT-MHR, and IRIS. Information on existing and planned validation databases (e.g., spent fuel isotopic assays and decay heat calorimetry) could be used to validate depletion and decay heat. Sensitivity analyses, based on methods developed in recent years at ORNL under RES sponsorship could help assess their applicability to the respective fuels and operating parameters, and help prioritize further data needs. Participation in cooperative programs for acquiring new experimental data, as well as conducting code-to-data and code-to-code benchmarking activities could help fill remaining infrastructure gaps.

- Shielding and material fluence analyses for PBMR and GT-MHR. Any 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.

For ACR-700 nuclear analyses, the following information would be needed:

- Nuclear Data Libraries. Modern libraries of many-group nuclear data will be reviewed to assess the need to incorporate and test the deuterium photo-nuclear data for ACR-700.

- Scoping Analysis for ACR-700 Neutronics, Depletion, and After-Heat. A scoping analyses would be performed using neutron transport methods to gain insights on modeling and validation issues related to reactivity feedback, reactivity control and shutdown, power distributions, depletion of fuel and integral absorbers, reactor kinetics parameters, photoneutron effects, and after-heat power. The insights would be used to assess the need to develop and test ACR-700-specific lattice physics and depletion models, as well as the need to produce few-group nodal diffusion data for use by PARCS for calculating static and transient power distributions in initial, transitional, and equilibrium ACR-700 core loadings.

- Identification of Representative ACR-700 Core Configurations. Code modeling and tools to analyze on-line fueling strategies for ACR-700 core design will be developed, as necessary, to identify a set of initial, transitional, and equilibrium core configurations that (in terms of key parameters such as peak channel and bundle powers) would represent the expected ranges of ACR-700 core operating configurations.

- Physics Tests. An assessment of physics tests would be performed to validate accident scenario analyses.

IV.2.2.2.4 Application of Research Results

Fundamental to reactor safety analysis is the ability to predict the fission and decay heat sources that arise under credible normal and accident conditions. Results from the research activities described above will be applied to enable and support the staff's independent assessment of nuclear analysis issues associated with the respective advanced reactor designs.

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 re-evaluation 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 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 FPs into the environment. The ability to model the progression of severe accidents and estimate releases of FPs into the environment is needed to quantify risk and to address severe accident issues. As described below, the NRC has developed several codes to model severe accidents.

The NRC's severe accident codes are based on a large number of experiments performed in the 1980s following the Three Mile Island 2 accident, and include MELCOR, SCDAP/RELAP5, CONTAIN, VICTORIA, and IFCI. As NRC's consolidated accident code, MELCOR can model most aspects of a severe accident including T/Hs, 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 the development of a fundamental understanding of the phenomena of severe accident and FP transport. The recent NRC focus on severe accidents has included upgrading MELCOR and benchmarking it against the more specialized severe accident codes (e.g., SCDAP/RELAP5 and VICTORIA) and experimental results.

As part of NRC's review of advanced reactors, the 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 advanced 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 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 pressure 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 pressure 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 other advanced designs that differ from current generation reactors, both the types of sequences and the process by which FPs may be released from the fuel may be different than current generation LWRs. In HTGRs, FPs may be released as a result of diffusion during normal operation, by rupture of coated fuel particles as a result of accidents, and by vaporization during high-temperature degradation of the fuel.

The risk from HTGR operation is the risk from releases during normal operation, from accidents involving rupture of coated fuel particles. 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 should focus on FP transport and behavior in the primary system and containment or other structural buildings.

IV.2.2.3.3 Objectives and Associated 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 in terms of the available experimental data and other codes.

Code modifications to incorporate available models/data: The MELCOR code should be modified to incorporate available models/data, and enable its application to HTGRs. Such modifications will allow the FP release from the core and deposition in the reactor coolant system and containment to be modeled.

- Extend FP release models. Release models in the code will need to be expanded to capture current fission release models which are based on Core Source Term Release (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. The current oxidation models for various materials in the code will need 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 a self-sustaining graphite fire. In addition to the graphite fire, smoke and particulate formation should be considered.

- Update materials properties models. Fuel and structural material components in MELCOR will need to include graphite. Graphite/fuel degradation and relocation modeling should be considered, as well as strength and integrity of core supporting structures. The core description considered should be general enough to allow description of both prismatic and PBMR core design.

- Improve numerics. MELCOR's numerics will need to use 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, the code should be evaluated against available experiments. Also, input decks for selected advanced reactor designs should be prepared and code capabilities for selected performance scenarios demonstrated.

Assess the code against available experimental data and other codes: To achieve this objective, a literature review will need to 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 should 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.

Apply code to specific advanced reactor design: The results of the above research will be a version of the MELCOR integrated severe accident code capable of analyzing 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 the capability to support the staff's independent evaluation of the applicants' design from a severe accident perspective.

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 10 years to analyze a number of severe accident issues for operating reactors and advanced reactors, including the AP-600 reactor. Therefore, MELCOR can also 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 was related to in-vessel retention of melt, and this issue will be addressed for the 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 Organization for Economic Cooperation and Development (OECD) material scaling (MASCA) experiment is being performed to evaluate the melt chemical and thermal behavior in 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 the AP-1000, if in-vessel melt retention cannot be assured and in the event of reactor vessel breach, then 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 advanced LWR designs, the effect of high-burnup fuel on the evolution of severe accidents and source terms should be examined. It is envisoned that the MELCOR modeling of FP transport through the reactor system must account for unique features (e.g., helical tubes for IRIS) of the design.

  Research to investigate degradation and FP release characteristics of reactor cores with high-burnup fuel is an important research area for advanced reactors as well as for operating reactors.

For ESBWR, three important phenomena to be addressed in the area of severe accidents and source term are in-vessel melt retention by ex-vessel flooding, fuel-coolant interaction, and molten corium and concrete interaction.

- In-vessel melt retention by ex-vessel flooding. During a severe accident, a pool of water is likely to exist on the drywell floor. If enough water exists to cover the external surface of the reactor vessel lower head, it is important to evaluate whether the ex-vessel flooding can keep molten corium inside the vessel and prevent vessel lower head failure.

- Fuel-Coolant Interaction. During a severe accident involving molten corium relocation into the water-filled lower plenum, an energetic molten corium-water reaction may occur. As a result, the pressure spike and shock wave that may lead to vessel failure must be evaluated.

  Similarly, an energetic ex-vessel, molten corium-water reaction may occur when the lower head of the reactor vessel fails and the molten corium relocates into a pool of water on the drywell floor. The pressure spike and shock wave that may lead to containment failure must be evaluated.

- Molten Corium and Concrete Interaction. If the ex-vessel flooding (or other means of severe accident management) fails to prevent vessel failure, the core debris will relocate onto the drywell floor and will interact with the water there and the concrete floor. It can lead to basemat failure by erosion or overpressurization. Fission products can be released to the environment. The ability and effectiveness of an overlying water pool to thermally stabilize and cool the core debris must be addressed.

For ACR-700, melting and relocation of core debris could be described by existing codes, provided they are modified for the ACR-700 geometry. Code validation of such models will need to be assessed. The code models that deal with ex-vessel phenomena can be utilized with suitable input for ACR-700 without significant modification to the codes themselves.

Necessary core package modifications needed in MELCOR would include a pressure/calandria tube failure model, a fuel failure propagation model, possible modifications to debris/melt progression models, calandria and shield tank failure models, and ACR-700 specific geometry models.

The MELCOR models for fission product release and transport are suitable for analyzing source terms for ACR-700 accidents that develop slowly. Relatively minor modifications would be required to accommodate ACR-700 design differences. Data and models are quite weak, however, for energetic scenarios like fuel-coolant interactions and reactivity excursions. Releases from steam explosions have been studied to a limited extent, but good release models do not exist for steam explosions or rapid reactivity excursions.

Two conditions are of special importance: (1) heat transfer to the moderator (D_2O) for scenarios in which the pressure tubes heat up and sag, and (2) movement of molten core debris for sequences in which cooling by the moderator is not sufficient to stop accident progression.

Databases on fission product release and transport include (1) direct release of fission products and actinides from the fuel as the result of fuel fragmentation and dispersal following a large reactivity transient, and (2) release of fission products from fuel-coolant interaction.

Since 1994, significant modeling and data development efforts for using a version of the MAAP code to analyze severe accidents have occurred. Documentation of these efforts as they apply to ACR-700 should be obtained and reviewed to help inform our research efforts in this area.

Related NRC and international research activities: As discussed in Section IV.2.3, "Fuel Analysis" the Federal Republic of Germany (FRG) has performed irradiation experiments for TRISO-coated particle design with low enriched UO_2. The experiments included accident simulation testing up to 1600°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 (GCRs).

The IAEA has also published many reports of meetings of technical specialists working in the area of HTGR fuels utilizing coated fuel particles (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 FPs in CFP and transport of FP (1992–1996).

Since 1985, 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 including 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 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 materials testing reactors (MTRs).

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 heatup simulations with FP release measurements and post irradiation examinations (PIE).

The 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 (SiC) by palladium.

For the AP-1000, the larger debris mass and power density for the 1000 megawatt (MW) core will impact the effectiveness of the external reactor vessel cooling (ERVC) concept. The ERVC analysis was performed for the 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. 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 Penn State University and 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 the AP-1000 (i.e., 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 the AP-1000 will be performed.

The NRC is assessing its participation in France's PHEBUS-2K program which will investigate core degradation and FP release for high-burnup UO_2 fuel. The NRC is also pursuing an agreement with the Institute de Radioprotection et de Sûrete Nucléire of France to obtain the VERCORS FP release data from high-burnup fuel. Both PHEBUS-2K and VERCORS data will be useful to validate NRC's severe accident code.

IV.2.2.3.4 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, as well as any resulting releases and transport of radioactive FPs within and outside the reactor system boundaries. This information will be critical in supporting resolution of 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 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 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) comprising 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 includes releases from the CFPs with initial 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 differ 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, the licensing basis and the safety analysis for modular HTGRs will largely hinge on the applicant's and the NRC's capability to confirm fuel FP release and 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; design-specific core operating conditions; design-basis accident conditions, and postulated accident conditions beyond design basis. Key factors within the design-specific plant operating conditions that are known to affect 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, core power increases, or significant local reactivity insertion events. Other factors potentially affecting 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), 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 infrastructure assessment.

Additionally, it will be important to qualitatively and quantitatively understand the relationship of 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, and fuel maximum particle power and residence time). It is expected that safety margin aspects will be developed by an applicant. However, a complete assessment of the safety margin would likely require NRC research since HTGR designers and applicants generally do not address conditions that go substantially beyond the licensing basis.

A range of significant fuel design, fuel manufacture, fuel quality, and fuel performance issues exist which will require research initiatives by the respective applicant/vendor. Exploratory and confirmatory NRC research will also be used to support safety findings and conclusions as discussed later in this section.

Additional insights that bear on the extent to which additional 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. The 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 activities should be pursued to fill infrastructure gaps without duplicating previous applicable reference work.

With respect to the ALWRs, confirmatory research should be conducted 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.

