ADVANCED REACTOR RESEARCH

INFRASTRUCTURE DEVELOPMENT

ASSESSMENT

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EXECUTIVE SUMMARY

On April 18, 2003, the staff of the U.S. Nuclear Regulatory Commission (NRC) forwarded to the Commission SECY-03-0059, "NRC's Advanced Reactor Research Program." This document provided the staff's assessment of the U.S. Nuclear Regulatory Commission's (NRC's) advanced reactor technical infrastructure development and safety research that would be needed to support the review of new and advanced reactor applications planned and envisioned at that time. However, since the issuance of SECY-03-0059, the potential advanced reactor design applications that may be submitted for NRC review and approval have changed significantly. Such applications include (1) a potential license application for a very high temperature reactor (VHTR), which may be constructed at the Idaho National Laboratory in connection with the Next Generation Nuclear Plant (NGNP) Project, as directed by the Energy Policy Act of 2005, (2) a potential license application for the Toshiba Super Safe, Small and Simple (4S) liquid metal-cooled fast reactor which may be located near Galena, Alaska, (3) a potential design certification application for the Pebble Bed Modular Reactor (PBMR), (4) a potential license application for a commercial advanced fast burner reactor to be used for nuclear fuel recycling as part of the Global Nuclear Energy Partnership (GNEP) initiative; and (5) a potential license application for the High Temperature Teaching and Test Reactor (HTTTR) which may be sited near the University of Texas of the Permian Basin. In addition, several of the technical issues and research needs identified in SECY-03-0059 subsequently became part of the research and development program plans of either foreign or domestic advanced reactor design, development, or research organizations. Finally, the NRC implemented selected safety research and development in some areas of several of the higher priority HTGR-specific and generic technical arenas documented in SECY 03-0059 and achieved and documented results.

Because of the changes in the potential advanced reactor applications and the external advanced reactor research and development situation, as well as the progress made in implementing the higher priority NRC safety research and development planned earlier, the staff conducted a comprehensive reassessment of NRC's advanced reactor infrastructure development needs and updated its associated research and development program plans. The update focused on (1) advanced non-light-water reactor designs involving high (or very high) temperatures, graphite-moderated, gas-cooled thermal reactor technology and (2) liquid metal-cooled fast reactor technology. Potential reactor design applications in the first category include the NGNP VHTR, the PBMR, the Gas Turbine-Modular Helium Reactor (GT-MHR), and the HTTTR. Potential reactor designs in the second category include the Toshiba 4S reactor and the GNEP advanced burner reactor (ABR).

The technical infrastructure reassessment for high-temperature gas-cooled reactors (HTGRs) rebaselines and updates the earlier infrastructure assessment and information documented in SECY-03-0059. It considers foreign and domestic safety research and development activities that have been planned or implemented since SECY-03-0059 was issued. The reassessment also includes generic technical infrastructure development and safety research arenas (e.g., human performance and advanced instrumentation and controls) that are applicable to HTGRs, liquid metal reactors (LMRs), and advanced light-water reactors (LWRs). For HTGR

technology, the staff conducted the reassessment in sufficient depth and detail to identify the safety research and development that the NRC would need to conduct to support the review of an HTGR application.

For LMR designs and technology, the staff conducted a technical infrastructure survey at a higher level than the HTGR infrastructure reassessment. This survey identifies the key LMR technology safety and technical issues and infrastructure research and development needs. The survey provides a framework and starting point for the scope and direction of a later followup in-depth LMR technology infrastructure assessment. When conducted, the LMR infrastructure assessment will identify the safety research and development that the NRC would need to conduct to support the review of a LMR application.

The scope of the current reassessment does not include the technical infrastructure development and safety research that may be needed to support the review of new LWR (e.g., AREVA EPR, GE ESBWR, and the Westinghouse reactor IRIS reactor, Mitsubishi Heavy Industries US-APWR) applications. The staff will document these needs separately, as needed, on an LWR-specific basis.

The NRC will assign priorities to resources for technical infrastructure development and safety research to support HTGR design reviews, as well as to resources for generic technical arenas. Technical infrastructure development to support the NRC safety review of these designs will involve the development of staff expertise, analytic tools and methods, experimental facilities, and data. In the near-term, the staff expects the highest priority HTGR-specific technical infrastructure development and safety research to be in the areas of materials performance, fuel performance, nuclear and thermal fluid analysis, and accident source term analysis.

The HTGR infrastructure assessment and LMR infrastructure survey identify, respectively, the gaps in the NRC's independent technical capabilities and the NRC's reactor safety research for HTGRs and LMRs. Neither the 2003 assessment, described in SECY-03-0059, nor the current reassessment delineates the research that the NRC will conduct independently. Rather, they identify gaps in the NRC's information and capabilities in terms of required expertise, analytic tools, methods, and data. In this regard, applicants have the primary responsibility to demonstrate safety and to conduct the research needed to support both the plant design and development and the technical basis for the safety analysis. To a large extent, research conducted by the applicant will significantly reduce the research that the NRC will need to conduct. The NRC can and will obtain information through domestic and international cooperation, as well as through research and development conducted by the designers and developers of the new plants. Accordingly, in prioritizing and budgeting of resources for research and development activities, the NRC will consider obtaining information from other sources, giving due consideration to its responsibility as an independent regulatory agency.

The staff conducted the HTGR infrastructure reassessment in a generic manner so that it applies to the range of HTGR plant designs currently being developed for potential NRC safety review. These include the very high temperature gas-cooled reactor design for the VHTR (which is considered the leading design concept for the NGNP reactor), VHTR versions of the

PBMR, the General Atomics GT-MHR and the AREVA ANTARES HTGR. The PBMR and GT-MHR HTGR designs are also being considered for potential deployment as commercial electric power generating facilities within the United States. The concepts considered in the LMR infrastructure survey include the Toshiba 4S reactor and the GENEP ABR. The General Electric Power Reactor Innovative Small Module (PRISM) reactor design is considered among the potential candidates for the latter and was included in the basis for the survey.

The HTGR infrastructure technical needs assessment arenas and activities are linked to nine key research areas:

(1) Technical review infrastructure (including the potential development of HTGR-specific risk-informed, performance-based regulatory guidance and standard review plans)

(2) accident analysis (including probabilistic risk assessment methods and assessment guidance, human factors, and instrumentation and control)

(3) reactor/plant systems analysis (including thermal-fluid dynamics, nuclear analysis, and accident fission product transport and source term analysis)

(4) fuels performance analysis

(5) materials analysis (including nuclear graphite and metallic component performance)

(6) structural analysis (including reactor civil structure and reactor core internals structural performance) and reactor safety hazards posed by a connected hydrogen production facility

(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)

(9) nuclear safeguards and security

Human factors and instrumentation and controls are considered generic arenas applicable to all reactor designs and technologies. The LMR infrastructure survey addressed reactor/plant systems analysis (including thermal-fluid dynamics, nuclear analysis, and severe accident and source term analysis), fuels analysis, materials analysis, and structural analysis.

CONTENTS

FIGURES

TABLES

ABBREVIATIONS

ABR	advanced burner reactor
ACS	auxiliary cooling system
ADAMS	Agencywide Documents Access and Management System
AGR	advanced gas-cooled reactor
ALARA	as low as reasonably achievable
ALMR	advanced liquid metal reactor
AMPX	A Modular Code System for Processing X-sections
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ASTRA	Advanced Gas Reactor in Kurchatov Institute, Russia
ATLAS	Advanced Thermal Analysis Software
ATWS	anticipated transient without scram
AVR	Arbeitsgemeinschaft Versuchsreaktor
BDBE	beyond design-basis event
C	Celsius
CANDU	Canadian deuterium uranium reactor
C-C	carbon-carbon
CESAR	a critical experiment facility in Grenoble, France
CFD	computation fluid dynamics
CFP	coated fuel particle
CFR	*Code of Federal Regulations*
CONTAIN	containment and computational code
CORSOR	core source term release code
cm	centimeter
CRP	coordinated research project
DBE	design-basis event
DHS	U.S. Department of Homeland Security
DOE	U.S. Department of Energy
E.C.	European CommissionEM electromagnetic
ENDF/B	Evaluated Nuclear Data File, Volume B
EPZ	emergency planning zone
EURATOM	European Atomic Energy Community
eV	
F	Farenheit
FLUENT	computational fluid dynamic code made by FLUENT, Inc.
FOM	figure of merit
FP	fission product
GA	General Atomics
GCR	gas-cooled reactor
GNEP	Global Nuclear Energy Partnership
GRSAC	Graphite Reactor Severe Accident Code
GT-MHR	gas turbine-modular helium reactor
GWd/t	gigawatt days per ton
HCLPF	high confidence in low probability of failure
HFE	human factors engineering human failure event
HFR	high flux reactor
HLW	high-level waste
HPB	helium pressure boundary
HRA	human reliability analysis
HSI	human-system interface
HTGR	high-temperature gas-cooled reactor
HTR	high-temperature reactor

HTR-10	10-megawatt electric (MWe) high-temperature reactor
HTR-F	high-temperature gas-cooled reactor fuel technology
HTR-N	
HTTR	high-temperature test reactor
HTTTR	high temperature teaching and test reactor
I&C	instrumentation and control
IAEA	International Atomic Energy Agency
IFCI	
IHTS	intermediate heat transport system
IHX	intermediate heat exchanger
INEEL	Idaho National Engineering and Environmental Laboratory
INL	Idaho National Laboratory
INET	Institute of Nuclear and New Energy Technology
IRIS	International Reactor Innovative and Secure
IRPhEP	International Reactor Physics Experiment Evaluation Project
ISI	inservice inspection
JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute
KAHTR	a critical experiment facility in the Jülich Research Center
k_{eff}	
kg/s	kilogram per second
LEU	low-enrichment uranium
LLW	low-level waste
LMR	liquid metal reactor
LOFC	loss of forced circulation loss of forced cooling
LWR	light-water reactor
m	meter
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
MONK	
MORECA	code that simulates MHTGR accidents developed by ORNL
MPa	
MWD/MTU	
MVP	
MWd/t	megawatt days per ton
MWt	megawatt thermal
NACOK	Natural Conversion of Air through the Core with Corrosion (test facility at Jülich Research Center)
NDE	nondestructive examination
NEA	Nuclear Energy Agency
NERI	Nuclear Energy Research Initiatives
NEWT	ORNL computer code
NGNP	Next Generation Nuclear Plant
NJOY	Nuclear Data Processing System from Los Alamos National Laboratory
NMSS	Office of Nuclear Material Safety and Safeguards
NPP	nuclear power plant