The first IRIS core is expected to employ standard (less than 5 percent) 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 percent 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 bi-isotropic (BISO) CFPs involving a layer of buffered isotropic pyrolytic carbon were developed and used in cylindrical fuel compacts at Peach Bottom Unit 1. 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 SiC 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 UK's Atomic Energy Agency 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 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 FRG, in the 1970s, 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 MTRs and the AVR and included aspects such as accident simulation testing. The FRG program was aimed at establishing the concept of a 1600°C limit for pebble fuel elements with TRISO CFPs. The concept was that TRISO CFP failures would not occur until well above 1600°C, while the peak transient fuel temperature for a modular HTGR design would not exceed 1600°C during the most severe postulated accident. The FRG MTR fuel irradiation testing research on CFPs investigated such aspects as particle performance (i.e., failure); 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 Jülich Research Center (KFA). 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 (CRPs) 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 the 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 FPs in CFP and transport of FP (1992–1996). The proceedings from these meetings have been published and are publicly 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 and 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, JAERI has 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 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 including 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 DOE has established an Advanced Gas Reactor Fuel Development and Qualification Program. The early efforts of program plan will focus on the development and qualification of fuel that would be used in the GT-MHR prismatic core. The fuel manufacture will be based on the German fuel kernel and fuel particle coating fabrication processes. The major elements of the draft program plan involve fuel fabrication technology, fuel irradiation testing, irradiated fuel accident condition testing and PIE, and fuel performance modeling, fission product transport and source term. The fuel fabrication element involves the laboratory scale and later the production scale manufacture of coated-particle fuel that is intended to meet fuel performance requirements. It includes process development for kernels, coatings, and compacting; quality control (QC) methods development; and process documentation needed for technology transfer. This effort will produce fuel and material samples for characterization, irradiation, and accident testing. The fuel irradiation testing activities will provide data on irradiated fuel performance for process development and fuel behavior during normal operating conditions. This element is also intended to support development and validation of fuel performance and fission product transport models and codes as well as irradiated fuel for accident condition testing and PIE. The accident condition testing and PIE activities are intended to provide data on the performance of kernels, coatings and compacts during accident conditions to demonstrate acceptable accident performance of the fuel and to support the development and validation of models and codes. The fuel performance modeling activity is intended to addresses the structural, thermal, and chemical processes that can lead to coated-particle failures. Model development will be aimed at developing and validating fuel performance models and codes to support fuel fabrication process development and fuel safety performance analysis. The fission product transport and source activity is aimed at addressing the transport of fission products produced within the coated particles and to provide a technical basis for source terms under normal and accident conditions.

The E.C. is currently sponsoring an approximately $16M, 4-year research program on HTGRs. The E.C. 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 TRISO 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 heatup simulations with FP release measurements and PIE.

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 under 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 knowledge 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 megawatt days per ton (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.

The MIT has established a high-temperature pebble bed reactor research project for student research. One area of student research is improved CFP performance modeling, including migration of FPs through coatings and the 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 also 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 database 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. In addition, these tests will serve 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 facilities will be available for irradiation testing until after CY 2005.

IV.2.3.2 Purpose

The purpose of the regulatory research infrastructure assessment 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 must 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, 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 infrastructure assessment 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. Analytic 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 also provide the staff with an independent capability to calculate TRISO particle fuel source term for these same conditions.

IV.2.3.3 Objectives and Associated 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 safety issues in the area of performance and qualification of HTGR and ALWR fuels. The specific objectives are as follows:

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 beyond the design basis for parameters 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 insight needed to judge the acceptability of an applicant's fuel irradiation test program (e.g., test methods, quality assurance program, data analysis methods).

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

Fuel analytical model and methods development: The purpose of this development effort would be to:

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

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

Fuel fabrication technology expertise: The expertise would 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 targeted fuel quality and fuel performance over the life of the plant's fuel supply.

HTGR fuel irradiation testing:

- <u>Issues</u>. Virtually all of the past and ongoing worldwide irradiation testing research of HTGR fuel designs with TRISO CFPs include 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 heatup test following irradiation) to qualify the fuel involved accelerated irradiations in MTRs. A well-established and thorough understanding of the mechanics and properties (e.g., creep) of CFP behavior, failure, and FP release does not exist to allow one 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 real time MTR fuel irradiations or after fuel irradiations in a power reactor, would be needed to resolve this issue.

  Virtually all of the accident simulation tests for TRISO CFPs involved so called "ramp and hold" temperature increases. These typically consist of increasing fuel temperature at about 50°C/hr up to a set temperature (e.g., 1600°C, 1700°C or 1800°C) and then holding the fuel at the set temperature for several hundred hours while FP release measurements are taken. The results of ramp-and-hold tests up to 1600°C, for qualified fuel, show that no additional CFP failures occur. However, in the FRG, there was at least one test in which the temperature was controlled to closely simulate the predicted accident heatup curve to 1600°C for a design-basis reactor coolant pressure boundary failure. For this test, CFP failures were observed to occur. Additional post-irradiation accident simulation tests that closely simulate the predicted temperature curve for a design-basis reactor coolant pressure boundary failure would be needed 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 large pipe 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. Experiments on unirradiated HTGR fuel in air and water at HTGR accident temperatures have been conducted. 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 CFP 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 (less than 1 second). Some limited testing was conducted in Japan for a postulated control rod ejection accident in support of the HTTR licensing, this scenario 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 and 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 may to be useful.

Only limited worldwide testing has been conducted on previously qualified FRG or U.S. HTGR CFP fuel for conditions that go 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°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.

- Preliminary 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 include 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 satisfactory 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 United States 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 strong 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: Testing of unirradiated German archive pebble fuel fabricated for the AVR and testing of HTGR production fuel for demonstration of prototype HTGR plants that may be built in the United States. 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, while Table 3 summarizes a proposed testing plan for GT-MHR production fuel compacts. Testing of the German archive fuel would provide information on the acceptability of traditional testing methods, insights into the 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 cooperative agreements described below; any proposed testing would not duplicate but will capitalize on testing performed by DOE.

This plan assumes NRC would closely follow and obtain beneficial information from the DOE Advanced Gas Reactor Fuel Development and Qualification. The NRC emphasis for fuel testing will be on understanding the safety margins, by exploring conditions that are beyond the fuel design-basis conditions associated with normal operations and postulated accidents. It is expected that the DOE program will also provide test data that can be used for developing and validating NRC fuel performance analysis models and data that can be used to confirm an applicant's fuel performance analysis. Further, following the DOE program will provide the staff knowledge of fuel testing 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 procedure as well as the accident heatup simulations with FP release measurements and PIEs. The NRC will also support the retrieval of data from past HTGR experiments with the aim of constructing a searchable fuel database.

HTGR fuel analytical model and methods development:

- Issues. The body of irradiation and accident simulation (heating) tests has enabled the development of analytic tools for evaluating HTGR fuel performance during reactor operating conditions and postulated accident conditions. These tools have endeavored to model the various particle failure mechanisms that have been identified, including internal overpressure 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 temperatures; chemical interaction of FPs with the SiC layer leading to SiC degradation and failure, and mechanical overstress of the SiC layer due to external loading on the particle layers. Models have been developed for each of these potential failure modes. These models have been used by fuel designers to help quantify margins and by safety analysts in calculating mechanistic source terms.

- Preliminary plans. As a first step, a review of all ongoing research will be performed, 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 with the European Union to develop and validate analytic 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 analytic 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.

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 of U.S. fuel is three orders of magnitude higher than the gas release rate during irradiation of the German fuel. A recent 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 United States 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 variations in the layer characteristics such as micro structure, layer bonding, and layer anisotropy. The differences in particle characteristics resulted in significant variations 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 an HTGR's license 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 technical and regulatory advantages and disadvantages, however. An additional policy issue is whether a plant can be licensed before fuel testing is complete, which relates 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 infrastructure assessment 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.

- Preliminary plans. A major research element of the E.C. HTR-F is to re-establish knowledge 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. 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 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.

  Finally, a major element of the DOE Advanced Gas Reactor Fuel Development and Qualification Program is fuel fabrication technology. The fuel fabrication element involves the laboratory scale and later the production scale manufacture of coated-particle fuel that is intended to meet fuel performance requirements. It includes process development for kernels, coatings, and compacting; QC methods development; and process documentation needed for technology transfer. The development activities include, for example, fuel process studies to understand how coating conditions are related to coating layer properties and how layer properties effect fuel particle performance during irradiation. It is expected that the DOE fuel fabrication technology research activities will provide NRC with significant insights, information and knowledge in the area of TRISO fuel manufacture. As a publically funded program, information developed by the DOE would be available to the NRC at no cost.

HTGR fuel condition monitoring effectiveness assessment: Because of the importance of fuel integrity to the HTGR safety case, defense-in-depth against loss of fuel integrity is critical. Fuel manufacturing specifications and quality controls, fuel irradiation testing, fuel accident condition testing, and fuel performance code analysis all play a role in assuring defense against unexpected elevated fuel particle failure rates during either plant operation or licensing-basis events. An additional important defense against loss of fuel integrity is also provided by HTGR core condition monitoring systems. These systems are intended to detect elevated fuel particle failure rates during plant operations. Core condition monitoring systems are also to be relied upon to monitor the condition and capability of the fuel in the core and to maintain the expected level of integrity during licensing-basis events.

HTGR core condition monitoring systems typically detect manufacturing-related particle defects or irradiation-related particle failures by monitoring noble gas activity in the circulating helium coolant. Fission gas release measurements from fuel irradiation testing are used to correlate the magnitude of fission gas release due to all causes of fuel particle failure, diffusion, and release mechanisms. These systems must be effective so that remedial actions can be taken when core-wide failure fractions show signs of increasing above the expected levels.

An important research issue is whether HTGR core-condition monitoring systems have the capability to detect significant "latent" fuel particle failure conditions (i.e., "weak fuel" or "weakened fuel") which have not yet been manifested as elevated fuel particle operational failure rates. Latent failure conditions might occur due to systematic undetected errors in either the manufactured fuel quality (e.g., incorrect particle layer coating rate) or operating conditions that are significantly outside the fuel design envelope (e.g., elevated fuel operating temperature). Such latent failures may or may not be detected and actions may or may not be taken prior to an event resulting in a core-wide failure fraction above the predicted level.

The associated research will investigate the capability to detect weak fuel caused by out-of-specification manufacture and fuel weakened by operating conditions well above the design parameters. The research will assess whether such fuel would generally be detectable during operations as a result of elevated (higher than expected) coolant activity caused by elevated (higher than expected) particle failure rates. The research will also seek to assess whether fuel that was weak or weakened due to specific manufacturing or operating conditions might not result in elevated particles (and coolant activity) during operations but would be sufficiently degraded to cause higher than expected fuel failure rates during heat-up accidents. The research will involve a combination of irradiation testing, accident simulation testing, and sensitivity studies using fuel performance analytic tools.

Weakened fuel performance from operations will be assessed by fuel irradiation tests conducted at significantly higher than design operating conditions followed by accident condition heat-up testing. Fission gas release measurement data for both tests will be the principal basis to assess the potential for inducing and detecting weakened fuel which would fail during an accident.

Sensitivity studies with analytic codes will be used to assess whether fuel that was weak (in various ways) due to manufacturing errors would result in detectable increases in fuel failure rates or whether there are conditions of weakness due to manufacture that would not be evident as increased failures until the accident condition. Available historical irradiation test data, operational data, and accident simulation test data will also be studied for any evidence on the capability of core-condition monitoring systems to detect weak or weakened fuel.

For ESBWR fuel analytical model and methods development: The major fuel behavior phenomena include (1) the initial stored energy (temperature profile) in the fuel pellets for LOCAs, (2) fuel deformation during LOCAs, and (3) pellet-cladding interactions during power ramps. Since the ESBWR fuels are similar to the fuels used in the operating BWRs, the existing fuel behavior codes such as FRAPCON (for steady-state calculations) and FRAPTRAN (for single-rod transient, power-ramp calculations) are sufficient for modeling the ESBWR fuel behavior, pending further investigation. If the burnable poison loadings in the ESBWR fuels are different from those in the current BWR fuels, additional FRAPCON assessment against data may be needed. Furthermore, if the stability analyses (discussed in Section 3.2, Nuclear Analysis) indicate flow and power oscillations, FRAPTRAN calculations will be required to analyze the fuel behavior during the transient.

For ACR-700 fuel analytical model and methods development: The major fuel behavior phenomena to be analyzed include (1) the stored energy initial conditions (temperature profile) for loss-of-coolant accidents, (2) deformation of fuel and pressure tubes during loss of flow events, and (3) pellet-cladding interactions during power ramps. Three different computer codes are needed for these types of analyses.