NRC	U.S. Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development
ORECA	gas-cooled reactor accident code developed by ORNL
ORNL	Oak Ridge National Laboratory
RC	reinforced concretePARCS Purdue Advanced Reactor Core Simulator
PBMR	Pebble Bed Modular Reactor
PBMR Pty.	Pebble Bed Modular Reactor Company, Pty. Ltd.
PBR	modular pebble bed reactor
PCS	power conversion system
PIE	postirradiation examination
PIRT	phenomena identification and ranking table
PMR	modular prismatic block reactor
PRA	probabilistic risk assessment
PRISM	Power reactor Innovative Small Module
PROTEUS	High-Temperature Reactor Configuration of the Proteus Critical Experimental Facility in Switzerland
PVRC	Pressure Vessel Research Council
PyC	pyrolitic carbon
R&D	research and development
RAPHAEL	Reactor for Process Heat and Electricity
RCCS	reactor cavity cooling system
RES	Office of Nuclear Regulatory Research
RVACS	reactor vessel auxiliary cooling system
SANA	Self-Acting Removal of the After-Heat of the HTR Modular Reactor (test facility at the Jülich Research Center)
SAPHIRE	Systems Analysis Program for Hands-on Integrated Reliability Evaluation
SCALE	Shielding and Criticality Analysis for Licensing Evaluation
SCALE/CE-KENO	
SCC	stress corrosion cracking
SCDAP/RELAP5	
SiC	silicon carbide
SRP	standard review plan
SSC	structure, system, and component
TEXAS	
T/F	thermal-fluid
T/H	thermal-hydraulic
THTR	Thorium Hochtemperaturreaktor
TNT	trinitrotoluene
TRISO	tri-isotropic
TRITON	Transport Rigor Implemented with Time-Dependent Operation for Neutronic Depletion
UIS	upper internal structure
ULOF	unprotected loss of flow
USS	ultimate shutdown system
VHTR	very high temperature reactor
VHTRC	Japanese test facility
VICTORIA	radionuclide transport and decommissioning codes
VSOP	Very Superior Old Programs (a German code)
wt%	weight percent
ZrC	zirconium carbide

4S	Super Safe, Small and Simple

I. INTRODUCTION

I.1 Background

On April 18, 2003, the staff of the U.S. Nuclear Regulatory Commission (NRC) forwarded to the Commission SECY-03-0059, "NRC's Advanced Reactor Research Program." This document provided the staff's assessment of NRC's advanced reactor technical infrastructure development and safety research that would be needed to support the review of new and advanced reactor applications planned and envisioned at that time. However, since the issuance of SECY-03-0059, the potential advanced reactor design applications that may be submitted for NRC review and approval have changed significantly. These applications now include (1) a potential license application for a very high temperature reactor (VHTR), which may be constructed at the Idaho National Laboratory in connection with the Next Generation Nuclear Plant (NGNP) Project, as directed by the Energy Policy Act of 2005, (2) a potential license application for the Toshiba 4S (super safe, small and simple) fast liquid metal-cooled reactor, which may be located near Galena, Alaska, (3) a potential design certification application for the Pebble Bed Modular Reactor (PBMR), (4) a potential license application for a commercial advanced burner reactor (i.e., an LMR) to be used for nuclear fuel recycling, as part of the Global Nuclear Energy Partnership (GNEP) initiative, and (5) a potential license application for the High Temperature Teaching and Test Reactor (HTTTR), which may be sited near the University of Texas of the Permian Basin. In addition, several of the technical issues and research needs identified in SECY-03-0059 subsequently became part of the research and development (R&D) program plans of either foreign or domestic advanced reactor design, development, or research organizations. Finally, the NRC implemented selected safety research and development in some areas of several of the higher priority HTGR-specific and generic technical arenas documented in SECY 03-0059, and achieved and documented results.

Because of the changes in the potential advanced reactor applications and the external advanced reactor R&D situation, as well as the progress made in implementing the higher priority NRC safety R&D planned earlier, the staff conducted a comprehensive reassessment of NRC's advanced reactor infrastructure development needs and updated its associated proposed safety research and development plans. The update focused on (1) advanced non-light-water reactor designs involving high (or very high) temperature, graphite-moderated, and gas-cooled thermal reactor technology and (2) fast liquid metal-cooled reactor technology. Potential reactor design applications in the first category include the NGNP VHTR, the PBMR, the Gas Turbine-Modular Helium Reactor (GT-MHR), the ATARES reactor, and the HTTTR. Potential reactor designs in the second category include the Toshiba 4S reactor and the GNEP advanced burner reactor (ABR), including the PRISM reactor.

The technical infrastructure reassessment for high-temperature gas-cooled reactors (HTGRs) rebaselines and updates the earlier infrastructure assessment and information documented in SECY-03-0059. It includes consideration of foreign and domestic safety R&D activities that have been planned or implemented since SECY-03-0059 was issued. The reassessment also includes generic technical infrastructure development and safety research arenas (e.g., human performance, advanced instrumentation and controls (I&C)) that are applicable to HTGRs, liquid metal reactors (LMRs), and light-water reactors (LWRs). For the HTGR technology arena, the staff conducted the reassessment in sufficient depth and detail to identify the safety R&D needed to support the review of a potential HTGR application.

For LMR design and technology, the staff conducted a technical infrastructure survey at a higher level than the HTGR reassessment. This survey identifies the key LMR technology safety and technical issues and infrastructure safety research and development needs. The survey also provides a framework for the scope and direction of a followup in-depth LMR technology infrastructure assessment. When conducted, the LMR infrastructure assessment will identify proposed NRC safety R&D needed to support the review of a LMR application

The HTGR infrastructure assessment and LMR infrastructure survey identify the gaps in the NRC's independent technical capabilities and the NRC's reactor safety research for HTGRs and LMRs, respectively.

II. ROLE OF NRC RESEARCH

The applicant and designer are responsible for demonstrating the safe performance of their proposed plant design and use of new reactor technologies. It is expected that the plant designer for the application will conduct research to support the technical basis for the application and the plant safety analysis. In this regard, it is expected that research will be conducted by the applicant and designer to: demonstrate sufficient margins to safety significant SSC design and safety limits, search for and identify, assess and resolve, significant reactor and plant safety issues involving large uncertainties; develop, verify and validate the proposed safety analysis evaluation methods; provide the technical basis for requirements, criteria, codes or standards that are proposed for the design basis for plant licensing; understand and quantify the failure thresholds for safety significant SSCs; examine and analyze "what if" questions that are needed to establish requirements and analyze plant safety performance for events and conditions beyond the design-basis and; support NRC regulatory and licensing decisions

The NRC will conduct research necessary to help support the licensing review. The agency's research will focus on:

- Developing adequate staff technical knowledge, expertise and capabilities to independently review and effectively evaluate the acceptability of the application, including the safety analysis and the technical basis for the safety analysis.

- Independently confirming the technical basis for requirements and criteria needed for plant licensing and, the regulatory guides and standard review plans needed for developing an acceptable application and an effective and efficient staff review,

- Developing an independent analytical capability to confirm: safety analysis evaluation methods and safety analysis results and; the adequacy of proposed resolutions of safety issues and/or the development of the technical basis for staff proposed reactor safety enhancements,

- Adequately confirming or interpreting existing technical information for which there is significant uncertainty or, adequately validate and scope-out technical issues involving significant safety or risk implications to justify the need for followup resolution by the applicant.

The NRC research will not duplicate research conducted by the applicant or designer. The staff will generally refer research needed to resolve issues identified by the NRC's regulatory offices as a result of their reviews to the applicant or design organization.

While assessing challenges posed by advanced reactor designs and technologies, the staff will consider what research the applicant or designer has conducted or will conduct as part of the application, as well as what additional research will be needed to support the reviews conducted by the NRC's regulatory and licensing offices. The general principle to be used for funding a specific research activity is that the applicant is responsible for any data that are needed to support regulatory decisions on safety cases for a particular reactor design. In this context, if the NRC believes it is important to explore issues involving uncertainties, the applicant and designer will conduct the research. However, if it is necessary to develop capabilities to independently check licensee results, the NRC will use its own resources.

The NRC will pursue potential cooperative research to support its regulatory review needs. In this regard, when both the NRC and the nuclear industry organization benefit from such research, or if it is difficult to determine whether industry or the NRC is the principal beneficiary, and/or NRC's specific additional research needs can be effectively and efficiently addressed within the larger scope of the industry research plans, the NRC may enter into joint partial funding of the planned research. However, it is essential that the NRC's independence be maintained in the planning and implementation process, that the quality and integrity of the data be maintained, and that all legal and administrative requirements be met to ensure that the NRC remains independent in regulatory decisionmaking. These guidelines also apply to NRC's relationships with other Government agencies such as the U.S. Department of Energy (DOE) and the national laboratories that support DOE in conducting the R&D needed to support the licensing of advanced reactor concepts involving innovative designs and new technologies.

While the applicant and design organizations, including DOE and its national laboratories, conduct research on advanced reactor designs, NRC research focuses on ensuring an adequate technical basis for the safety requirements, criteria, and standards that these new designs must meet. However, an applicant, designer, or DOE may propose specific safety requirements, criteria, and standards and conduct the research needed to support the NRC's review of such proposals. In such cases, the NRC may need additional research beyond that conducted by DOE or by the applicant. The designer, the applicant, or the NRC may fund research needed to establish acceptance criteria associated with a new safety standard or requirement or to address specific issues for a particular reactor design. Alternatively, the research may be jointly funded by the NRC in cooperation with the designer or applicant.

Others with a vested interest may also conduct research (e.g., the generic and technology-neutral research sponsored by DOE or industry-supported organizations). Experience with advanced LWR reviews, indicates that the scope, schedule, and resources for such research programs are extensive and that the NRC staff can benefit from worldwide developmental research and experience. Mindful of the appropriate roles and consistent with the NRC Strategic Plan, the agency 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, and international nuclear organizations such as the Nuclear Installations Inspectorate, the Nuclear Energy Agency (NEA), and the European Union. In addition to offsetting costs, sharing research facilities and leveraging resources to minimize duplication can achieve significant efficiencies. Ensuring that the regulatory process does not impede the use of new technology to improve safety or reduce costs is an important part of the NRC's Strategic Plan.

In general, NRC research infrastructure needs center on the development of expertise, tools, and methods that support the agency's mission by identifying, understanding, and resolving potential safety issues and establishing requirements for advanced reactor designs. The development of NRC's 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 require modifications. The NRC will also need new codes to independently evaluate the safety performance of HTGR or VHTR 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 staff focuses primarily on areas where important gaps exist (e.g., in technological knowledge, in understanding risk-significant uncertainties, or in characterizing and understanding the degree of conservatism in safety margins). Computer models validated by experiments are important tools to bridge gaps in technology. 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, the agency uses preapplication reviews to identify the necessary new (or modifications of existing) codes and standards early in the process.

Traditionally, the NRC performs two types of research in support of the regulatory process. These are (1) research to support the technical basis for regulatory decisionmaking and (2) research necessary to scope out uncertainties involving significant safety or risk implications so as to gain insight into safety margins and failure thresholds. 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 research products support safety evaluation reports or guidance in the form of regulatory guides, Standard Review Plan (SRP) sections, or NUREG reports.

However, even a well-funded and focused program of nuclear safety research cannot transform the regulation of advanced nuclear power plants (NPPs) into a process in which decisions flow exclusively from scientific and technical knowledge. The NRC will need to consider defense-in-depth and safety margins to offset limitations in state-of-the-art knowledge and understanding. Similar to other complex technologies, advanced reactor regulation will require a complex blend of technical knowledge applied within the context of Commission policy and prudent regulatory decisions. Therefore, in assigning program priorities, the NRC 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 emerge. 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.

Reviews of advanced reactor designs and research findings may raise a novel set of questions. Answering these questions by examining their 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 (FP) 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. The NRC will conduct routine peer reviews of the research products and anticipated schedules for specific research activities to instill confidence in the scope and quality of the research; these reviews will include frequent interactions with the Advisory Committee on Reactor Safeguards and the Advisory Committee on Nuclear Waste to obtain feedback, guidance, and involvement from the Office of Nuclear Reactor Regulation and the Office of Nuclear Material Safety and Safeguards (NMSS).

II.1 Objectives and Structure

The infrastructure reassessment and infrastructure survey focus, respectively, on HTGR and LMR designs and technologies. Generic infrastructure development needs that apply to HTGRs, LMRs, and new LWR designs are also included. The primary objective of the HTGR infrastructure reassessment is to identify the key technical issues associated with an HTGR application and safety analysis that will require safety R&D to resolve. For those HTGR design and technology aspects and technical aspects with generic applicability to both HTGRs and LMRs, the reassessment describes the technical issues in sufficient depth and detail to identify the specific proposed NRC safety R&D. For LMR design and technology aspects, the objective is to provide an initial limited-depth survey of the key safety and technical issues and research needs associated with LMRs which will facilitate the efficient conduct of a more detailed and indepth follow-on infrastructure assessment later. The follow-on LMR assessment would be of sufficient depth and detail to enable the proposed LMR technology-specific NRC safety R&D to be comprehensively identified at a level similar to that of the HTGR infrastructure assessment.

Within this context, for those technical aspects either involving HTGR design and technology or having generic applicability to both HTGRs and LMRs, the information presented identifies the following:

- technical and safety issues and pathways to methods and tools to resolve these issues

- the flow of information among the various technical areas

In updating the NRC advanced reactor infrastructure assessment, the staff drew from many sources of information. These included information on the status of NRC's previously planned advanced reactor safety research; on advanced reactor safety research planned or implemented by national or foreign organizations, stored within NRC's HTGR knowledge management center and obtained from international HTGR technical conferences and workshops; obtained as a result of HTGR preapplication reviews; obtained from organizations currently pursuing research in connection with HTGR design and development activities; obtained from DOE on research activities being planned or implemented in connection with DOE's NGNP VHTR and LMR development programs; and included in earlier NRC preapplication reviews of LMR designs. The update included the relative priorities previously assigned to research areas and opportunities for international cooperative research.

For HTGRs, the staff also took advantage of the insights and information from the Exelon-sponsored PBMR preapplication review, the General Atomics (GA) GT-MHR preapplication review, and the more recent PBMR preapplication review sponsored by Pebble Bed Modular Reactor, Pty. Ltd. (PBMR Pty). The staff considered technical information from the High Temperature Reactor International Conferences of 2002, 2004 and 2006, as well as information obtained through staff participation in recent research coordination meetings for IAEA Coordinated Research Projects (CRPs) for high-temperature reactor (HTR) fuels (CRP-6) and HTR accident analysis codes (CRP-5) and in the American Society for Testing and Materials committee to develop a nuclear graphite standard. The staff also considered NGNP preliminary reactor design information and DOE-sponsored NGNP R&D program plans and other supporting documents. Information on the GNEP ABR preliminary reactor design, the 4S reactor design and PRISM design were also considered. In its assessment, the staff included R&D program information provided by HTGR technology design, development, and research organizations; generic and HTGR-specific R&D documents published by the NRC and other organizations; and technical issues identified by completed HTGR technology R&D. Finally, the staff also considered advances in computational analysis capabilities, codes, and methods; codes and standards development activities in support of potential HTGR applications; and historical NRC LMR review documents (e.g., the Clinch River Breeder Reactor, PRISM).

To facilitate the identification of important research areas for HTGRs and LMRs, the staff used a top-down approach as shown in Figure 1.

Nuclear Reactor Safety Arena

For the update, the staff aligned the research areas and activities with the four cornerstones of reactor safety:

(1) accident prevention
(2) accident mitigation
(3) barrier protection
(4) offsite protection

Figure 1 shows the alignment and identifies the associated key research areas. Activities linked to these areas include the following:

Key Research Area	Activities
Requirements/Guidance Development	Development of risk-informed and performance-based review guidance
Accident Analysis	Probabilistic risk assessment (PRA), human factors, and I&C
Reactor/Plant Analysis	Thermal-fluid (T/F) analysis, nuclear analysis, FP transport, and source term analysis
Fuels Analysis	Fuel performance and FP transport analysis
Materials Analysis	Metallic, composite, and graphite component performance analysis
Structural Analysis	Civil structure and reactor internal performance and analysis for external challenges, including coupled industrial facility hazards analysis methods
Consequence Analysis	Dose assessment and environmental impact studies
Security/Safeguards Analysis	Security assessment methods

As in the earlier assessment, research products resulting from these activities generally either support a technical basis for resolving specific safety issues or support another research area.

Nuclear Materials Safety and Nuclear Waste Safety Arenas

Advanced HTGR safety research activities for nuclear materials safety and nuclear waste safety focus on supporting regulatory activities at the front end and back end of the nuclear fuel cycle:

- Front end—uranium enrichment, fuel fabrication, transportation, and storage
- Back end—storage, transportation, and disposal of spent fuel and low-level waste

Section III.3 of this report discusses infrastructure development and safety research needs associated with these arenas.

Safeguards and Security Arena

Infrastructure development and safety research needs for safeguards and security will support other regulatory offices, principally the Office of Nuclear Security and Incident Response. Research areas include proliferation issues, evaluation of security measures, and material control and accounting (MC&A) systems needed to prevent and detect nuclear material diversion over the entire fuel cycle. Section III.4 of this report briefly discusses safety research to support these regulatory aspects.

The Office of Nuclear Regulatory Research (RES) will support the Office of Nuclear Security and Incident Response and other NRC offices and agencies with information needed for their assessments in these areas. This coordinated research support will respond to any new issues emerging from Government-wide initiatives on homeland security.

Advanced Reactor Research Infrastructure
Key Research Areas and Areas for Examination

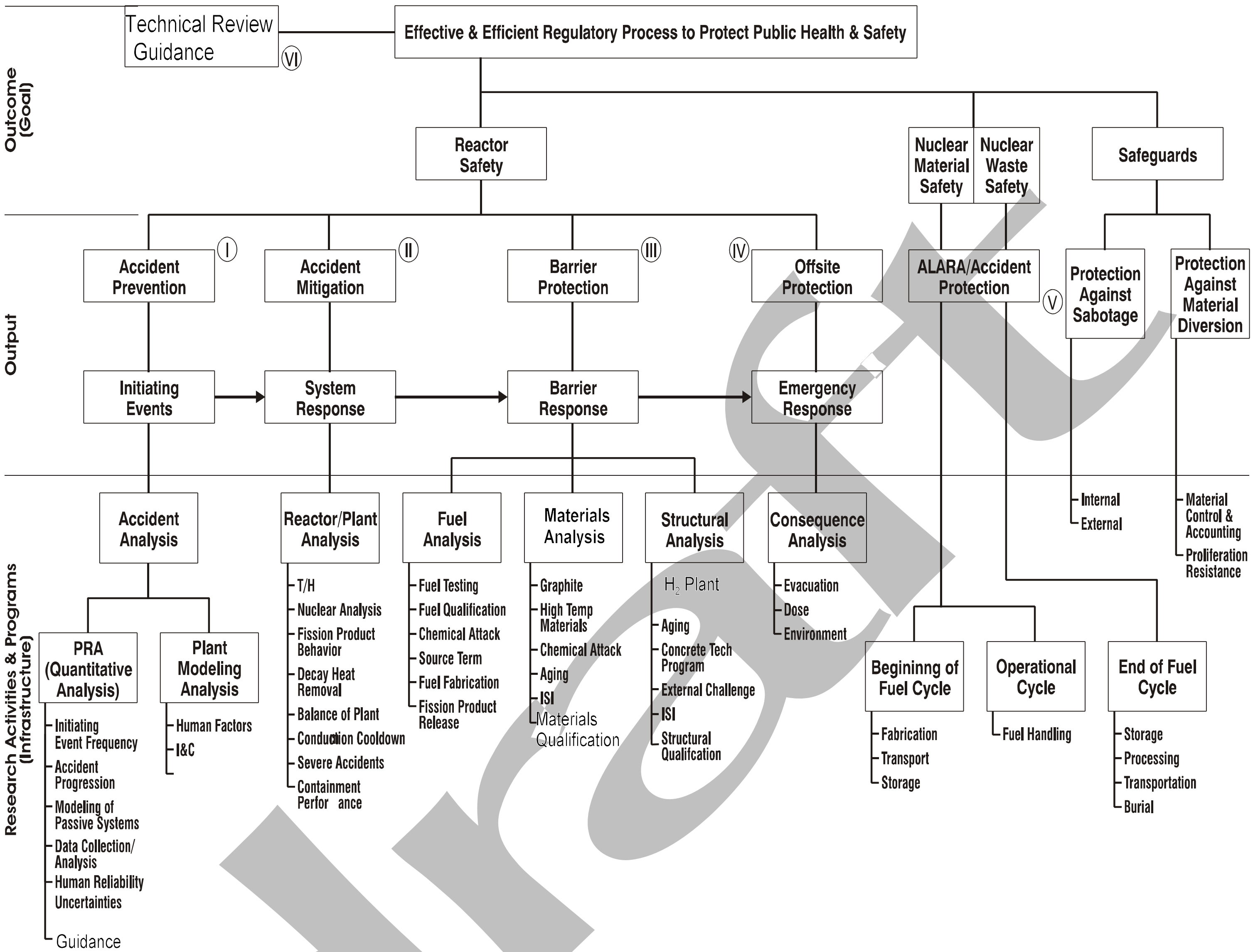

Figure 1 Key Research Areas for Examination

Figure 1 Key Research Areas for Examination

III. KEY INFRASTRUCTURE DEVELOPMENT AREAS AND ACTIVITIES

III.1 Technical Review Infrastructure Development

The staff has devised and implemented a plan to develop a regulatory structure for future plant licensing. The objective of the structure is to enhance the effectiveness and efficiency of future plant licensing in the longer term. As originally envisioned, a goal of the regulatory structure (or framework) was to provide new a risk-informed, performance-based and technology neutral regulations for new plant licensing. The staff developed the framework in a technology-neutral manner so that it would accommodate different reactor technologies. The framework is risk informed to identify the more likely safety issues and gauge their significance and performance based to provide flexibility. The framework includes defense-in-depth provisions to address known and unknown uncertainties.

The staff also developed a technology-neutral framework/guideline for the regulatory structure which includes (1) a set of technology-neutral requirements and (2) guidance for applying the framework on a technology-specific basis.