- <u>Steady-state fuel rod behavior</u>. Detailed thermal and mechanical analysis of single fuel rods under steady-state conditions is sufficient for many applications such as stored energy for loss-of-coolant accidents and the release of fission products to the fuel-cladding gap. The FRAPCON code was developed by NRC for this purpose, and it is currently being upgraded for LWR applications. Relatively straightforward modifications to that code should be sufficient. The CANLUB lubricant between the pellets and cladding would have to be accommodated. A heat transfer package for horizontal flow would have to be developed, and the cladding oxidation model would have to be modified for the different oxygen potential of ACR coolant water.

- <u>Transient fuel rod behavior</u>. For LWRs, single-rod transient analysis can be performed with the FRAPTRAN code to analyze cladding stresses resulting from power ramps and to analyze deformation and rupture during loss-of-coolant accidents. With modification, FRAPTRAN will be able to analyze the power ramp conditions in ACR-700, but the deformation and rupture during loss-of-coolant accidents will require a multi-rod capability (see below).

- <u>Transient behavior of multi-rod bundles</u>. During loss-of-flow transients, fuel heatup results in heatup and sagging of the horizontal pressure tubes which can come into contact with calandria tubes. Contact with the calandria tubes is thought to provide sufficient heat transfer from the fuel to the moderator in the calandria tank to terminate the transient.

  The critical phenomenon in this situation is pressure tube heatup and deformation. This behavior is influenced by many rods in the fuel bundle. Flow in this horizontal geometry will be stratified at times, with rods at the bottom in water and rods at the top in steam. Heat transfer regimes for different rods will vary and axisymmetric conditions cannot be assumed (as in LWRs). A multirod model will therefore be required.

  The severe accident code, MELCOR, has to be modified to model a multi-rod configuration in a horizontal geometry for ACR analysis. Work that has been performed with the SCDAP/RELAP5 code to model horizontal fuel elements for the N-reactor should be reviewed in terms of the kind of modifications mentioned above for FRAPCON and FRAPTRAN to provide a version of MELCOR for analysis of ACR-700.

For IRIS fuel analytical model and methods development:

- <u>Issues</u>. For the IRIS fuel design, the research will provide data and code analyses, as appropriate, 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 may need to be considered. (It should be noted, however, that industry has the primary responsibility for demonstrating that performance targets for each of the items listed below will be met.)

– 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 UO_2 fuel pellets.

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

Related NRC and international research: Argonne National Laboratory could be called upon 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 could be used to extend code assessment to burnups greater than 75 GWd/t and to incorporate new cladding properties as needed.

Additional work at Halden could be used to obtain information 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 Application of Research Results

The intended safety characteristic of the TRISO 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 operating and accident conditions. Given the significance of the fuel barrier for the HTGR designs, the fuels research program will be used to provide insights on the 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 the 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 heatup profile to simulate longer term fuel heatup (e.g., loss of helium cooling due to loss of forced circulation following a reactivity insertion pebble compaction).

Table 3. GT-MHR Production Fuel Irradiation Tests

#	Irradiation Purpose	Burnup Increment (GWd/t)					Safety Test Δ
		0 to 25	25 to 50	50 to 75	75 to 100	100 to 125	
1	Archive Compact						N/A
2	Archive Compact						N/A
3	Design Max Fuel Temp	------Accel------	-------Accel-----	------Accel-------	------Accel------	**Δ**	1800^{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 heatup profile to simulate longer term fuel heatup (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 Background

A key research area important to safety is the behavior of metallic and graphite components performing the 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 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 can 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 research area.

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 (aside from unique designs such as the ACR-700 discussed separately in Section IV.2.4.3), 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 United States 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 Pressure Vessel Research Council [PVRC] reports) for taking into account the effects of the operating environment in the fatigue design of components.

Although the American Society of Mechanical Engineers (ASME), through its ongoing 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, during design and review of ALWRs, the effects of the environment must be appropriately accounted for in the fatigue design and evaluation of components. Work with ASME should be continued 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 an 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 circumstances suggest a need for evaluating the 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 Purpose

The NRC staff needs to develop independent research and expertise in the high-temperature materials area for HTGRs to evaluate and establish a regulatory technical basis regarding the safety of advanced reactors. These advanced reactor designs are significantly different from LWRs, in terms of the materials used, such as high-temperature metals and graphite; higher coolant temperatures; and coolants that do not change phase. These advanced designs also experience different degradation mechanisms, such as creep, and behavior of metallic and graphite components in this environment will also differ from that of LWRs.

In HTGRs, graphite acts as a moderator and reflector, as well as a major structural component, providing channels for the fuel and coolant gas, and control and shutdown rods, and acting as a thermal and neutron shield. Additionally, graphite components are employed as supports. Graphite also acts as a heat sink during reactor trip and transients. During reactor operation, many 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 a 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 1980s 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. In order to develop NRC capability, the staff needs to review and evaluate this research and that which has progressed since the 1980s and 1990s, particularly with respect to the temperatures, coolant environment, and materials used, to determine applicability to current HTGR designs.

The NRC staff needs to develop independent research capability in the materials area beyond the licensing basis to understand safety margins, 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. Thus, 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 databases for calculating fatigue, creep, and creep-fatigue interaction lifetimes of components in high-temperature applications; (3) the effects of impurities, including oxygen, on 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 and methodology for prediction of irradiated graphite properties from the as-received non-irradiated 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, and bridging. This issue is discussed in the section on Nuclear Analysis.

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 1980s for application to the liquid metal breeder reactor. Of particular note is the work conducted by the PVRC in its 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 the presence of 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 were 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 operating lifetime of 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 associated with 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 time frames less than 40 years. However, the sensitization rate is exponential with temperature, and at the higher operating temperatures of HTGRs, there is a 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 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 a much smaller diameter and therefore possesses a 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 a long time 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 and 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. 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 in the coolant is required. Further, 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 created by the 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 to 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.

Description of issues, ISI and monitoring: A number of potential issues related to the inspection of some HTGR and ALWR reactor components exist. Because some of these reactors are designed to operate for long periods of time between scheduled short-duration shutdowns for maintenance or refueling, ISI intervals may be long and the amount of inspection conducted limited. Therefore, the effectiveness of various ISI programs as a function of the frequency of inspections and the number and types of components inspected needs to be evaluated. Additionally, many internal components are not easily accessible for ISI, and the impact of not inspecting these components needs to be assessed. An alternative to conducting periodic ISIs during reactor shutdowns is to conduct continuous online, 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.

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; and crystallite size and uniformity determine the as-received 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, the physical and mechanical properties of the graphite. Important properties that change with irradiation are density, thermal conductivity, strength, and dimensions. These changes have safety implications since they could degrade structural integrity, core geometry, and cooling properties. Some of these changes are not linear with irradiation dose. Graphite strength 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 can 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, property change data due to irradiation is needed in addition to the as-received properties. Development of irradiation data on graphite is difficult, expensive, and time consuming. Therefore, reactor designers/ vendors have proposed to use radiation data from studies conducted on older graphites and attempt to use graphites produced in a similar manner. However, the as-received 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 as-received graphite raw materials characteristics, composition, processing, and properties.

Graphite corrosion and oxidation can occur in HTGRs from oxidizing impurities in (or added to) 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 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 British 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, it 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 (possibly resulting in a thicker component), the excessive cracks in the component may grow during service and cause failure. Specific impurities 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, standards need to be developed to establish the acceptable physical, thermal, and mechanical properties, composition, and manufacturing variables for nuclear grade graphite.

IV.2.4.3 Objectives and Associated 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 needs to 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 need to 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 would be updated to incorporate correlations developed from more recent research. Research on graphite will need to be conducted to (1) evaluate performance under high levels of irradiation; (2) develop correlations for irradiated properties from as-received properties; (3) develop data on oxidation kinetics; (4) evaluate variation in properties through the thickness of large blocks; (5) develop standards for nuclear grade graphite, and (6) develop an understanding of the mechanics of pebble flow. A description of this research for metallic components, ISI, and graphite components follows:

Metallic components: Carburization, decarburization, and oxidation of HTGR high-temperature metals need to be studied as a function of time and temperature in helium gas with impurities, including oxygen. Different levels and ratios of impurities would be studied. Metallographic studies and mechanical testing would be conducted on the exposed samples to determine the degree of deterioration and loss of strength. The objective of these studies and tests 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 needs to 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 need to 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 these phenomena, then the database 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 needs to 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 needs to 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 would 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 may occur.

Thermal aging and sensitization research needs to be conducted on high-temperature alloys used in HTGRs using samples in the as-received and the welded conditions. Samples would 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 temperatures expected. Mechanical property testing would 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 would 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. An opportunity exists to evaluate and validate these potential degradations by conducting research on components removed from operating reactors. An international research program needs to be conducted on components removed from the AVR, including microstructural studies and mechanical tests. Microstructural studies would 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 would establish if stress corrosion cracking, crevice corrosion, general corrosion, and oxidation have occurred. Mechanical tests on materials removed from the AVR would be conducted to determine if any degradation in materials properties has occurred. Fatigue and creep tests would 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 E.C. and Japan, the NRC staff identified several areas that address NRC research objectives. Work of interest in the E.C. includes (1) review of RPV materials, focusing on previous HTRs, in order to set up a materials property database on design properties, (2) compilation of existing data on materials for reactor internals and selection of the most promising alloys for further development and testing, and (3) 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 (1) research on a pressure vessel steel containing 9 percent chromium (irradiation testing, fatigue, creep-fatigue, tensile, fracture toughness); both heavy-section base metal and weldments are included in the studies; (2) 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 (3) 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 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, discussed at the workshop on HTGR safety and research issues (October 2001, US NRC, Rockville). Studies should be conducted of the effects of vibrations and service conditions to determine the reliability of this insulation because 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 E.C. 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 infrastructure assessment have been shared with the international community, in particular with Japan and the E.C. The NRC staff has met with technical staff and officials of the E.C. and JAERI to discuss potential cooperation. The E.C. has agreed with the importance and need for the research outlined in the NRC infrastructure assessment and welcomes NRC to participate in its high-temperature materials research program known as HTR-M. 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 infrastructure assessment will be addressed in the EC's current program and its future program initiating in 2003. Some of the key work possibly not fully addressed in the E.C. programs is in the areas of (1) effects of the helium environment with impurities on degradation of materials, and (2) aging and sensitization. Exchange of NRC research results in these areas could be used for cooperation with the E.C. HTR-M programs.

ISI and monitoring: In the nondestructive examination area, research needs to be conducted to evaluate the impact of different ISI plans on structural integrity and risk. The key variables in the study would 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, would 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 online monitoring of structural integrity.

Because some components are inaccessible and because the interval between ISIs may be too long, research needs to be conducted to evaluate continuous monitoring of reactor components for crack initiation and crack growth and for leak detection. Acoustic emission techniques would 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 would 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 when acoustic emission techniques were developed, validated, and codified for application to LWRs. The research, methods, and techniques for HTGRs and ALWRs would take advantage of the knowledge gained in earlier work. Similar acoustic emission techniques need to 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 would 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 would 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 demonstrate the advantages of continuous online monitoring of reactor structural integrity and leakage. The results will also provide technical databases for incorporating the techniques into codes and standards.

Areas of international cooperation and exchange would involve work planned by the E.C. 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 online continuous monitoring techniques for structural integrity and leak detection using HTGR test reactors.