The framework structure is a top-down approach to translating the mission of the Atomic Energy Act (protecting the public health and safety) into a set of technology-neutral (or technology-specific) requirements. The framework includes criteria and guidance for the following:

- safety, security, and preparedness expectations
- risk expectations
- licensing expectations
- treatment of uncertainties (defense-in-depth)
- performance-based concepts
- PRA technical quality

The framework includes future plant technical licensing policy issues that will need to be identified for Commission consideration and decision. The policy issues level of safety, treatment of integrated risk for multiple reactors at a single site, containment versus confinement, application of a plant PRA for the selection of licensing-basis events, including design-basis accidents and the selection of safety-related structures, systems, and components (SSCs), construction and use of a frequency-consequence limit curve among others. The staff has developed preliminary initial guidance for each of these issues, and stakeholders have provided extensive feedback.

On May 4, 2006, the Commission issued an Advance Notice of Proposed Rulemaking (ANPR) to consider the spectrum of issues related to risk-informing the reactor regulations. Specifically, in the ANPR, several policy and technical issues were raised related to development of a regulatory structure for future plant licensing. These issues included such items as level of safety, integrated risk, probabilistic approach for the licensing basis, safety classification, treatment of uncertainties (defense-in-depth), PRA technical quality. The staff will consider the stakeholder comments received as a result of the ANPR on the various technical issues in finalization of the regulatory structure for future plant licensing.

The staff is also currently conducting a limited scope preapplication review of a licensing approach being proposed for the PBMR by the reactor's designer-developer, PBMR Pty. The company has stated that it intends to pursue a design certification for the PBMR under 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants." The scope of the staff's PBMR preapplication review involves four technical white papers submitted by PBMR Pty. These white papers are entitled (1) "Probabilistic Risk Assessment Approach for the Pebble Bed Modular Reactor," (2) "Licensing Basis Event Selection for the Pebble Bed Modular Reactor," (3) "System Structure and Component Classification," and (4) "Defense-in-Depth." The PBMR white papers deal with many of the same technical and licensing policy issues that the staff has addressed in its work to develop the risk-informed and performance-based framework for future plant licensing. Although the approaches taken by the staff and PBMR are in many respects very similar, there are also important differences which will need to be resolved. The staff intends to use the completed technical work and insights achieved in developing the framework for future plant licensing and the stakeholder comments on the framework to inform its review of the proposed PBMR licensing approach and identify issues requiring resolution.

The technical review infrastructure development needs identified in the NGNP licensing strategy will, in the near term, inform the identification of needed support for the review of HTGR or VHTGR applications. This could involve aspects such as draft SRPs, draft regulatory guides, and Commission licensing policy decisions related to increased use of the plant PRA in developing the plant licensing basis. To the extent that the NGNP licensing approach in the NGNP licensing strategy allows greater use of probabilistic information for the selection of licensing-basis events and safety-related SSCs of the sort proposed by PBMR Pty. for licensing the PBMR, the staff will develop a specific R&D plan to develop the technical review infrastructure to support the staff's review of the applicant's proposed approach to making greater use of PRA. The staff's completed work on developing the technology-neutral framework, stakeholder comments on the framework, the PBMR preapplication review insights, and Commission policy decisions will inform the development of this infrastructure.

III.2 Reactor Plant Safety Analysis

III.2.1 Accident Analysis

III.2.1.1 Probabilistic Risk Assessment

III.2.1.1.1 Background

During the past 30 years, the NRC/Atomic Energy Commission has performed Probabilistic Risk Assessments (PRAs), and has promoted their use as a means of developing LWR plant risk perspectives and identifying improvements. As a result, the NRC has developed the capability to use PRAs in regulatory decisionmaking for current generation LWRs and to a large extent advanced LWRs.

Modular HTGR applicants will rely on PRAs as an integral part of their design development and license applications. Integrating the plant PRA into the design development process and the licensing process creates new challenges in the preparation and maintenance of PRAs. The expanded use of the PRA for licensing decisions requires greater completeness, defensibility and transparency of the PRA. Traditionally, the scope of LWR PRAs has been confined to the analysis of beyond design basis accidents (i.e., accidents that can lead to severe core damage in LWRs). However, for modular HTGRs, the scope of the PRA will include the full spectrum of off-normal events including frequent, infrequent and rare initiating events and event sequences. Additionally, the designers of modular HTGRs propose that the plant PRA be used as the primary input for the selection of the plant's licensing basis events, including abnormal operating occurrences (AOOs), design basis accidents (DBAs) and beyond design basis accidents (BDBAs). Modular HTGR PRAs are expected to involve event sequences which result in a spectrum of radiological release and event sequences that involve a spectrum of core damage states. The PRAs will also need to address the dose consequences of these event sequences as measured at the site boundary, LPZ boundaries and at one mile.

The scope of a modular HTGR PRA is also broader than that typically considered in today's PRAs. In addition to at-power and shutdown reactor operation, it also needs to be able to support the assessment of non-traditional events, for radionuclide sources at the plant outside of the core such as those associated with the onsite spent fuel storage tanks, the reactor coolant system fission product cleanup system and the online fueling system.

LWR licensing and regulatory decision needs provided the basis for existing PRA guidance, requirements and standards. Metrics such as core damage and large early release may not be applicable to certain advanced reactor designs such as modular HTGRs. The current set of PRA levels that are used for LWR regulatory and licensing decision addresses event sequences leading to core damage, containment response and public-health consequences. These PRA levels are specific to LWRs and are expected to be not generally applicable to advanced reactor designs such as HTGRs. Therefore, in addition to issues associated with the role of the PRA, the applicability of the available guidance needs to be assessed and updated to reflect the application of plant PRAs in the licensing and regulatory framework.

Consideration of uncertainties is vital to understanding risk. Uncertainties will need to be addressed in the calculation of both the frequency and the consequence of event sequences and in the evaluation of event sequences against the frequency-consequence limit curve that has been proposed for modular HTGR licensing. Therefore, an important aspect of the licensing approach proposed for modular HTGRs, will involve identifying, evaluating, and addressing uncertainties.

Advanced reactor designs, such as modular HTGRs, will make extensive use of passive systems, structures and components and rely on inherent characteristics of the design to ensure safety, and place little, reliance. If any, on the use of active systems. As a result, potential errors and omissions during design, manufacture, fabrication, construction or testing of plant SSCs could adversely impact plant safety performance. Undetected latent errors and failures are particularly important for advanced designs, which are expected to place greater reliance on factory fabrication (as opposed to field fabrication) for SSCs.

III.2.1.1.2 Purpose

The purpose of PRA infrastructure development would be to develop the regulatory guidance (and standard review plan) needed to provide an acceptable approach for evaluating whether a proposed plant PRA is adequate to be used in making the licensing decisions described in Section III.2.1.1.1. The scope and detail of the guidance will need to be sufficient for the staff's PRA review (1) to be focused and consistent in determining the technical quality of the PRA, and (2) to allow the staff to understand the strengths and potential weaknesses in plant in terms of its design and operational safety aspects. In conjunction with this guidance, development of the necessary standards, associated detailed technical guidance, and data will be needed, particularly in such areas as passive system performance, digital instrumentation and control, event initiation and accident progression. This development may require new tools and methods to be developed.

III.2.1.1.3 Objectives and Planned Activities

A good understanding of how the PRA is to be used in the for advanced reactor licensing, specifically with regard to the design, construction, operation and maintenance is needed. Such an understanding will provide input to the needed guidance and the potential need for new methods, tools, and data to be developed.

The plant PRA that will be provided in an application is expected to be developed during the design phase and evolves as the design matures. That is, the PRA, as it is developed, will influence the plant design development decisions and as the plant design development decisions are made, the plant PRA is updated. Fabrication and construction errors associated with passive systems can invalidate safety analysis and PRA assumptions. Therefore, the identification of adverse latent conditions that could occur during fabrication and construction will be critical to ensuring safety. Accordingly, risk-informed inspection procedures will be needed to help focus inspections to enhance the identification of such conditions. In addition, construction changes need to be assessed with respect to their effect on the plant PRA and on the level of plant safety. On completion of the design, construction and startup phases, the updated PRA reflecting the final design, construction and testing experience will be used to support licensing and regulatory decisions including decisions associated with plant operations. It is expected that, because the PRA for advanced reactors will provide such an important role in licensing decisions, it will need to be maintained and updated to support regulatory oversight an regulatory decisions during the plant operating lifetime. In addition, it is expected that a risk-informed philosophy will be integrated into the operation of the plant at a greater level than that of the current plants.

Specifically, for advanced reactors, such as modular HTGRs, PRA guidance will be needed to support licensing as it relates to design, construction, operation and maintenance in the following functional areas:

- Technical Requirements
- Quality Assurance Criteria
- Consensus Standards
- Assumptions and Inputs
- Analytical Methods
- Analytical Tools
- Independent Peer Review
- Documentation
- Configuration Control

The required scope of the PRA and the corresponding technical requirements for each technical element will be identified. Specifically, high-level requirements are provided for all the technical elements of a PRA required to calculate the frequency of accidents, the magnitude of radionuclide release, and the resulting consequences. Requirements are also provided for the scope of the PRA which is defined by identification of the complete set of challenges including both internal and external events during all modes of operation.

Given the integration of the PRA into the licensing process, completeness, defensibility and transparency will be needed. This will require more rigorous controls directed at the completeness and the quality of the PRA than for the PRAs associated with current operating LWRs.

One acceptable approach to demonstrate conformance with the regulatory position is to use a national consensus PRA standard or standards that address the scope of the PRA used in the decision making. The PRA standard must be applicable to the design of the plant.

Assumptions used in the PRA need to be realistic and defensible with their basis and application clearly documented. PRAs should not be overly conservative or make optimistic assumptions and should not rely on expert judgement except in situations where there is a lack of available technical information regarding the condition or response within the PRA, or a lack of analytical methods upon which to base a prediction of a condition or response. The PRA should take credit for SSCs to the extent by their deterministic design analyses and should take credit for human actions based on their probability of success. Guidance will also be needed to review the technical basis and calculation of PRA uncertainties, including an assessment of the uncertainty of the results for important or key assumptions both individually and in logical combinations.

The analytical methods used in a PRA need to be sufficiently detailed as to purpose, method, assumptions, design input, references and units such that a person technically qualified in the subject can review and understand the analysis and verify the adequacy of the results without recourse to the originator. Where possible, analytical methods need to be consistent with available codes and standards and checked for reasonableness and acceptability. Method-specific limitations and features that could impact the results need to be identified.

PRA quantification software, thermal/hydraulic codes, structural codes, radionuclide transport codes, human reliability models, common cause models, etc., are typically used in the PRA quantification process. These models and codes shall have sufficient capability to model the conditions of interest and provide results representative of the plant and need to be used only within known limits of applicability. As errors in such programs may significantly impact the results, it is necessary that the development and application of the computer programs, spreadsheets or other calculation methods exhibit a high level of reliability as ensured through a documented verification and validation process. Verification is a systematic approach to ensure the model or computer code correctly represents the model or code’s design. Validation is the demonstration that the verified models or codes meet the requirements.

A PRA needs to be peer reviewed. An adequate peer review is one that is performed by qualified personnel, according to an established process that compares the PRA against the characteristics and attributes, documents the results, and identifies both strengths and weaknesses of the PRA.

A PRA needs to be documented such that a person technically qualified in PRA can review and understand the analyses and verify the adequacy of the results without recourse to the originator. The documentation needs to be traceable and defensible with sources of information both referenced and retrievable. It needs to support the determination that the PRA is performed consistent with the applicable standards and the technical requirements contained within the framework and its implementing requirements. The documentation also needs to be maintained current with the plant configuration and the PRA model. The methodology used to perform each aspect of the work needs to be described either through documenting the actual process or through reference to existing methodology documents. Key sources of uncertainty need to be identified and their impact on the results assessed. Key assumptions made in performing the analyses need to be identified and documented along with their justification to the extent that the context of the assumption is understood. The results (e.g., products and outcomes) from the various analyses need to be documented. This documentation entails both submittal and archival documentation.

The PRA needs to be maintained throughout the construction, and operation phases of the plant. Therefore, a PRA configuration control program should be developed early in the design process and needs to be in place at the time the PRA is submitted for NRC staff review.

III.2.1.1.4 Application of Research Results

The results of the above efforts will provide the necessary information to develop the needed regulatory guidance for an acceptable PRA as it is used to support the design, construction, and operation of the plant. The guidance will provide the needed information to support the PRA in the following activities:

- Generate a Complete Set of Accident Sequences
- Develop a Rigorous Accounting of Uncertainties
- Evaluation of the PRA Results Against Quantitative Acceptance Criteria
- PRA Supported Assessment of Security
- Identification and Characterization of the Licensing Bases Events
- Identification and Characterization of the Special Treatment SSCs
- Support the Development of the Environmental Impact Statement (EIS) and the Severe Accident Mitigation Design Alternative (SAMDA) Analysis
- Maintain a Living PRA
- Risk-informed Inspections
- Support the Development of the Technical Specifications (or equivalent)
- Support the Development of Inspection, Testing and Preventative Maintenance
- Support the Development of Procedures and Training
- Support the Development of Emergency Preparedness
- Assess and Manage Operational Risk
- Assess and Manage Plant Changes
- Monitor SSC Performance
- Maintain a Risk-informed Training Program

III.2.1.2 Instrumentation and Controls

III.2.1.2.1 Background

Advanced LWRs (e.g., AP1000, evolutionary power reactor, economic simplified boiling-water reactor) will provide the first opportunity for vendors to build new reactor control rooms in this country. The design and construction of new plants will utilize the advances made in the development of many of the current LWRs in other parts of the world. The advanced LWRs are expected to have fully integrated digital control rooms at least as modern as those of the N4 reactors in France or the advanced boiling-water reactors in Japan. In addition, the desire for much smaller control room staffs will also push the designs of the plants in the direction of much greater automation, similar to the changes seen in fossil-fired power plants.

The commercial version of the NGNP will involve the coupling of multiple modular HTGRs to advanced balance of plant systems to produce electricity, process heat, and/or hydrogen. This coupling will involve new I&C requirements, including new sensors, data integration, displays, and operational and maintenance philosophies. These new elements are not simply the evolutionary development of the analog I&C and operations and maintenance approaches used in current plants; support of new missions will require fundamentally different hardware, software, sensor systems, staffing, and operational philosophies.

Some HTGR I&C systems will operate in conditions significantly different from those of the current generation of NPPs. Consequently, it is expected that several new kinds of sensors may be developed to monitor these different conditions. Temperature, pressure, flow, and neutron instrumentation may be required to operate in higher temperature environments and use different methods for performing design and safety calculations (e.g., drift, calibration, response time). Current regulatory guidance and tools may need to be enhanced or established to support the review of these systems. Because of longer fuel cycles and much longer times between maintenance outages, HTGR designs may require more extensive online monitoring, diagnostics, and predictive maintenance. Instrumentation will be needed to support this increased automated surveillance. Additionally, the process by which these systems integrate with the control systems needs to be understood.

Other industries have used modern control theory controllers in highly automated control rooms to increase plant availability and decrease operator workloads. Planned HTGR plant designs will involve multiple reactor modules. The use of multiple modular plants may require more complex automated control schemes for both safety and balance of plant systems and highly automated control rooms. These complex controls could include simple feed-forward controllers, nonlinear controllers, neural-fuzzy controllers, or even more advanced methods that the NRC has not yet reviewed. Modular HTGR designs may combine discrete safety control and trip capabilities within the same controller. The effect of these control algorithms on the operational modes of modular designs requires study. The staff may also need to acquire or develop review guidance and tools to analyze and license these advanced control methods.

Additionally, the design of modular HTGRs calls for highly automated operation with reduced supervision by plant operators for long periods of time. These operations may include automated startups, shutdowns, and changes of operating modes. For example, modular HTGR plants will operate as a single large nuclear plant, with perhaps as few as three operators for as many as eight reactor modules per plant. Modular HTGR operations with reduced supervision will require more highly automated and integrated control systems for normal operations and transient and accident conditions. The NRC will need to enhance its understanding of how control and safety systems will be designed to cope with partial failures of interconnected systems, particularly at the switchyard and control room.

In anticipation of the NRC reviews of both advanced LWRs and modular HTGRs, RES conducted a study to evaluate which new I&C systems might need a detailed investigation to support licensing reviews of future reactor control, instrumentation, and protection systems. The study is documented in NUREG/CR-6842, "Advanced Reactor Licensing: Experience with Digital I&C Technology in Evolutionary Plants," issued in April 2004 (ADAMS Accession No. ML041910046). This study concluded that the NRC should initiate early interactions with potential plant vendors to ascertain potential licensing issues in terms of reactor control, instrumentation and protection systems, software, architecture, and human machine interfaces. Early interaction on the research and regulatory development efforts is needed to support the reviews of new instruments, control algorithms, digital technology, or architectures to ensure that the completion of these reviews is not on the critical path for the design certification reviews of advanced LWRs or modular HTGRs.

The study highlighted several new issues that must be resolved to support the review of advanced LWRs. These include improved review methods and criteria for advanced, fully integrated, cockpit-style control rooms and for new technologies such as field programmable gate arrays. Other issues include the need to review system architectures to ensure proper analysis of diverse functions that share common electronics and the need for regulatory positions on digital systems self-testing. Additional issues involve design certification requirements for more advanced reactor concepts such as a modular HTGR currently being proposed for the NGNP reactor and the LMR being considered for the ABR within the GNEP initiative. The commercial version of these non-LWR designs will most likely use a multiple modular plant design approach that will require more complex control schemes for both the safety and balance of systems including the switchyard.

The national and international research community has been involved in R&D of advanced control and monitoring systems for NPPs for the past 5 years. The international community, particularly in Europe, Japan, and Korea, has developed advanced control rooms and performed extensive research in the areas of automation of plant operations and advanced plant monitoring and diagnosis. Thus, there are significant opportunities for international cooperation in this area.

GA is performing detailed control system design studies using plant simulators to optimize control system designs. PBMR Pty. is also investigating the uses of advanced control systems. These R&D activities are being performed by the vendors and through joint efforts with other organizations including universities and U.S. national laboratories such as Oak Ridge National Laboratory (ORNL) and Idaho National Laboratory (INL). There may be an opportunity to capitalize on these research programs, particularly in the areas of advanced control algorithms and control of multiple reactor modules.

I&C is a major area of research outlined in the DOE Long-Term Nuclear Technology Research and Development Plan (http://nuclear.gov/nerac/ltrdp-ne.html). Several proposed research topics in the plan are of particular importance to HTGRs. These include robust communications and wireless sensors, smart instrumentation and condition monitoring, distributed computing, advanced control algorithms, and online monitoring. As part of the implementation of this long-term research plan, DOE has developed six Nuclear Energy Research Initiative (NERI) programs in this area. These include research in the areas of automatic generation of control architectures, self-diagnostic monitoring systems, smart sensors, and advanced instrumentation to support HTGRs. In recognition of the importance of I&C to the success of the NGNP project and GNEP initiative, DOE has recently established a research cross-cut program in this area with the objective to develop these new technologies. The NRC research program will need to work cooperatively with DOE to make use of the information and insight gained as part of this research.

The fiscal year 2005–2009 NRC Digital System Research Plan (ADAMS Accession No. ML061150050) outlines research in several areas of emerging I&C technology and applications that will be used in advanced reactors. These include smart transmitters, wireless communications, advanced predictive maintenance, online monitoring methods, and enhanced cyber security issues. The NRC has recently started new research programs in the areas of wireless communications and online monitoring to support the development of review guidance for these I&C technologies. The guidance will apply to current reactor design I&C upgrades and the review of advanced reactor design applications. Additional R&D described in this section is needed to develop the staff's knowledge and tools to review these new I&C technologies and applications.

III.2.1.2.2 Purpose

The NRC will use the results of the research programs described above to develop regulatory guidance, acceptance criteria, and procedures for the review of advanced I&C proposed for use in advanced NPP safety systems.

III.2.1.2.3 Objectives and Planned Activities

The objectives of this research will be to develop the regulatory infrastructure (review methods and tools) to support the review of new and advanced reactor applications. Some of the new technology will be used for upgrades to operating plants as well as for new and advanced reactors (e.g., better review methods and criteria for advanced, fully integrated, cockpit-style control rooms and unreviewed technology that include field programmable gate arrays). The NRC Digital System Research Plan for fiscal years 2005–2009 or the research plans for the advanced LWRs document the research to support resolution of these issues. Research planned for this program will include only those projects proposed to address open issues associated with HTGR and LMR designs.

Research will be needed in the areas described below.

Analysis of the requirements and potential safety issues involved with instrumentation to support licensing reviews of HTGR and LMR design, construction, and operation. This research area involves gaining a better understanding of how the requirements were developed and which review methods are the most appropriate for advanced neutron detectors, temperature sensors, and other instrumentation in HTGRs and LMRs. The staff will apply the research to support the review of prototype plant instruments proposed for use in advanced reactors.

Analysis of the requirements and potential issues involved with NGNP and GNEP ABR instrumentation. The GNEP facilities will involve both advanced fuel reprocessing technologies and ABR technology. This effort would review the requirements for and the development of new instrumentation for monitoring high temperature and very high temperature process heat, hydrogen, and other unique aspects of these facilities. The staff will develop the instrumentation review guidance needed to support licensing of these facilities.

Develop analytical models for autonomous control of advanced reactors. This research area entails the development of information and models needed to review and examine the advanced autonomous control methods that will be used in advanced reactors. It will involve the review of current methods in use in other technology sectors, such as natural gas power stations.

Develop regulatory criteria for control systems used to integrate the control of multimodular plants and review advanced control algorithms that may be used in safety systems in advanced reactors. This area of research will investigate the degree and the manner by which systems will be integrated in multimodular plants. It will also investigate the points at which control and safety systems are integrated and the extent of automated actions. Additionally, this effort will develop information on the current methods likely to be used in advanced reactors and investigate the potential issues of using these algorithms in a reactor setting.

Analyze advanced diagnostic and prognostic methods needed to support licensing of advanced reactors. Research in this area will entail the review of both current and developmental methods and systems proposed for these reactors and their integration into safety systems and systems important to safety.