Graphite: Research needs to be conducted to evaluate graphite for HTGR application. This would involve studies of the performance and degradation of graphite under high levels of irradiation and temperature. A review would be conducted of available high-dose irradiation data for nuclear grade graphite, including unpublished data from ORNL taken under the DOE nuclear power reactor-Modular High-Temperature Gas-Cooled Reactor program. High-dose irradiation data on “old graphites” needs to be evaluated to determine their applicability to “new graphites." The data would be utilized to determine the behavior of current graphites planned for HTGRs under operating conditions. In general, a lack of data exists for the high-dose, high-temperature regime of the HTGR operating environment; additional research needs to be conducted on current graphites planned for HTGRs to determine high-dose material behavior, properties, and degradation. Experiments need to be conducted at different temperatures at high-dose irradiation in a high-flux test reactor. Microstructural evaluations, 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 also needs to be performed to determine the irradiated graphite properties from as-received graphite properties. As-received graphite properties are determined by the raw materials and manufacturing process. Important parameters would be identified, such as coke, pitch characteristics, and graphitization temperature. A number of different graphites would be selected with carefully varied parameters. Studies would be conducted to establish the as-received properties of the graphites. Selected properties to be measured are x-ray crystallinity; density; open and closed porosity; pore size distribution; grain size and size distribution; grain orientation and orientation distribution; thermal expansion; thermal contraction; thermal conductivity; absorption cross-section; sonic Young’s modulus; stress-strain behavior; strength and strength distribution (Weibull modulus), and fracture toughness. In addition, chemistry of the graphites, including impurities, would be established. Due to the anisotropy of manufactured graphite, the materials properties would be determined for two orthogonal directions since graphite exhibits transverse isotropy. The graphites would then be irradiated at systematically varied irradiation doses and temperatures significant to HTGRs. Following irradiation, the materials properties would be re-evaluated to determine the effect of irradiation and to establish correlations between the initial as-received properties and the post-irradiation properties that could apply to any particular graphite that may be used in HTGRs.

Investigations need to be undertaken to understand oxidation effects on the physical, thermal, and mechanical 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 need to be conducted to determine weight loss and loss of mechanical integrity due to oxidation of graphite samples. The heat generated from oxidation of graphite dust and the potential detrimental effect on surrounding components due to this elevated temperature needs to be studied. Research needs to be performed to determine if oxidation occurs along binder paths through the bulk graphite which could lead to diminished fracture, fatigue, and creep resistance of components.

Research on large blocks of graphite needs to be conducted to characterize the through-thickness variability of key properties in full size blocks and to establish the variability between batches of graphite. Large graphite blocks to be used for reflector material would be sectioned, tested, and evaluated to determine if properties measured on thin graphite components can be extrapolated to large blocks. Graphite materials properties are typically anisotropic and vary with the forming method and size of the final fabricated component. The sectioned large-block specimens would be tested to measure important parameters such as strength, fracture toughness, density, thermal conductivity, coefficient of thermal expansion, level of chemical impurities, isotropy, and absorption cross-section. Based on the results obtained, an assessment will then be conducted to determine if the large-block bulk properties would vary under high-temperature and high-dose irradiation in a manner similar to thin sleeve graphite material.

Staff efforts will be directed toward development of consensus standards for nuclear-grade graphite. Design and fabrication codes are also needed. The NRC staff will work with the international community, industry organizations, and professional societies to develop a nuclear-grade graphite material specification consensus standard. The standard would specify requirements 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 graphite components to address processes such as strength, fracture, fatigue, creep, irradiation damage, dimensional stability, oxidation, and any other appropriate design and fabrication considerations for HTGR service.

A review and evaluation should be undertaken of experimental data, analyses, and appropriate models for predicting pebble flow through and across a PBMR reactor core. Evaluations would 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 need to 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 E.C. research effort is currently reviewing the state-of-the-art on graphite properties in order to set up a suitable database. The E.C. 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 E.C. began extensive characterization and irradiation testing of five different graphites (two from Union Carbide, two from Superior Great Lakes, and one 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, the E.C. plans to address a considerable amount of work related to high-temperature metallic components. However, a key area possibly not fully addressed in the E.C. program is the correlation of as-received graphite properties and manufacturing parameters to irradiated graphite properties. Exchange of NRC research results in this area could be used for cooperation with the E.C. HTR-M programs.

The UK is conducting ongoing research on graphite properties and has had experience with operating GCRs that may be useful for NRC cooperation. As part of international cooperation with the UK, the NRC has assigned a staff member from RES to the NII for three months to develop expertise on graphite behavior under high-temperature and high-irradiation conditions and to develop knowledge of the inspection and monitoring programs of graphite in HTGRs. The NRC staff member will have discussions with experts on the reasons for a lack of available correlations of as-received graphite properties with irradiated graphite properties. The NRC staff work will 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 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 plans for developing correlations between irradiated graphite properties and initial as-received properties.

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 its program. In this effort, the staff would obtain information on the scope and objectives of NII's graphite research at the University of Manchester. The staff can obtain details from the 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 this research and to explore different methods for NRC participation as appropriate.

The staff member will develop recommendations for requirements of a nuclear grade graphite material specifications standard, and for a graphite component design code. This effort would be performed in collaboration with NII and other experts to outline one or more potential standards for the manufacture, composition, and required properties for nuclear grade graphite. The NRC staff member will also obtain, review, and discuss with NII and other experts different codes and procedures available for structural, fatigue, and creep analyses for the design of high-temperature graphite components. The staff will evaluate these codes and develop recommendations on the need to update them 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 from the DRAGON test reactor experiments performed on graphite and fuels in the UK and to evaluate this information for applicability to currently proposed HTGRs.

ACR-700: This section describes the materials engineering research needs anticipated for the ACR-700. The composition and operating environment of risk-significant metallic components in the ACR-700 are significantly different from those in BWRs and PWRs that research may be needed to determine the long-term integrity of these components. The ACR-700 design features include horizontal Zircaloy pressure tubes rather than a large reactor pressure vessel, heavy water moderator, and on-line refueling. Differences from previous reactor designs that would affect performance of materials are the more compact reactor design, the use of enriched fuel rather than natural uranium, higher fuel burnup, light-water coolant, higher coolant pressure and temperature, and higher steam pressure and temperature.

Potential research needs for the metallic components in the ACR-700 design are (1) the effect of the environment on component fatigue and creep life; (2) irradiation damage and embrittlement; (3) the performance of dissimilar metal welds and material compatibility; (4) accessibility of components for inservice inspection, and (5) component material behavior under severe accident conditions.

- Environmental Effects on Component Degradation. Materials research needs to consider the effect of the operating environment on expected component creep life, fatigue, and degradation due to irradiation damage and delayed hydride cracking. For example, impurities in pure water are known to cause reduction in fatigue life of components in LWRs. In pure water with oxygen levels in the parts per billion (ppb) range (similar to oxygen levels in PWRs), the fatigue life of austenitic stainless steel components is reduced relative to that in air, while there is no reduction in fatigue life to austenitic steel for oxygen levels in the parts per million (ppm) range (similar to BWRs). On the other hand, for oxygen levels in the ppm range, while the fatigue life is not reduced compared to that in air, the susceptibility of these materials to stress corrosion cracking (SCC) is increased. For ferritic steels a reduction in fatigue life occurs when oxygen levels are in the ppm range, but no reduction in fatigue life occurs at the ppb level. Ferritics are more resistant to SCC, but less resistant to erosion corrosion. Therefore, it is important to know what the materials and environment (chemistry and stress levels) will be in the ACR-700 so that susceptible combinations of materials and environments will be avoided and the components can be designed for the expected 60-year plant life.

  Applicable codes and standards are currently undergoing review to incorporate research results on the effects of environment on the reduction in fatigue life of ferritic and austenitic steel components for LWR applications. 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 reports) for taking into account the effects of the operating environment in the fatigue design of components. Although the American Society of Mechanical Engineers (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, during design and review of ALWRs, the effects of the environment should be appropriately accounted for in the fatigue design and evaluation of components.

  The role of pressure tubes in the ACR-700 design is in many respects similar to that of primary-coolant-boundary components in a BWR or PWR. Therefore, long-term structural integrity of the pressure tubes is very important in assuring safe operation of the reactor over its design life. The design of the ACR-700 calls for replacing pressure tubes (“retubing”) either after 30 years of operation or earlier if determined necessary on an individual tube basis. An adequate technical basis needs to be developed for determining the adequacy of the pressure tubes to perform reliably for 30 years when considering the operating environment, including factors such as stress, temperature, chemistry, and radiation limits and mechanisms such as fatigue, SCC, delayed hydride cracking (DHC), and brittle failure to avoid potential pressure tube rupture.

With increasing neutron fluence and service time, fracture toughness decreases significantly (to less than 25 percent of its original value) for the Zr-2.5Nb pressure tubes. Similar to ferritic pressure vessel steels, an irradiated Zr-2.5Nb pressure tube exhibits an increasing temperature ductile-brittle-transition (DBT) phenomenon. The DBT temperature (DBTT) is strongly influenced by hydride structure and distribution and reaches ≈260–280°C after about 5–10 years of operation in current plants. The exact roles of hydrides and irradiation-induced damages are not well understood. ACR-700 plants will have higher channel powers and a harder neutron energy spectrum. This may increase the irradiation damage to the pressure tube material and thus the DBTT could reach 260–280°C in a shorter time.

Zr-2.5Nb pressure tubes are susceptible to failure by subcritical crack growth phenomenon which is commonly attributed to delayed hydride cracking mechanism. After 10–15 years of operation, the subcritical crack growth rate of Zr-2.5Nb pressure tubes reaches a value a few orders of magnitude faster than the stress-corrosion crack growth rate of austenitic stainless steels in a BWR, (i.e., 10^{-7} to 10^{-6} m/s vs. 10^{-10} to 10^{-9} m/s). Since this is an important degradation mechanism for the Zr-2.5Nb alloy and crack growth is even higher than SCC in stainless steel, DHC needs to be better understood under the more severe conditions of temperature, fluence, and coolant in the ACR-700.

The high-pressure-boundary components of an ACR-700 are exposed to H_2O coolant at 278–325°C. The exact specifications of the structural components are not well known. Depending on the chemistry of the water coolant (e.g., dissolved oxygen level, electrochemical potential), components fabricated from austenitic stainless steels or Ni-base alloys are susceptible to SCC in this temperature range. Therefore, an understanding of the effects of the coolant water chemistry, material type, fabrication procedure, and stress state is needed to determine the susceptibility of components to failure by SCC.

Research also needs to be conducted to characterize pressure tube sagging and anisotropic irradiation-induced diffusion. The diameter of a pressure tube decreases slightly and the length increases significantly over years of service. The horizontal position can cause a pressure tube to sag significantly during service. This sometimes leads to contact with the surrounding calandria tube. The sagging behavior is strongly influenced by the position and properties of supporting garter springs. If the hotter pressure tube sags and touches the colder calandria tube, a hydride blister forms at the contact spot and grows inward from the outer diameter surface of the pressure tube. This process can eventually lead to crack initiation and a through-wall penetration, as has been observed in the past in both Zircaloy-2 and Zr-2.5Nb pressure tubes. Therefore, understanding tube sagging behavior at the higher temperature conditions of the ACR-700 is considered a high priority.

The pressure tubes will be slightly thicker, which is beneficial to reducing sagging. This is one cause of delayed hydride cracking that could initiate a dynamic crack propagation event. The operating pressure and temperatures of the ACR-700 are expected to be higher. The axial-crack-driving force is very sensitive to the saturation pressure of the coolant. The higher temperature in the ACR-700 will increase the saturation pressure. This has not been examined in detail, but the thicker ACR-700 pressure tube will result in a hoop stress that helps to compensate for the higher saturation pressure. Evaluation of the crack-driving force is needed since it is not known at this time if it will be higher or lower in the ACR-700.

The ACR-700 design incorporates a large number of bent pipes and joints. A technical basis for the evaluation of the potential for long-term erosion-induced failure of such locations may also be needed. The cold-bent A106B pipe has been susceptible to stress-corrosion cracking in recent years. If this pipe system is changed to stainless steel in the future, then consideration will need to be given to the fact that the light-water coolant could initiate stress-corrosion cracking of the stainless steel.

- Material Compatibility. The design arrangement of calandria tube, pressure tube, end fittings, and various internal components brings dissimilar metals into contact using fittings and joints. Research activities would focus on the compatibility of materials in these components in the reactor environment. Stress corrosion, fatigue, and creep crack initiation and growth rates can be enhanced at dissimilar metal welded or rolled joints. The potential for enhanced or increased hydride blister formation, delayed hydride cracking, ductile-brittle transition temperature, and irradiation damage may also need to be investigated.

  The two ends of a Zr-2.5Nb pressure tube in the ACR-700 are rolled-joined to Type 403 martensitic steel end fittings. During operation, the rolled joints are colder in temperature and higher in stress (due to rolling-induced residual stress). Both factors are conducive to more pronounced hydrogen uptake and hydriding in the Zr-2.5Nb portion of the rolled joint. The potential for Type 403 martensitic steel end fittings to become susceptible to irradiation-induced degradation of fracture toughness needs to be considered.

- Inservice Inspection. The impact of on-line refueling of the ACR-700 on ISI effectiveness will need to be considered. Long time periods between reactor shutdowns and limited accessibility of reactor components in the compact design may render ISI ineffective as a method to detect unanticipated component cracking and degradation. Continuous on-line monitoring may be necessary if limited ISI is found to be inadequate.

  In the ACR-700 design, the core is more compact, so that the multitude of feeder pipes coming off the fuel channels will be much closer to each other, thus making inspection more difficult. The feeder pipes are about 3” in diameter, so that they do not fall under many of the ASME Code inspection rules that apply for 4” diameter or larger pipes. The new ASME Code Case being proposed for external corrosion of small diameter pipes will need to be considered for application to the ACR-700 feeder pipes.

- Material Performance During Severe Accidents. Pressure tube swelling during severe accidents will need to be understood. During severe accidents, temperatures can climb into the rapid creep regime (secondary and tertiary creep) and pressure tube behavior can become unstable. The creep failure of smooth and defect-free pressure tubes can be modeled using Larsen-Miller Parameter type data correlations, however pressure tubes with surface scratches, cracks, pits, inclusions, or other defects fail more readily. Correlations and models have been developed for predicting steam generator tube failures with pre-existing flaws. Research to develop similar correlations and models for the Zircaloy ACR-700 pressure tubes, to better understand safety margins and modeling uncertainties under severe accident conditions, will need to be considered.

During a severe accident, the accelerated rate of tube sagging and the resultant sagging effect on tube integrity and heat distribution need to be examined. The fuel channel is designed to achieve a 30 year operating life. The material may become aged and embrittled due to hydride blisters, elevated DBTT, or other degradation mechanisms over this time period in the ACR-700 reactor environment. The resulting response of the pressure tubes to severe accident conditions needs to be evaluated.

An initial small leak in a pressure tube can be detected by reactor operators in about 20 minutes. After detection of a leak, however, it takes a relatively long time to shut down reactor power due to the fact that rapid power shutdown will put the pressure tube below the DBT temperature when the hoop stress is still high. Thus, it is necessary for reactor operators to slowly decrease the coolant pressure, in conjunction with a gradual decrease in power, and hence, the temperature of the pressure tubes. This safe shutdown process seems to require 10–20 hours. The impact on leak-before-break of crack growth rates under ACR-700 environmental conditions during the safe shutdown period will need to be considered.

Pressure tube leakage is essentially an initiating event that could lead to a small- or a large-break LOCA. Compared to current reactors, the design life of the ACR-700 is longer (60 years), the retubing interval is longer (30 years); the operating temperature of the pressure tube is higher (278–325°C); the irradiation levels are higher; the coolant may be more aggressive; and the operating hoop stress is higher (12–13.2 MPa). Therefore, an understanding of the potential for pressure tube failure via hydride blistering and DHC is needed.

The Canadian Nuclear Safety Commission (CNSC) has identified a Generic Action Item entitled, "Pressure Tube Failure with Consequential Loss of Moderator." The concern associated with this CNSC Generic Action Item is that a pressure tube could have an axial rupture due to toughness degradation from radiation damage or hydride formation. Fuel bundles inside the pressure tubes could potentially act as projectiles during the rupture event. Additionally, a concern exists that the axial crack could turn in the circumferential direction and result in a guillotine break. The whipping of the pressure tube with a guillotine break could break the thinner surrounding calandria tube, and may then cause failure in subsequent calandria and pressure tubes. Establishment of an accurate crack-growth-rate data base is more important for the ACR-700 than for the LWR from the leak-before-break (LBB) perspective because the margin for LBB could be much smaller in the ACR-700 than in the current LWRs.

In past service history, two pressure tube axial ruptures[1] have occurred. The Canadian Deuterium-Natural Uranium Reactor (CANDU) Owners Group has conducted several full-scale tests, which were reviewed by the CNSC. That review showed that the full-scale multi-channel burst tests that have been conducted up to that time did not

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[1] The two pressure tube failures in service were at Bruce "A" Unit 2, pressure tube N06, and Pickering Unit 2, pressure tube G16.

adequately represent conditions that reflect the worst-case operating condition (i.e., the crack-driving force and water temperature in the tests were too low compared to service conditions), and the pressure tube materials had much higher toughness than irradiated or hydrided tubes in service would have. For pressure tube rupture issues in the ACR-700 plant, it is important to note that the same Zr-2.5Nb material used in current LWRs will be used in the ACR-700 for the pressure tubes thus making them susceptible to irradiation damage and hydride embrittlement. Consequently, additional evaluations are needed to assess the pressure tube integrity for ACR-700 approval. Since pressure tube ruptures have occurred, this is a realistic severe accident consideration that is driven from material property aging effects.

IV.2.4.4 Application of Research Results

Results from the research described will provide the necessary information to estimate the probability of component failure as input to 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, as well as 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 to provide the technical basis or criteria for materials acceptability. Aging effects and degradation due to the high-temperature helium environment and high radiation dose 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 needed 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 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, and HTGRs. As part of its commitment to participate in the development of industry standards, 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.

Research is needed to evaluate the containment, confinement, aging, inspection, material aspects, and challenge of external events for the HTGR and ALWR designs. Based on the findings, 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 nuclear power plants (NPPs). Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," is one of the new guides. It lists both the Lawrence Livermore National Laboratory (LLNL) and 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 in which the use of the absolute value of the estimate was important. 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 experience on seismic response of deeply buried structures, research insights will be 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 the effects of high temperatures on the 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 1990s, 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 to evaluate 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 the 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 pre-stressing hardware.

IV.2.5.2 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 needs 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 ALWR designs are upgrades, advancements, and simplifications to certified designs or currently operating 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 applied 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 are 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). These standards are currently utilized in domestic operating nuclear power plants. While these reactors utilize some structural design and construction processes similar to the ALWRs' and to existing operating nuclear power plants, certain unique structural design aspects need to be evaluated.

In the area of probabilistic seismic hazard assessment, research needs to 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, known as 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 needs to 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 needs to be conducted to focus on accumulating the existing database, expanding the database, and evaluating the impact of high temperatures 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 (1) the need for evaluation criteria because the existing U.S. codes and standards do not address composite structures (concrete-filled steel plate modules) and (2) the need for verified design/analysis methodology for unique types of modules, such as the concrete-filled steel plate wall and floor modules.

Research needs to be conducted to develop methodologies for RI-ISI of containment and associated components such as liners, bellows, and pre-stressing hardware. This research needs to 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 Objectives and Associated 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 need to 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 guidelines and the technical basis for regulatory criteria that reflect the latest knowledge and understanding in this area. 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., developed by LLNL and EPRI, making use of a set of guidelines developed by the NRC and DOE with EPRI, known as the SSHAC methodology. With a single update methodology accepted by NRC, the controversy associated with selecting one of the current two methodologies, will be reduced, if not eliminated.

Implementation of the SSHAC methodology is to be primarily carried out by the NRC making use of panels of seismicity and ground motion experts. The NRC staff, with contracted assistance, would (1) assemble the expert panels; (2) elicit from them the basic seismic hazard data; (3) compute the individual seismic hazard assessments; (4) analyze and interpret the results; and (5) become experts in the methodology and its use.

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 research and development (R&D) organizations and regulators needs to be reviewed for applicability and to determine gaps where additional research is needed. Analytical and experimental research needs to be conducted to develop seismic and structural analysis models of reactor vessel internals and core support structures and to perform seismic analyses for horizontal and vertical earthquakes. The assumptions and limitations of existing finite element analysis codes would be evaluated for applicability to nonlinear configurations, such as the HTGR reactor components consisting of nonductile graphite core reflectors and supports. A special need exists 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 within 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 would 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 would 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 may 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 would 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 would 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 would 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. Earlier research on LWR severe accidents conducted by Sandia National Laboratories also accumulated significant data on the effects of high temperatures on the properties of concrete. Oak Ridge National Laboratory has also assembled information on concrete subjected to high temperature. Lessons learned from facilities at which concrete was found to be subjected to high temperatures for long durations would also be investigated and utilized.

Modular construction: Modular construction has not been used in the United States for nuclear power plants but some techniques have been used in Japan. Many designers have proposed the use of 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, joints, and connections as well as appropriate damping values for seismic analyses. Presently, U.S. codes and standards guidance are lacking, or nonexistent, 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 would focus on developing evaluation criteria that would facilitate review of reactors that use modular construction. The NRC staff would use the results of earlier research described in NUREG/CR-6486. Calculation methods would 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 (American Concrete Institute 349, "Nuclear Safety Related Concrete Structures," and American Institute of Steel Construction, N690, "Nuclear Facilities-Steel Safety Related Structures-Design Fabrication and Erection") and required code changes would be made. Regulatory guidance would 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 ISI of structures: Because of the commitment to risk-informed processes, it is anticipated that existing ISI requirements for containment structure and structural components would be replaced or augmented by RI-ISI programs. Research needs to be conducted to develop RI-ISI methodologies for ISI of containment and associated components such as liners, bellows, and pre-stressing 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.

The ASME has formulated a Task Group to develop methodologies for RI-ISI of containments. The NRC staff would actively participate in this Code activity while independently developing the methodologies for RI-ISI of containments. Research for this item would 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. The ISI parameters, such as the amount of inspection and frequency of inspection, would be based on the risk categorization of the structural component. It is expected that the RI-ISI approach would result in focusing inspections on risk-significant areas while reducing unnecessary regulatory burden.

IV.2.5.4 Application of Research Results

The end product of this work would 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 would 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 would be that the probabilistic hazard estimates from the implementation of the SSHAC guidance and associated methodology would 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 safe-shutdown earthquake determination. Possible outcomes of new seismic analysis techniques would 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 would 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 would 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 Background

Offsite consequence analysis is the final aspect of PRA, 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 Purpose

Normal input to NRC's Level 3 evaluation code, MACCS2, 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 taken into account. 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 Objectives And Associated Activities

There are 87 parent and daughter radionuclides presently considered in MACCS. The impact on offsite 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 need to be added to the library. If new chemical forms are important, revised dose and uptake factors need to be made available. Other analyses would give a final list of radionuclides produced, but this research would evaluate the biological importance. In a similar manner, the Level 2 analyses will give the chemical form of the released material, but this research would evaluate the needed factors.

IV.2.6.4 Application of Research Results

Research results would 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 need to be commensurate with the calculations utilized in choosing the current 10-mile EPZ for today's light-water reactor plants. These calculations are referred to in NUREG-0654 (Federal Emergency Management Agency-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." The choice of the size of the EPZ is also discussed in this document. 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 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 (1) fission reactor neutronics, both static and dynamic; (2) 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; (3) radiation transport and attenuation as applied to the evaluation of material damage fluence, material dosimetry, material activation, radiation protection, and radiation detection, and (4) nuclear criticality safety (i.e., the prevention and mitigation of critical fission chain reactions ($k_{eff} \geq 1$) outside reactors).