III.2.1.2.4 Application of Research Results

The results of this research will provide both tools and methods needed to perform independent licensing reviews of the new I&C technologies that will be an integral part of existing and advanced reactors. These programs will supply the information needed for revisions to Chapter 7 of the SRP and the supporting regulatory guides and review procedures.

III.2.1.3 Human Factors

III.2.1.3.1 Background

The commercial LWR nuclear plants being considered for near-term deployment are expected to use legacy reactor technology. They are also expected to upgrade the balance of plant design to include simplified or passive safety features, digital I&C systems, and computer-based control rooms (Generation III).

Generation III+ and IV plants may be very different from today's plants in terms of reactor technology, advances in I&C, new approaches to human-system interface (HSI) design, and new concepts of operations and maintenance.

III.2.1.3.2 Purpose

Higher levels of understanding of human factors issues will likely be required to develop the tools needed to review and monitor these new plants. Additionally, Generation III+ and IV plant design may require paradigm shifts in how human factors issues are anticipated and reviewed.

Human factors research will be needed to help ensure that NRC regulations and review guidance will adequately support safety reviews of the human factors aspects of all generations of NPPs. Human factors reviews of Generation III+ and Generation IV plant designs will pose new regulatory challenges and require support from human factors research to provide tools, methodology, and a technical basis for regulatory acceptance of advanced NPPs.

The NRC needs to maintain human factors technical expertise to evaluate and establish these new technical bases. This reassessment lays out a program of research to develop the needed technical bases, review guidance, and regulations. As an additional benefit, this research will reduce the uncertainty in the licensing process by increasing confidence in the aspects of advanced plant design related to human performance.

III.2.1.3.3 Objectives and Associated Activities

<u>Recent Research</u>. The NRC describes research conducted to identify potential human performance issues and the technical basis needed to address them in the draft NUREG titled "Human Factors Considerations in New Nuclear Power Plants" (NUREG/CR-XXXX). This research provides some assurance that NRC regulations and review guidance will adequately support safety reviews of the human factors aspects of new generations of NPPs.

Issues identified in the subject NUREG/CR refer to an aspect of new reactor development, design, or evaluation for which available information suggests that human performance may be negatively impacted, or it is suspected that human performance may be impacted, but additional research is needed to better understand and quantify the impact.

The research defined seven high-level topic areas to organize, cluster, and integrate this wide variety of issues. The first five topic areas are derived from a concept of operations model:

(1) roles of personnel and automation
(2) staffing and training
(3) normal operations management
(4) disturbance and emergency management
(5) system maintenance and change management

The following two additional high-level topics account for research issues did not fall within the scope of the concept of operations model:

(6) plant design and construction
(7) human factors engineering (HFE) methods and tools

To complete this effort, researchers must assign priorities to each of the issues included in these topics. They will accomplish this by subjecting each of the issues to a phenomena identification and ranking table (PIRT) process for further analysis. The PIRT analysis will do the following:

- further define the human performance issue
- identify the significance of the issue and a basis for the significance determination
- identify the availability of information to address the issue
- identify the class of reactors to which the issue applies

Based on these factors, a team of stakeholders from the industry, vendors, researchers, regulators, and knowledgeable public will prioritize these issues as items important to safety that require research. The team will complete this work in early 2007.

Research Plan. The human factors in the advanced reactor research plan are based on the issues identified in the above cited draft NUREG-XXXX and organized within the high-level topic areas listed previously. The NRC may reprioritize the plan based on the results of the PIRT. A brief description of the research needs in each high-level topic area follows.

1. Roles of Personnel and Automation

This research topic addresses the relative roles and responsibilities of personnel and plant automation and the relationship between the two. The overall level of automation in new reactors is expected to be much higher than in today's plants. Advances in digital technology offer new and more flexible types of automation that allow personnel interaction at varying levels.

Automation can change the operators' role in monitoring, detection, and analysis of off-normal conditions, situation assessment, and response planning. Research is needed to determine the effect of these changes on operator safety performance and on review methods to ensure that applicants have adequately accounted for the effects of changes they propose. Examples of such technology include computerized procedures, computerized operator support systems, and intelligent agents.

2. Normal Operations Management, Intelligent HSIs

This topic addresses research related to how the plant will be operated during normal evolutions such as startup, low power, full power (base load or load following), and shutdown.

One of the lessons learned from plant modernization programs is that the full impact of technological changes is often not anticipated. This stems in part from limitations in knowledge about the effects of technology on human performance.

Knowledge of the operating experience of Generation III and III+ reactors is also limited. Research is needed to obtain and analyze information from vendors, utilities, and regulatory authorities as a means to identify the necessary future research guidance and research to fill gaps in existing safety review guidance.

Several operational aspects of new reactor designs that differ from those of the current LWRs are modular plant operations, management of new reactivity, load following operations, and continuous fueling. These new operational approaches require various levels of research to determine their impact on human safety performance.

HSI technology that personnel use to perform their tasks is changing. Examples of HSI technology considerations in new plants include interfaces with more diverse automation systems, use of sensors and condition monitoring, use of digital communication networks, the design of large information systems, and advanced controls.

One impact of the new I&C and HSI technologies is that the HFE aspects of the plant increase in complexity and opacity. Evaluating the effect that these systems have on human safety performance will require technical bases and guidance. Based on current trends, it is likely that HSIs will continue to become more intelligent. The knowledge and reasoning bases of these systems will be diverse (e.g., application of knowledge engineering or use of formal analysis rules). At present, the NRC's "Human-System Interface Design Review Guidelines" (NUREG-0700), issued in 2002, does not have sufficient guidance to address the review of the technical bases for intelligent HSIs.

3. Normal Operations Management, Safety Culture

Diversity in new designs and operational concepts may have an impact on safety culture. One research topic that arises is how safety culture is transmitted to personnel at individual units. A related question is how combining units with different original cultures under a single large operating entity affects safety culture.

4. Disturbance and Emergency Management

This topic includes research related to how plant design addresses plant I&C disturbances and handles abnormal events. While improved computer support will be available, personnel may deal with situations in which they must manage plant operations under conditions of degraded I&C and HSIs.

A variety of events, such as instrument failure, computer failures (hardware and software), seismic events, fire and smoke damage, internal flooding, and loss of electrical power, may cause I&C degradation. These events may result in a variety of failures, ranging from individual control room instruments to more significant degradations such as the loss of all displays.

The staff will need guidance to review the methods an applicant proposes for mitigating the effects of degraded I&C conditions on human safety performance, such as the ability to detect digital system failure and ways to transition to backup systems.

5. System Maintenance and Change Management

This topic addresses concepts for system maintenance, installing upgrades, and configuration management. Since digital I&C systems and computer-based HSIs develop rapidly, they will require more frequent changes because of obsolescence.

Since equipment will become obsolete much faster than in current plants, applicants need to ensure that maintenance of installed equipment will continue and that adequate vendor support will be available. Also, vendors will be offering enhancements as their product lines and associated functional capabilities evolve.

The staff must assess the effects of these changes to determine the impact on operations, maintenance, and training. The regulatory process will have to keep pace with these changes through flexible guidance based on up-to-date technical bases.

6. Plant Design and Construction

New reactors will employ digital I&C systems relying heavily on software for critical monitoring and control functions. This reliance on software is a particularly important aspect of the design of new plants. One of the biggest issues in software development and maintenance is human error. The human performance aspects of the design and evaluation of digital systems and software are an important research topic which could result in new review guidance.

7. HFE Methods and Tools

This topic pertains to the design and evaluation of the HFE aspects of a plant (i.e., the resources used by HFE personnel, whether as part of vendor organizations, licensees, or the NRC, to accomplish their roles and responsibilities). This topic is especially relevant to NUREG-0711, "Human Factors Engineering Program Review Model," issued in February 2004.

The availability of computer support has sparked many advances in these methods and tools. Advanced reactors are expected to use dynamic function allocation which changes the function of the operator based on the state of the plant.

Increasingly, plants are using levels of automation to help crews maintain better awareness of the automatic actions and to be more informed when disturbances in the automation occur. Thus, the role of operators who have the ability to perform a variety of functions in the control and management of automated systems would have to be assessed for its impact on safe operations. Techniques such as virtual reality and human performance modeling are being used on a more routine basis for system design and evaluation.

An important area of HFE is HRA. Methods and data may be lacking for application to new designs that incorporate increased automation, alternative concepts of operations, and intelligent interfaces. Research is needed to determine the acceptability and limits of such tools.

Another emerging research area is the use of knowledge engineering techniques (i.e., techniques for identifying and documenting the knowledge of subject matter experts). When this knowledge is combined with simulation and analysis tools, a powerful knowledge base is created upon which to improve operations and maintenance performance.

This information can be applied to the development of more intelligent interfaces (such as intelligent alarm processing and analysis) and to intelligent agents that reflect the knowledge of experts. Efficient methods to obtain and store such knowledge in integrated databases are needed. In addition, the staff needs review criteria to evaluate HSIs developed using the knowledge elicited from experts.

III.2.1.3.4 Application of Research Results

New plants will offer the potential for improved performance and safety. However, there are still challenges ahead, especially as personnel and technology are integrated into final designs. Findings from these research topics will provide the technical basis for developing regulatory review guidance to meet these challenges. This review guidance will help do the following:

- Set clear expectations for the NRC staff's evaluation of new designs, reduce regulatory uncertainty, and provide a well-defined path for new reactor licensing

- Identify the need for safety enhancements or other regulatory action

- Provide the technical basis or criteria for acceptability (input for potential rulemaking, branch technical positions, inspection guidance, or policy statements)

- Provide guidance for NRC reviewers (regulatory guides, SRP enhancements, NUREGs)

The staff will also need to monitor continuing industry developments in the area of human performance to identify new and emergent research topics so that they can be integrated into the plan as appropriate.

III.2.2 Reactor Plant Systems Analysis

A primary infrastructure need in the reactor systems analysis area is the development of appropriate databases and analysis tools to help make sound decisions on key technical and regulatory issues concerning HTGR safety and licensing. Data, tools, and methods are needed in the reactor systems and plant systems analysis areas to assess HTGR safety margins and to enable the staff to evaluate the safety analyses submitted by HTGR applicants. Research in the reactor systems analysis area is also needed to provide analytical support for the establishment of regulatory requirements and guidance for HTGR safety and licensing and to establish the technical basis for related policy decisions.

This section addresses infrastructure needs in the area of HTGR reactor systems analysis, which includes thermal-fluid analysis, nuclear analysis, and accident source term analysis. The planned approach for thermal-fluid analysis provides the data and modeling tools needed for predicting HTGR-specific heat transfer and fluid flow phenomena, including "multiphase (helium with air and/or water)" fluid flow with convection, conduction, and radiation heat transfer mechanisms in irregular and complex geometries. HTGR thermal-fluid analysis also supports HTGR fission product transport and source term analyses because the failure of HTGR coated fuel particles is, among other things, a function of fuel temperature during normal operation and accidents and many of the important fission product transport mechanisms during normal operation and accidents depend on local thermal-fluid conditions. HTGR nuclear analysis requires developing modern, general-purpose nuclear data libraries that support all nuclear analyses associated with reactor safety, materials safety, waste safety, safeguards, and security. Nuclear analysis safety R&D needs include development and testing of (1) reactor physics code capabilities 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. The planned approach for HTGR accident source term analysis addresses the data and analysis tools needed to evaluate (1) the magnitude, distribution, and chemical form of radionuclides within the power system pressure boundary during normal operation, (2) the progression of credible accident scenarios involving additional fuel failure accident phenomena, such as high temperature or high temperature with chemical attack conditions, and (3) modeling any releases and transport of radionuclides inside and outside the reactor system boundaries that result from the first two aspects.