This section addresses nuclear analysis issues encountered in nuclear materials safety and waste safety. Nuclear analysis research efforts for reactor safety and safeguards 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 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 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 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 and (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 the PBMR, GT-MHR, and IRIS.

Nuclear criticality safety at the front end of the fuel cycle[2]: Enrichment plants, fuel fabrication facilities, and transportation packages for LEU commercial LWR fuel materials and fuel assemblies are not presently licensed to handle uranium enrichments significantly above 5 weight percent (5wt%)^{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

[2]See also separate sections on Fuel Manufacture (Section IV.3.2) and Transportation and Storage (Section IV.3.3).

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.

- **Safety analyses for spent fuel management**[3]

Nuclear analysis issues for storing, shipping, and disposing of the high-burnup spent fuels and underburned fuels discharged from PBMR, GT-MHR, and IRIS will involve the assessment of modeling assumptions and approximations, needs for specific validation data, and validation of uncertainty treatments in the prediction of (1) long-term decay heat sources for cooling; (2) radiation sources for shielding; and (3) 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 Objectives and Associated Activities

The NRC research objectives for this area 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 research activities that pertain 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:

- <u>Preparation of modern cross-section libraries</u>. (See Section IV.2.2, Reactor Safety, Reactor Systems Analysis.)

Nuclear criticality safety at the front end of the fuel cycle:

- <u>Criticality Validation and Modeling Guidance for (a) PBMR, (b) GT-MHR, and (c) IRIS Fuel Materials</u>. 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.

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[3]See also separate sections on Transportation and Storage (IV.3.3) and Disposal (IV.3.4).

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:

- Validation and modeling guidance for applying burnup credit in criticality safety evaluations involving spent fuel. 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.

- Validation and modeling guidance on predicting decay heat and radiation sources in spent fuel. Building upon closely related work on burnup credit (see 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. 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 Application of Research Results

Results from the research activities described above would 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 advanced reactor fuel cycles. As outlined in the preceding sections, the nuclear analysis research activities would result in developing the staff's technical insights in these areas and applying those insights toward establishing independent review and analysis capabilities. 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 in the Section IV.2, Reactor Safety.

IV.3.2 Uranium Enrichment And Fuel Fabrication

IV.3.2.1 Background

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

IV.3.2.2 Purpose

Insights from activities at existing fuel manufacturing facilities in the areas mentioned above will be provided to identify safety issues and pathways to resolution.

IV.3.2.3 Objectives and Associated Activities

Reports to the NRC from the existing fuel manufacturing facilities need to 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 need to 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 Section IV.3.1.3, Nuclear Analysis for Material Safety and Waste Safety.

IV.3.2.4 Application of Research Results

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

IV.3.3 Transportation and Storage

IV.3.3.1 Background

Regulatory requirements and technical guidance documents already exist for the packages and casks used in transporting fresh fuel and spent fuel under 10 CFR Part 71, for the at-reactor storage of fresh and irradiated fuel under Part 50, and for the storage of spent fuel in casks under Part 72. However, some advanced reactor fuels will differ substantially from existing LWR fuels both in physical form (for instance, 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 such as (1) the assessment of high-burnup (80 GWd/t) cladding integrity for

IRIS spent fuels in storage and transport casks and (2) the application of burnup credit in the criticality safety evaluations for spent fuels from PBMR, GT-MHR, and IRIS[4] will take on significant new aspects in relation to the corresponding issues for conventional LWR fuels. Therefore, the continued applicability of existing requirements and technical guidance to the changed conditions may need to be reviewed. Transportation and storage of spent fuel are issues of especially high public concern.

IV.3.3.2 Purpose

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 will be evaluated.

IV.3.3.3 Objectives And Associated Activities

A review of the data and analyses supporting existing storage and transportation regulations, and associated technical and regulatory guidance documents, needs to be undertaken to determine continued applicability to advanced reactor fuels. Physical differences between existing fuels and proposed fuels need to be considered. If the existing data and analyses are found not to apply to proposed fuels, applicable data and analyses of similar types would be identified and provided, where feasible. The review would identify any areas in which changes or clarifications may be needed in the regulations and/or 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, would be addressed through the nuclear analysis efforts described in Section IV.3.1.1.

IV.3.3.4 Application of Research Results

Applicants and technical reviewers for the transportation and storage of proposed advanced reactor fuels would 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 High-Level Waste

IV.3.4.1.1 Background

The NRC staff currently uses a risk-informed and performance-based approach to assess the adequacy and limitations associated with the disposal of high-level radioactive waste (HLW) in a waste disposal repository in terms of meeting the design objectives to support regulatory requirements and compliance criteria. In the U.S., sufficient study and analyses have not been performed with respect to advanced reactors generating graphite and other types of HLW. For

[4]See the Section IV.3.1, Nuclear Analysis for Material Safety and Waste Safety.

the disposal of radioactive waste generated by advanced reactors (e.g., PBMR, GT-MHR, and IRIS), there are limitations in basic spent fuel behavior knowledge and long-lived radionuclide source term parameters and data. Qualified information is necessary to support regulatory decisions and to estimate the long-term dose and risks to the reasonably maximally exposed individual. Long-lived radionuclide inventories of advanced reactor spent fuel and radionuclide source term releases from advanced reactor spent fuel under repository disposal conditions are not available for advanced reactor fuel having high enrichments (>5–8% and possibly up to 20% ^{235}U) and burnup levels to 80 GWd/t.

In the absence of realistic data and information on advanced reactor spent fuel and reactor systems, the use of conservative estimates of model parameter values leads to an oversimplified performance assessment of a complex disposal facility that could significantly underestimate or overestimate individual exposure. In this case, opportunity and obligation exist to improve NRC's performance assessment capabilities. The advanced reactor research program needs to address uncertainties associated with the disposal of advanced reactor spent fuel and reactor materials to improve the efficiency, effectiveness, and realism of agency analyses and decisions involving the performance of HLW repositories. Research information needs include long-lived radionuclides source term releases, higher fuel burnup and enrichment parameters, the effect of increased storage volumes of advanced reactor spent fuel and materials (e.g., graphite) a re-evaluation of criticality codes, and the effect on transportation of an increased amount of advanced reactor spent fuel.

IV.3.4.1.2 Purpose

The purpose of the advanced reactor waste disposal research is to provide realistic data and information to obtain defensible estimates of radionuclide exposure to the reasonably maximally exposed individual from radionuclides transported to and released from a HLW repository containing advanced reactor spent fuel.

Research is needed to identify differences in radionuclides and concentrations in advanced reactor fuel from typical LWR fuel; to determine radionuclide inventories for advanced reactor fuel; to understand advanced reactor fuel behavior under repository disposal conditions; and to determine model, parameter, and data uncertainties to estimate radionuclide source term releases. The research on advanced reactor fuel behavior and radionuclide source terms would focus on PBMR fuel elements, TRISO-coated fuel particles, and other advanced reactor fuel types. Emphasis would focus on obtaining experimental data and information under varying enrichment, burnup, and chemical disposal conditions. Research on the transport of radionuclides in the environment, biosphere pathways, and volcanism release scenarios is not expected to be included in this research program, but could be performed in those situations in which the radionuclides present in advanced reactor fuel were found to be either different from or above the dose impact threshold of those radionuclides currently used in performance assessments using typical LWR spent fuel.

Further research is needed to develop confidence in advanced reactor performance assessment methodology and computational aspects by modifying or 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. Much of the data and information on FPs, transuranics and activated metals needed for risk-informed and performance based assessments for the licensing of repositories containing advanced reactor fuel is not available. If available, the data is generally of either poor quality or have been obtained under conditions that differ from those that could be expected to be present in a HLW repository. Parameter data generated by this research program will be used to quantify uncertainties associated with the disposal of advanced reactor fuel in a waste disposal repository and to update and modify source term computer codes used in HLW repository performance assessments.

Research in advanced reactor issues is also needed to understand the effects of the increased volume of waste generated by advanced reactor spent fuel. The spent fuel volume for PBMR is expected to be 10 to 20 times higher and GT-MHR two to five times higher per MWD than that generated by LWRs. The information will be used to re-evaluate the transportation assumptions in risk studies on fuel transportation and to evaluate the source term implications of different storage configurations necessitated by larger volumes.

In addition, higher enrichment issues must be evaluated for fuel fabrication plant operations and for potential handling and storage at the waste site. These latter issues are important because GT-MHRs may require enrichment of up to 20%, but current enrichment capacity at Paducah is only five%; current criticality codes are validated only for enrichments up to five %.

IV.3.4.1.3 Objectives and Associated Activities

Certain information necessary for this project will be provided by other activities discussed elsewhere in this plan. For instance, the inventories of long-lived radionuclides in advanced reactor fuel; the behavior, including the chemical form, of advanced fuel under varying enrichment and burnup conditions; and criticality tools validated for high enrichments will be provided by the nuclear analysis research.

The overall objectives of the advanced reactor waste disposal research program are to (1) scope and capture radionuclide source terms under varying burnup and repository chemical and physical conditions; (2) improve existing radionuclide source term models and computer codes for assessing the performance of a HLW repository containing advanced reactor spent fuel; (3) determine the releases of radionuclides from a repository containing advanced reactor spent fuel to the environment as a function of time up to 10,000 years; (4) require that analytical methods and all radiological, chemical and physical data used to predict radionuclide releases to and behavior in the environment be validated against critical experiments in order to establish the calculational bias and uncertainty; (5) provide data in probabilistic distribution and associated statistical parameters format for use in risk-informed and performance-based assessments; (6) quantify chemical effects that may impact the parameters that control radionuclide releases, mobility, solubility, sorption etc.; (7) assess impacts of increased volume of waste; and (8) evaluate increased transportation needs.

Research is needed for the following issues:

- Evaluate advanced reactor spent fuel behavior characteristics (e.g., microstructure, grain growth, texture, radionuclide distributions).

- Determine advanced reactor spent fuel dissolution rates in varying water environments (e.g., water, drip drop, aqueous film) and under varying physical and chemical conditions.

- Determine radionuclide release rates from advanced reactor spent fuel under varying physical and chemical conditions.

- Determine solubilities of important radionuclides released from advanced reactor fuel under varying chemical conditions.

- Obtain data on fuel rod or element corrosion/dissolution under repository chemical conditions. This would be performed only for those situations in which fuel cladding or fuel element differ from those cladding materials currently used in LWRs and disposed of in a HLW repository (i.e., graphite fuel elements).

- Evaluate repository near-field chemistry effects on advanced reactor spent fuel and cladding behavior under varying burnup and fuel enrichments conditions.

- Determine the presence of radiocolloids formed due to the presence of advanced reactor spent fuel, cladding, and other materials present in the repository.

- Determine transport of important radionuclides (e.g., ^{14}C from graphite) and sorption characteristics of radionuclides in unsaturated and saturated groundwater only for those radionuclides that may be unique to advanced reactor releases.

- Assess increased waste volume storage and transportation needs.

- Re-evaluate assumptions of NRC's Transportation Risk Study (NUREG-0170).

- Evaluate enrichment effects at fuel fabrication plants, on transportation of waste and in storage configurations.

IV.3.4.1.4 Application of Research Results

Many results would be incorporated into NRC's HLW performance assessment computer codes. The research results are expected to be used to provide an independent basis for evaluating and auditing the applicant's advanced reactor data, information analyses, models, and computer codes. 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 licensing. In addition, the research results are needed to support the development of regulatory criteria and resolve key technical issues associated with the licensing and approvals of a HLW repository containing advanced reactor waste.

IV.3.4.2 Low-Level Radioactive Waste and Site Decommissioning

IV.3.4.2.1 Background

Onsite radionuclide behavior and releases of radionuclides to the environment need to be understood from the perspective of low-level waste (LLW) disposal facilities containing waste streams (e.g., graphite materials) from advanced reactors (e.g., PBMR, GT-MHR, and IRIS), as well as from decommissioning sites containing materials used in advanced reactors. This research is needed 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 as part of the effort to ensure compliance with the regulatory requirements of 10 CFR Parts 61 and 20 and the policies in the Decommissioning Standard Review Plan (NUREG-1727).

Determining radionuclide releases from advanced reactor LLW and decommissioning materials under varying chemical and physical conditions is an important aspect in determining source terms and assessing the performance of LLW disposal facilities and reactor decommissioning sites. To calculate radionuclide releases from advanced reactor LLW disposed in LLW disposal sites and from decommissioning materials at decommissioned advanced reactor sites, consideration must be give to the radionuclide inventories (both for the LLW and decommissioned materials), radionuclide releases, and solubilities. For LLW, additional data and information on waste types and forms and information on waste containers are required.

For decommissioning advanced reactor sites, it is helpful to provide data and information about decommissioning activities in order to establish specific decommissioning plan requirements. For example, will certain radionuclides present in decommissioning materials present a unique decommissioning? For GT-MHR reactors, ^{14}C is generated in the graphite and carbon dust can pose a hazard during decontamination. Silver-110m can diffuse through the fuel, accumulate within reactor components and could pose a hazard from routine reactor operations. There is concern that novel decontamination methods may be needed and higher decommissioning costs may be encountered when dealing with the decommissioning of advanced reactors. This highlights the need to consider decommissioning issues during the design phase as required by 10 CFR Part 20.

Uranium enrichment produces depleted uranium as a by-product and, because of their higher enrichment needs, advanced reactors may cause additional environmental impacts. For a typical LWR between 6 an 8 tons of depleted uranium are produced per ton of fresh fuel. A GT-MHR could generate twice as much depleted uranium and create concerns about the impacts from disposal of depleted uranium mill tailings.

IV.3.4.2.2 Purpose

The purpose of the advanced reactor LLW and decommissioning site research is to (1) determine if the radioactive waste generated from the operation of the advanced reactors and the long-lived radionuclides present in the waste are different from the waste and distribution of radionuclides in current LWRs licensed by NRC; (2) determine if the Part 61 waste classification system makes some advanced reactor waste streams ineligible for disposal in LLW disposal facilities; (3) estimate how much LLW is generated by advanced reactors and determine the important waste streams and types of advanced reactor LLW; (4) determine if there are long-lived radionuclides present in advanced reactor LLW that are not present in LLW; (5) assess the importance of activated metals in advanced reactor LLW; (6) evaluate how much transuranic waste is generated by the advanced reactors; (7) determine if packaging and shipping requirements have to change; (8) assess disposal impact of higher enrichment on GT-MHR depleted uranium; (9) assess decommissioning and plant operation hazards; and (10) evaluate safe storage issues for waste generated by advanced reactors.

IV.3.4.2.3 Objectives and Associated Activities

The primary objective of the advanced reactor LLW disposal and site decommissioning research program is to provide experimental data and information that can be used to determine realistic radionuclide inventories in waste streams; calculate realistic radionuclide source term releases from LLW disposal facilities and decommissioned sites; determine decommissioning, operation, and maintenance hazards; evaluate the impacts of higher enrichment on the amount of depleted uranium; and assess safe store options.

Research is needed on the following issues to obtain the objectives:

- Characterize advanced reactor LLW waste and decommissioning materials for radionuclide and chemical content.

- For LLW, determine radionuclide concentrations by waste stream, waste type, waste form, and waste classification.

- Identify differences between advanced reactor LLW streams and radionuclides and the LLW waste generated by current LWRs.

- Determine radionuclide releases, including the chemical and physical factors affecting releases, by performing laboratory and field leaching studies on advanced reactor waste, including activated metals, and decommissioning waste materials.
- For important long-lived radionuclides which may be present only in advanced reactor waste and not in typical LLW waste, determine sorption coefficients using soils typically found at LLW disposal facilities and decommissioning sites.

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

- Understand issues involved in dismantling and dispositioning of graphite used as a moderator, reflector or other structural purpose.

- Determine if ^{14}C, other radionuclides, or carbon dust presents a unique decommissioning challenge or poses a hazard during decontamination.

- Determine if ^{110m}Ag or other radionuclides pose a hazard during routine operations.

- Assess the importance of using novel decontamination methods for decommissioning purposes.

- Develop assessment tool and evaluate the impact of high enrichment (up to 20% ^{235}U) on the amount of depleted uranium from fuel fabrication processes that must be disposed.

- Determine suitable safe storage parameters for spent fuel and waste materials from advanced reactors.

IV.3.4.2.4 Application of Research Results

The results obtained from this research program are expected to be used to (1) support the development of regulatory criteria (e.g., regulations, regulatory guides, policy guidance, standard review plans) for the disposal of LLW and depleted uranium generated by advanced reactor operations and the decommissioning of advanced reactor sites; (2) provide an independent basis for evaluating an applicant's data, information, analyses, and computer codes; and (3) provide a base of physical data and scientific information to quantify uncertainties in the technical basis of licensing waste disposal facilities, uranium milling sites, decommissioning sites, and safe storage assessments.

IV.3.5 Personnel Exposure Control During Operation[5]

IV.3.5.1 Background

Because 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 FP silver from the grains of the fuel into the gas stream. Silver-110m, 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 Purpose

The purpose of this research is to ensure that the operational aspects of new reactor designs minimize personnel exposure. This research would systematically search new designs for different exposure issues, such as the ^{110m}Ag issue for the PBMR and GT-MHR, and evaluate the issue of radiation streaming due to changes in graphite geometry.

IV.3.5.3 Objectives And Associated Activities

This work would 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

[5] Applies to reactor safety as well as materials and waste safety
[6] See related activities described in the section on nuclear-grade graphite under Reactor Safety.

associated radiation streaming issues[7] in view of potential concerns over vessel fluence[8] as well as radiation protection. In addition, evaluate different advanced reactor designs would be evaluated to identify any other issues that may pose radiological hazards that differ from those in conventional LWRs.

IV.3.5.4 Application of Research Results

Reviewers of the design as it relates to personnel exposure control would be provided with insights generated from this research.

IV.4 Safeguards

IV.4.1 Background

The fuel elements for PBMR and IRIS will be enriched up to 8 wt% ^{235}U; those for GT-MHR will be enriched 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 centimeters [cm] diameter), very large in number, and not individually marked with identifiers, thus making material control and accounting (MC&A) 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 MC&A measures throughout the fuel cycles of the respective advanced reactor designs.

IV.4.2 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 MC&A 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.

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[7] See related activities described in the Section IV.3.1, Nuclear Analysis for Material Safety and Waste Safety.

[8] See related activities described in the section on high-temperature materials under Reactor Safety.

IV.4.3 Objectives and Associated 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 need to be performed to develop a set of industries to serve as benchmarks. As part of the larger safeguards evaluation efforts, the relative ease and desirability of material diversion need to be examined through nuclear analysis activities described elsewhere. In addition, the technological barriers to extracting plutonium and other radionuclides from irradiated fuel materials would be described for the respective advanced reactor technologies.

Specific activities include:

Material diversion safeguards: Nuclear analysis tools and methods need to be used in the arena of material diversion safeguards for the assessment of weapons proliferation potential and radiological threats, material security technology, and the 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 raises 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) would 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.

Scoping studies on proliferation resistance of PBMR and GT-MHR fuel cycles: Postulated scenarios for overt and covert production of weapons-usable plutonium in the respective fuel cycles may need to be analyzed. Credible postulated scenarios should be developed which involve the introduction, early discharge, and diversion of standard fuel elements, as well as special Pu-production fuel elements. Calculations to predict associated radionuclide inventories, including the quantities and isotopic compositions of plutonium produced per fuel element would need to be performed. Using credible assumptions regarding specific MC&A and material security measures, a comparison of 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 could provide useful insights. 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).

Assessment of technical requirements for MC&A and material security in the PBMR and 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 could be assessed. This evaluation should consider standard, as well as special requirements for MC&A and material security technology. A comparison could be conducted between the material diversion potential of the PBMR and GT-MHR fuel cycles and that of the major reactor types in operation around the world today, including LWRs and CANDUs. Recommendations and options could then be developed regarding any special measures needed for reducing the diversion potential in the respective advanced reactor fuel cycles.

IV.4.4 Application of Research Results

This research would provide reviewers with relevant MC&A benchmarks from other industries, for use in establishing a technical basis for potentially 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 prepare NRC for independent regulatory review of advanced reactor applications and to develop the associated regulatory infrastructure including data, codes and standards, and analytic tools, a prioritization method is needed to help allocate available resources. The purpose of the advanced reactor research program prioritization is to provide an effective method for allocating resources among the different elements in the research program, taking 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 importance of the research to the development of policy recommendations.

The RES has developed and used the PIRT process as a tool for identifying and prioritizing research needs. The PIRT process, and related approaches previously used by RES (e.g., 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 Laboratory, September 1993.

(3) D.E. Carlson and R.O. Meyer, NUREG-1502, "Assessment of Database and Modeling Capabilities for the CANDU-3 Design," U.S. NRC, 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 the following issues: (1) AP-600 test and analysis need, (2) performance of high-burnup LWR fuels in reactor accidents, and (3) 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. This document will be used 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. The PIRT team will consist of 6 to 10 NRC staff and contractors or type (e.g., PRA, thermal and fluid flow, nuclear analysis, fuel fabrication and performance, FP transport, materials, systems, structures, and components, containment/confinement, human factors, I&C, maintenance and inspection).

For the PBMR and GT-MHR, this umbrella PIRT activity will build upon results from (1) the October 2001 NRC Workshop on HTGR Safety and Research Issues, (2) the June 4, 2001, ACRS Subcommittee on Advanced Reactors meeting, and (3) relevant NRC pre-application review and research efforts conducted during the 1985-1995 time frame for the DOE MHTGR design. Reference 2 above, an RES contractor's PIRT-like report on MHTGR safety evaluation will also be used.

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. The selected event sequences will initially encompass phenomena in the licensing-basis events 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., FP 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. As discussed in Section V, a systematic and logical 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 safety issues will be key. Discussions between the NRC and the applicant during the pre-application review phase should help identify the information gaps as well as the additional analytic 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 or uncertainties are large?

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

- Does NRC have the required material (e.g., German pebble fuel or decommissioned AVR in-vessel specimens) to be able to conduct the tests itself? 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 proactive 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 so that 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 NRC goals?
- What is the feasibility of a joint venture with the industry?
- Can the required information be purchased internationally or domestically for a 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?
- Does NRC have required core staff expertise? If not, can the agency hire new staff/retirees to bridge the critical skill gap?
- Does NRC have appropriate contractor staff and facilities to conduct and support the desired research, generate data, or develop the desired tools?
- How much time and resources are needed for quality checks or independent testing/validation of the end-product? Does NRC 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?)

- Does NRC have provisions in the budget for the next 5 years to support the research? What are the implications if the agency is not able to sustain the necessary research to completion?

VII. COOPERATIVE RESEARCH

Unlike proven LWR technology for which extensive LWR-related operational worldwide experience exists, operational experience related to the GCRs is limited, and of the available data, some may not be directly applicable. For instance, while the graphite-related 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), and operating conditions (higher temperatures expected in the HTGRs), as well as factors such as high enrichment and burnup, 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 (e.g., 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. The NRC’s active participation in ongoing research programs and new cooperative efforts with various international organizations needs to be formulated so as to deliver optimum mutual benefits while off-setting costs.