For HTGRs, coated fuel particles (CFPs) provide the primary FP barrier. The integrity of the CFPs depends on the maximum local fuel particle temperature during normal operation and accidents. Reactor system analysis methods, using a combination of codes and models for core neutronics, decay heat power, and system thermal-fluid analysis, predict HTGR local and global fuel temperatures. Researchers conducted experiments with unfueled graphite pebbles containing small thin metal wires having a range of melting temperatures in the German Arbeitsgemeinschaft Versuchsreaktor (AVR) pebble bed reactor. The “melt-wire” experiments showed that local core temperatures during normal full-power operation were significantly higher than the analytical codes used at that time had predicted. Moreover, the true maximum local operating temperatures of the AVR remain unknown because of the limitations of those experiments.

For modular HTGRs with passive accident decay heat removal phenomena, there are significant uncertainties in predicting the maximum fuel temperatures and maximum vessel temperatures during heatup accidents. Uncertainties involve factors such as irradiation-dependent and temperature-dependent graphite thermal conductivity and the integral effects of variable local power densities with conductive, radiative, and convective heat transfer through the core and surrounding structures. Appropriate measurements and system analysis tools will be needed to support the staff’s understanding and assessment of factors that govern reactor and plant temperatures and their uncertainties. The staff will need these to assess HTGR SSC temperature margins, such as fuel integrity temperature margins.

R&D, including sensitivity studies, with analysis codes and data will also be needed to assess the safety-related technical and policy issues (e.g., containment/confinement performance requirements) associated with HTGR accident radiological release phenomena which differ from those of operating and advanced LWRs. To meet the research needs in the reactor systems analysis area (i.e., nuclear analysis, thermal-fluid analysis, accident source term analysis), the agency will stress cooperative planning and implementation of domestic and international safety research to minimize costs and maximize benefits.

Reactor systems analysis infrastructure development will be needed to provide an adequate suite of reactor systems analysis tools (i.e., computer codes and methods) that provide the NRC staff with an independent capability to reliably predict HTGR system behavior and fission product release in response to initiating events. The NRC staff will use the suite of analysis tools, and the data to develop and validate them, to (1) conduct confirmatory analyses of an HTGR applicant’s safety analyses, (2) support development of the HTGR regulatory requirements by assisting, for example, in better understanding the modeling of key accident phenomena and analyzing the performance of safety-related equipment during licensing-basis events, and (3) conduct sensitivity studies to better understand uncertainties and safety margins. The reactor systems analysis infrastructure development will also provide information needed to support research program infrastructure development in other technical areas. These include providing fluence and temperature and pressure for the determination of loads and degradation of components associated with the materials analysis and fuel analysis areas as well as information on sequences and damage states for input to the evaluation of plant PRAs.

III.2.2.1 Thermal-Fluids Analysis

III.2.2.1.1 Background

HTGR safety, including accident fission product releases, depends on thermal-fluid dynamic behavior of the core and the integrity and functional performance of the major safety significant fission product barriers and components that comprise the core, reactor, and plant systems. The NRC staff, in its review of HTGRs for design certification, will need computational tools to enable it to evaluate safety issues and to understand the performance of the fission product barriers and SSCs during normal operations and accident conditions. The staff will also use these computational tools to provide insights to support the development and assessment of NRC technical requirements and guidance for HTGR safety analyses and licensing. This section describes the thermal-fluid technical and safety issues and infrastructure development needed to address these issues and areas.

Two HTGR designs, modular HTGRs with pebble bed cores (PBRs) and modular HTGRs with prismatic block cores (prismatic modular reactors or PMRs), are considered for thermal-fluid infrastructure development. The NGNP reactor is expected to have many of the same reactor safety and technical issues and infrastructure development needs as either a PBR or PMR even though the actual design type has not been selected at this time. For either reactor core type, applicants will be required to demonstrate adequate accident prevention and mitigative capabilities and, similar to light-water-cooled power reactors, will be required to demonstrate compliance with specific SSC design and safety limits. Specific HTGR limits have not yet been established. However, for purposes of determining thermal-fluid infrastructure development needs, figures of merit (e.g., surrogate acceptance criteria) for selecting which thermal-fluid analysis will be used to demonstrate compliance include the following:

- In-Core
  - Peak fuel temperatures must not exceed the maximum fuel qualification temperature.
  - Fuel oxidation must not exceed the maximum fuel qualification oxidation.
  - Fuel particle energy deposition rate does not result in a fuel temperature transient that exceeds some limit meant to ensure integrity of the fuel.

- Ex-Core

  – Reactor vessel and core support structures do not exceed acceptance limits so that core geometry is maintained.

In addition, the licensing basis for HTGRs is expected to involve regulatory limits on the onsite and offsite dose consequences for normal operation, anticipated transients, design-basis accidents, and accidents beyond the design basis. The source term used for these assessments depends on the initial FP release caused by circulating and plate-out activity (for the prompt release calculation) within the pressure boundary, as well as the FPs released from the fuel hours (or days) later in the accident (for the delayed release calculation) because of fuel accident heatup (and potential oxidation). The FP released within the helium pressure boundary during normal operations (which provides the basis for the initial FP accident release) requires an analysis for the three-dimensional temperature distribution in the core during normal operations as input to the coated particle fuel failure analysis, as well as an analysis of the steady-state FP inventory distribution within the helium pressure boundary resulting from the transport and deposit of FPs on internal surfaces. The FP release from the fuel many hours or days later in the accident (delayed release) requires a global analysis of the subsequent three-dimensional heatup temperature distribution in the core, combined with the potential spacial oxidation degradation effects of air ingress or water vapor ingress. The later release from the deposited and circulating inventory requires an analysis of the internal transient thermal-fluid flow conditions that provide the driving forces for transport of these FPs. Thus, the thermal-fluid analysis also provides a primary input to the FP transport and release (i.e., accident source term) analysis. In this way, the thermal-fluid analysis supports, and may be coupled to, the FP transport and release models in the accident (i.e., dose consequence) analysis and will be included in the infrastructure development.

Analysis and experimental investigation will establish demonstration of compliance with the core and other SSC design limits. HTGR systems must be modeled and various accident scenarios simulated to demonstrate that design limits are not exceeded. The best estimate codes to be developed for this purpose must be validated against experimental data obtained from well-scaled test facilities over the range of conditions expected in the accident scenarios. Uncertainties and biases in the codes also need to be clearly understood and quantified. This represents a challenge to the NRC for thermal-fluid infrastructure development because licensing and design approval of passive HTGR systems as the current capability of the NRC's codes and the available experimental database for validation are limited. The status and development needs for current NRC thermal-fluid analysis tools are described to provide an independent evaluation of the performance of passively cooled HTGRs.

III.2.2.1.2 Gas Reactor Designs

HTGR plant designs continue to evolve, but two types may be submitted for design certification or license application. PBR and PMR designs both use helium as the coolant and use a

Brayton cycle to obtain high thermal efficiencies. Plants with both direct cycle and indirect cycle (i.e., with an intermediate heat exchanger (IHX) between the reactor and balance of plant) are being considered. The NGNP reactor and plant design concept has not yet been selected, but for purposes of assessing infrastructure development needs, it is assumed that it will be a very high temperature gas-cooled reactor (GCR) with either a prismatic hexagonal block core or a pebble bed core utilizing either a direct or indirect cycle. The NGNP reactor design may also be smaller in power and size than either of the PBR or PMR designs being developed for commercial electric power generation. Regardless of the specific design, many of the thermal-fluid issues are common to both design types. Steady-state plant operating conditions are also similar, with total core power ranging from 400 to 600 megawatt thermal (MWt) and core outlet temperatures up to 950 °C currently being considered. Core operating pressures of 7 to 9 MPa are typical. Tables 1 and 2 list nominal full-power conditions for the PBR and PMR designs.

Table 1 PBR Thermal-Fluid Design Conditions

Reactor power (MWt)	400
Reactor inlet temperature (°C)	500
Reactor outlet temperature (°C)	900
Core inlet pressure (MPa)	9.0
Helium mass flow rate (kg/s)	193
Active core inside/outside diameters (m)	2.0/3.7
Active core height (m)	11
Outer reflector outside diameter (m)	5.5

Table 2 PMR Thermal-Fluid Design Conditions

Reactor power (MWt)	600
Reactor inlet temperature (°C)	490
Reactor outlet temperature (°C)	850
Core inlet pressure (MPa)	7.07
Helium mass flow rate (kg/s)	320
Active core inside/outside diameters (m)	2.95/4.83
Active core height (m)	7.96
Outer reflector outside diameter (m)	5.64

III.2.2.1.3 Accident Scenarios and Off-Normal Conditions

This subsection describes the accident scenarios of interest and points out the thermal-fluid phenomena that occur. While the full spectrum of accident scenarios considered as part of an HTGR design basis has not yet been firmly established, existing studies of PBMR and GT-MHR have shown that a loss of normal heat removal is an important type of accident to be modeled for assessment of design margins. For dose consequences, events involving the loss of pressure boundary are generally the most severe. Accidents in the former category include the loss of forced circulation (LOFC) with reliance on passive heat removal systems as an important accident scenario. If the system pressure boundary remains intact, the reactor pressure is maintained and the event is called a pressurized LOFC transient or a "pressurized cooldown." The coolant is not lost during this type of event, and the helium coolant remains at high pressure. Heat is removed by radiation from the core to the reactor pressure vessel wall, and then through successful operation of the passive reactor cavity cooling system (RCCS). Buoyancy and natural convection circulation play an important role in the core and reactor pressure vessel temperature distributions, with the chimney effects tending to make temperatures highest near the top of the reactor pressure vessel for the pressurized conduction cooldown. Thus, the primary thermal-fluid analysis needs are to determine the core temperature distribution and maximum fuel temperature, while the design criteria of interest internal to the core relate to the fuel, and the design criteria external to the core relate to maximum temperatures for the vessel and support system components.

For assessment of dose consequences, the scenarios of interest for in-core and ex-core temperature calculations involve the failure of the pressure boundary with the attendant LOFC. These events may range from a small leak to the postulated double-ended guillotine break of the cross-connection inlet/outlet pipes (or vessels) that are used to transfer helium coolant between the reactor unit and the IHX or directly to the power conversion unit. In loss of pressure boundary scenarios, the system eventually depressurizes to atmospheric pressure. For the large breaks in the design-basis event (DBE) or beyond-design-basis event (BDBE) category, decay heat is removed from the core by conduction, convection, and radiation, and from the vessel wall to the RCCS by radiation and convection. For small breaks in the anticipated operational occurrence (AOO) category, decay heat may be removed initially by the decay heat removal system and subsequently by the RCCS. Large breaks involving RCCS cooling are also referred to as "depressurized conduction cooldowns." Thermal convection effects for helium at atmospheric pressure tend to be insignificant for this type of event.