VII.2 Relevant International Efforts

There is extensive GCRs operational experience for GCRs in Germany and the 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 GCR designs. The existing AVR operational experience and data provide significant insights in identifying the future research needs. It is also believed that HTR-10 in China, HTTR in Japan, and HFR in the Netherlands will play a crucial role in providing the necessary experimental data and means for code validation to the international HTGR community. 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 for example, characterization of physical properties of new graphite and testing of new materials, under the Russian Federation and the E.C.'s high temperature GCR research 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

- participation in IAEA's various CRPs; (e.g., the one on evaluation of HTGR Performance, another related to fuel technology development and one on code validation using data from HTR-10 and HTTR)

- efforts related to a graphite database development under the sponsorship of IAEA

VII.3 Workshops and Meetings

The staff hosted and participated in the following conferences and workshops:

In the year 2001, three advanced reactor workshops and one conference 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 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 a vested interest, such as vendors, builders, and potential applicants. Participants included various HTGR experts from China, the E.C., Germany, Japan, Russia, South Africa, the United Kingdom, United States 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; the potential for several opportunities for international cooperative research were identified. Various venues for future international cooperation were also identified. In October 2001, at the Nuclear Safety Research Conference sponsored by NRC's RES, a special session was held on safety and research issues related to the advanced reactors. This conference provided a forum for dialogue among various participants to develop useful insights in planning future research programs.

In the year 2002, the NRC sponsored one workshop and one conference, and the staff participated in various meetings, conferences, and workshops. An NRC staff member was also part of the US delegation that visited the Russian Federation. These activities are briefly described below:

In February 2002, the RES Director 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 advanced reactor designs being considered worldwide and also served as a forum for valuable discussions on advanced reactor concepts being studied and for identification of safety issues and research needs.

In March 2002, at the Regulatory Information Conference sponsored by NRC's NRR, the staff conducted a session on new reactor licensing. The theme of the session was "Regulatory Challenges Associated With New Plant Licensing Actions." A panel of experts from NRC, DOE's Near-term Deployment Roadmap Panel, NEI, and a public interest group representative responded to a variety of questions from the audience. Also in March 2002, NRR sponsored a Legal and Financial Issues Workshop.

In SECY-01-0207, "Legal and Financial Issues Related to Exelon's Pebble Bed Modular Reactor (PBMR)," dated November 20, 2001, the staff provided its preliminary position on operator staffing, fuel cycle impacts, financial qualifications, decommissioning funding, minimum decommissioning costs, antitrust review, number of licenses, annual fees, financial protection, and testing requirements for a combined license.

On March 27, 2002, the staff held a public workshop to discuss the positions presented in SECY-01-0207 with Exelon and other stakeholders. Following the workshop, GA submitted written responses to the issues discussed in the paper, and NEI requested a meeting with the staff for additional discussions. Based on these interactions, the staff revised its positions, as appropriate. A revised Commission paper, including recommendations on policy issues related to the legal and financial matters of the PBMR, was submitted for Commission approval on October 17, 2002, (SECY-02-0180), "Legal and Financial Policy Issues Associated with Licensing New Nuclear Power Plants."

In April 2002, the staff participated at the International Conference on Nuclear Engineering-10 in Washington, DC. Also in April, the staff attended the HTR conference held in Petten, the Netherlands. An NRC staff member was part of the US delegation that visited the Russian Federation in July 2002. The purpose of this visit was to determine first hand the status of the design of the Russian GT-MHR, to learn the technical safety challenges in building the new reactor, and to observe the technical capabilities of the Russian companies who might manufacture components that could be used in the United States for a commercial GT-MHR.

VII.4 Cooperation and Exchange of Experts

Agreements for future corporation with Japan and China are being finalized. Details of an agreement for cooperation with the E.C. are also being discussed. Furthermore, foreign regulators assigned to the NRC have participated in advanced reactor activities such as the pre-application review of the AP-1000 design. Recently, an NRC staff member has been assigned for 4 months to the NII to learn about regulatory challenges (technical safety issues and regulatory practices) in using graphite as a structural component and as a moderator in GCRs. The work plan includes a judicious mix of interaction with industry and academic technical experts in the UK, and the regulatory personnel at NII; observer-status inspection trips with NII inspectors to licensee plants; active interactions with researchers at University of Manchester; participation at the IAEA meeting on graphite properties database development; participation at the 3rd International Graphite Specialists meeting in Parma, Ohio, and safety assessment training conducted by the Nuclear Safety Directory of the Health and Safety Executive. It is anticipated that this exchange of experts will continue.

The staff continues additional dialogue with international partners to explore prospects for future cooperation on HTGR-related efforts.

VII.5 Prospects for Future International Cooperation

Since the beginning of the PBMR pre-application review process in 2001, US delegations have visited South Africa, the 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, such as the Nuclear Safety Research Conference, held in November 2001, and the Regulatory Information Conference, 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 E.C. representatives. 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 along with their technical counterparts and other key researchers to further discuss the details of possible future cooperation.

It is anticipated that future NRC collaboration with the E.C.'s High Temperature Reactor Technology Network research program is expected to be extremely beneficial. In particular, this cooperation will be in the areas of: (1) reactor physics and fuel cycle, (2) high-temperature materials (including nuclear-grade graphite) performance, (3) fuel performance and qualification, and (4) licensing framework development. Recently, steps have been completed to initiate cooperation with the E.C. as part of the formal NRC-European Atomic Energy Community agreement, and also with Japan and China. Formal agreements are currently being finalized. Additional dialogue is necessary to formulate formal agreements with 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 continuing exchange of experts, as presently being done with the NII in the UK.

The staff is also interested in exploring prospects of working with the British and Canadian regulators on activities related to the licensing of the ACR-700. In that context, cooperation with other CANDU countries, such as the Republic of Korea and India, might also be explored in the future.

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 ventures with existing or 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, to give an up-to-date status of various associated and ongoing collaborations.

VII.6 Participation in the IAEA-Sponsored Programs

The IAEA’s documented data from various CRPs, as well as international conference proceedings found in various TECHDOCs, represent a significant information base. In 1991, a specialists' meeting was held on the subject of graphite development for GCRs. This meeting was jointly sponsored by IAEA and the JAERI. In 1995, another meeting on graphite moderator life-cycle behavior was held in the UK (TECHDOC-901). With support from Japan, South Africa, the UK, and United States (U.S.), the IAEA has begun development of a database related to irradiated nuclear graphite properties. The objectives of this effort are 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, the objectives of the International Working Group on Gas Cooled Reactors are 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°C, R&D on very high-burnup fuel, R&D and component testing of high efficiency recuperator designs, and materials development for turbine blades (up to 900°C) for long creep life. The duration of this CRP is from 2000 through 2005. Continued U.S. participation in this and similar CRPs will certainly be beneficial.

The NRC plans to actively participate in the IAEA-sponsored research and development efforts related to the HTGRs, including various CRPs. The NRC staff currently assigned to NII under the technical experts exchange program will represent NRC at the IAEA meeting on graphite properties database and will also participate in the 3rd International Graphite Specialists meeting to be held in Parma, Ohio. Additionally, following participation in April 2002, at the HTR conference held in Petten, the Netherlands, the staff has been discussing NRC participation in CRP-6. This CRP on coated particle fuel development is part of IAEA’s HTGR fuel technology development research program, and would encompass the following HTGR-related topics: fuel performance, fuel performance modeling and characterization, quality assurance/control, fuel irradiation and accident testing, and fuel licensing issues. Efforts are in process to formalize the details of the scope of work.

VII.7 Participation in OECD/NEA Activities

The NRC anticipates a proactive role in future NEA activities. In early 2002, the RES Director 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 High Temperature Reactor Technology Network 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.8 Cooperation Through DOE

“Nuclear Power 2010" is a DOE initiative intended to build a new nuclear power plant in the U.S. by the end of the decade. Under this initiative, the government and the private sector will work together to (1) identify sites for nuclear power plants; (2) demonstrate the efficiency and timeliness of key NRC processes for licensing of new plants, and (3) conduct research needed to make the safest and most advanced nuclear plant technologies available in the U.S. To this end, DOE is planning to share part of the applicant’s cost of demonstrating the 10 CFR Part 52 licensing process. DOE continues to support HTGR development, and has begun research, with NRC cooperation, to ensure that these technologies can be considered as options in the U.S.

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 NERI program, various reactor designs and high-burnup and enrichment-related research projects are being conducted at various organizations, including 24 U.S. universities, 10 DOE national laboratories, 20 industry organizations, and 24 foreign R&D organizations. There are nine ongoing projects under NERI that relate to the GCR technology. The GCR fuel irradiation program and GCR fuel technology R&D efforts are currently being planned. Of the NERI programs, the projects related to GCRs 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 United States universities, and the Polytechnical Institute of Milan, Italy.

The International Nuclear Energy Research Initiative efforts include collaborative agreements between the United States and France, as well as the United States and Korea on gas reactor technology. The United States-France agreement of May 2001 relates to the joint development of advanced nuclear systems. This agreement is part of DOE's International Nuclear Energy Research Initiative 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 United States 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 agreement between the United States and the Republic of Korea the areas of collaboration include R&D related to 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 burnup, thorium, and particle fuels); and innovative safety research (including advanced computational methods for seismic, thermal-hydraulic, and nuclear analysis).

VII.9 Domestic Efforts

In April 2002, Exelon announced its termination of the PBMR pre-application review activities. Initiation of pre-application review of the GT-MHR continues to maintain urgency for 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 include different advanced reactor concepts. Subsequently, the NRC signed a cooperative research agreement with MIT for the purpose of improving NRC's access to state-of-the-art knowledge and information on advanced reactors. The period of the agreement is from August 2002 to August 2005. The funding ceiling for the agreement is $1.5M. The subject areas to be covered include advanced light-water as well as non-light-water technologies. The MIT is currently conducting research at their Center for Advanced Nuclear Energy Systems on a number of topics associated with gas-cooled and light-water reactors, as well as exploring regulatory improvements using risk perspectives. Under this cooperative agreement, MIT plans to pursue development of a prototypical Pebble Bed Modular Reactor. Through its participation, NRC expects to gain insights into the technological and regulatory issues that will assist in developing an appropriate regulatory framework for addressing the safety issues with this technology.

In a separate venture, the staff is working with Information Systems Laboratories, Inc. on 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. Prospects for additional cooperation with MIT on HTGR-related research topics, such as high-temperature materials, may be explored.

VII.10 DOE-Sponsored Research and Other Initiatives

For many years ending in the early 1990s, DOE sponsored the 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 1990s, DOE initiated a new program called the NERI. The 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 EPRI focus on advanced light-water reactors and research to optimize the operations of the current operating fleet of nuclear plants. The EPRI, in cooperation with the NEI 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.

Finally, a major element of the DOE Advanced Gas Reactor Fuel Development and Qualification Program is fuel fabrication technology. The fuel fabrication element involves the laboratory scale and later the production scale manufacture of coated-particle fuel that is intended to meet fuel performance requirements. It includes process development for kernels, coatings, and compacting; quality control (QC) methods development; and process documentation needed for technology transfer. The development activities include, for example, fuel process studies to understand how coating conditions are related to coating layer properties and how layer properties effect fuel particle performance during irradiation. It is expected that the DOE fuel fabrication technology research activities will provide NRC with significant insights, information and knowledge in the area of TRISO fuel manufacture. As a publically funded program, information developed by the DOE would be available to the NRC at no cost.

VII.11 Industry and University Research

GA has an ongoing 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, and would be of interest to the NRC.

As mentioned earlier, NRC has signed a cooperative agreement with MIT through which NRC will have access to state-of-the-art research results and to scientific personnel involved in cutting-edge research. 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 United States universities, national laboratories, and industries, this research involves international collaborations with Germany, Russia, China, Japan, and South Africa.