Convection cooling and natural circulation within the vessel become important, however, in depressurized LOFC events involving the ingress of ambient air into the primary system. Depending on the break or location of the opening into the primary system, relatively high-density air diffuses into the vessel, and the oxygen in the air oxidizes graphite in the core supports, core or reflectors, and active fuel region. Heat generated by oxidation of the graphite can result in higher rates of circulation within the vessel and convective heating or cooling of the core. The amount of oxygen available limits the oxidation in the active core region, and the graphite oxidized maybe restricted to that in the lower reflector and bottom of the active fuel.

Water ingress can be a concern in depressurized LOFC events if the system design includes water-cooled heat exchangers or intercoolers. Even though the water-cooled system components operate at a pressure significantly less than the helium coolant, if the boundary between the two systems should fail, the water-cooled system could overpressurize and water could enter the primary. While transport of water through the primary and into the core appears to be a very low probability event, water ingress into the core would result in a reactivity insertion and possible recriticality.

The NGNP is being designed to produce hydrogen and would couple the reactor to the hydrogen production plant via an IHX. The IHX would be used to transfer heat from the reactor core to a high-temperature electrolysis system. The high-temperature electrolysis system uses electricity to break down the water molecules into constituent elements, creating noncondensable hydrogen that circulates on the secondary side of the IHX. At very high temperatures, noncondensable hydrogen has the potential to diffuse into the helium coolant at high rates through metallic pressure boundary components. Research has evaluated the performance of IHXs for hydrogen production. However, because of the very high temperature requirements, design of an adequate IHX will be difficult, and evaluation of its performance from a safety perspective will be important in determining the overall plant safety. High-temperature electrolysis is only one design suggested for hydrogen production coupled to the NGNP reactor. In general, other sections of this infrastructure assessment, such as the analysis of high-energy external events initiated by events originating in the hydrogen production plant, address the safety and regulatory implications of coupling a hydrogen production plant with an NPP. However, since hydrogen gas can potentially be introduced into the primary system during a postulated IHX break, the effect on core natural circulation flow rates and overall core cooling should be addressed (most likely as a variation in primary system fluid properties).

III.2.2.1.4 Full-Power Operation Safety and Technical Issues

During full-power operation, the flow of helium is forced through the core at relatively high rates. The fuel is cooled by forced convection. Consequently, buoyancy and thermal radiation, which are important during most accident scenarios, are not significant. However, during normal operation, flow from the core outlet may be subject to "hot streaking," and significant bypass of coolant may occur in certain regions within the core. "Hot streaking" refers to local high-temperature jets that exit the core and enter the core support lower plenum region. If these high-temperature jets do not mix sufficiently with the cooler helium coolant flows entering the lower plenum, the hot jets will impinge on lower plenum structural components and potentially on internal components within the cross-connect vessel. This may create elevated or unacceptable thermal stresses within the components which might lead to unacceptable structural degradation or failure.

Bypass refers to helium coolant that is diverted to low-power or unpowered regions of the core, such as the inner reflector in a PBR core or through small passages between the adjacent hexagonal graphite reflector blocks or fuel blocks in a PMR core. The decrease in local convective cooling flow rate from higher power to lower or unpowered core locations can result in higher than expected local fuel temperatures and higher core exit temperatures.

Nonuniform heating can also be a source of local hot spots. Another cause of hot streaking and uncertainty in determining bypass flow is the effect of strong heating on a gas that shows a large variation in transport properties with temperature. The reduction in gas density with strong heating can cause acceleration of the flow in specific regions of the core

III.2.2.1.5 Dominant Physical Processes

The previous section on accident scenarios noted several physical processes that play important roles. Most of these phenomena occur in both the PBR and PMR designs and thus can be considered generic in terms of applicability and occurrence. This subsection discusses and summarizes these phenomena and features which are expected to be important considerations in infrastructure development for HTGRs.

<u>Molecular diffusion</u>. In depressurized LOFC events with air or water ingress, molecular diffusion initially transports air or water vapor through the lower plenum to the lower reflector. The rate of diffusion of air

through helium in the lower plenum determines when oxidation of the lower core reflector and fuel near the bottom of the core begins.

Natural circulation and buoyancy. Natural convection flow and heat transfer are important in several events. In the pressurized LOFC event, natural convection acts to make temperatures in the core and vessel relatively uniform. Rising hot plumes of helium entering the upper plenum may cause the upper reactor vessel head temperature to rise to unacceptably high temperatures. In the depressurized LOFC event with air or water ingress, natural circulation becomes important in heating or cooling the core following the start of oxidation.

Graphite oxidation. The oxidation of lower reflector graphite and fuel near the bottom of the active core can occur during depressurized events with air ingress. The heat of reaction enhances circulation within the core and vessel. Availability of air is important in the evaluation, as there may not be enough oxygen to sustain significant oxidation rates since the reactor cavity is well below grade. In addition, the heat release from graphite oxidation is approximately equal to decay heat and thus represents a significant contributor to heatup of the fuel.

Pressure drop through a pebble bed core. Simulation of the flow resistance through a pebble bed core is important in obtaining the temperature distribution in the fuel. Bypass around the pebbles near the central and outer reflectors will depend on the resistance through the pebble bed and wall drag along the reflector flow path. The simulation may need to account for random variations in the packing fraction and their impact on the flow.

Core heat transfer. Heat removal from the core to the reactor vessel wall depends on several individual heat transfer mechanisms. In a pebble bed core, heat is transferred by conduction and radiation from pebble to pebble and from pebbles to graphite structures. Convection to the coolant occurs, but it is not as effective as conduction and radiation without forced circulation. In depressurized LOFC events, the effective core conductivity is the dominant parameter and uncertainty for heat removal from the core. This effective conductivity depends on the relative contributions of conduction heat transfer through the core, reflector, and vessel structures.

Reactor cavity heat transfer. Similar to core heat transfer, reactor cavity cooling is dominated by radiation heat transfer. The RCCS in both the PBR and PMR designs relies on passive natural circulation (through vents) and radiation heat transfer to remove heat from the reactor pressure vessel. In effect, the RCCS acts as the link between the reactor pressure vessel and the ultimate heat sink (e.g., the ground).

Reactivity insertions. Two types of events have the potential to cause reactivity insertions resulting in recriticality. Depressurized LOFC events with water ingress may transport sufficient water to the core to cause a significant reactivity insertion. Pressurized LOFC events with anticipated transient without scram (ATWS) can also achieve recriticality later in the event after xenon decay. Simulation of this process depends not only on the core neutronics but also on the temperature and water vapor distribution in the core.

III.2.2.1.6 Material and Thermophysical Properties

Helium transport properties. Transport properties for helium at high temperature under strong heating conditions vary considerably [3] and can lead to high uncertainty in heat transfer at low Reynolds numbers and in natural circulation calculations. Correlations that fail to account for this property can be significantly in error, as for low Reynolds number flow in a circular tube. Typical convective heat transfer correlations were found to overpredict the heat transfer coefficients.

Gas mixture properties. HTGR events do not always involve a single media. Mixtures of helium and air, helium and water vapor, and helium with hydrogen and water vapor need to be considered. Gas mixture properties, however, are not necessarily linear with composition, and analysts should use techniques such as those described by Reid et al. to determine transport properties in gas mixtures.

Solid material properties. In the pressurized LOFC events, thermal radiation from the core to the vessel and from the vessel to the RCCS are dominant heat transfer mechanisms. To properly evaluate radiation, analysts must know the emissivities of the fuel, vessel, and reactor cavity. These can be functions of surface temperature and surface condition. Emissivity, as well as the effective conductivity of graphite, which can be a function of irradiation history, temperature, orientation, and annealing effects, is also an important factor in depressurized LOFC events.

III.2.2.1.7 Experimental Facilities/Needs

Experimental data represent the foundation for code development and assessment and are also essential in understanding the behavior of a system during a simulated accident. As discussed in previous sections, natural circulation and molecular diffusion are important processes that are expected in some events. Both processes, however, are strongly dependent on geometry because of the effects of the geometry on flow resistance and fluid mixing. This subsection discusses available and planned sources of data that may be used for assessment. First, it describes large-scale integral test facilities and then separate effects tests. The discussion includes both existing and near-term planned test facilities.

Integral Nuclear Reactor Test Facilities. HTGR research and test reactors have provided a significant amount of operational performance data. These include the high-temperature test reactor (HTTR), a 30 MWt prismatic reactor, operated by the Japan Atomic Energy Research Institute (JAERI), and the high temperature reactor (HTR-10), a 10 MWt pebble bed reactor operated by China's Institute of Nuclear and New Energy Technology (INET). These research reactors, which achieved first criticality in November 1998 and December 2000 respectively, have been running a variety of tests and are capable of running integral experiments to provide data that cannot be obtained at separate effects test facilities. Both the HTTR and HTR-10 have completed a variety of operational tests confirming their power performance and various plant systems. The HTTR has achieved a helium outlet temperature of 950 °C and is currently running control rod withdrawal tests and primary coolant flow reduction tests at various power levels. China's HTR-10 completed loss of offsite power tests, helium blower trip tests, and turbine trip tests in 2003. Both the HTTR and HTR-10 are planning additional tests including a complete loss of forced cooling (LOFC) test, an all-blackout test, a pressurized conduction cooldown test, and an ATWS test Both the HTTR and HTR-10 also plan to run RCCS experiments.

A third research and test reactor, the high temperature teaching and test reactor (HT^3R), proposed by the University of Texas system and GA, would also have the capability to run integral experiments. In addition to the proposed reactor, various laboratories would be included to allow the study of methods for production of synthetic fuels and hydrogen and methods for generating electricity with increased efficiencies. This project is currently in a preconceptual design phase.

Separate Effects Test Facilities. Several HTGR test facilities have the capability of producing thermal fluid code validation data by simulating the core with electric heaters. The JAERI Ingress Test Rig has been used to study molecular diffusion, natural convection, and chemical reactions in thermal and isothermal environments Several other facilities have completed or are planning water-cooled and/or air-cooled RCCS tests including Argonne National Laboratory's Natural Convection Shutdown Test Facility, Germany's Inactive Decay Heat Removal Facility, and Seoul National University's RCCS Facility.

PBMR Pty., the developer and vendor of the PBMR, is operating experimental heat transfer facilities in South Africa in cooperation with IST Nuclear, North West University, and M-Tech. These include the Helium Test Facility, PBMR Micro Model, and the Heat Transfer Test Facility. These facilities are planning core heat transfer tests, auxiliary plant systems tests, and full-scale design life tests

The Forschungszentrum Jülich (Jülich Research Center) in Germany operates two test facilities, the Natural Convection of Air through the Core with Corrosion (NACOK) Facility and the Self-Acting Removal of the After-Heat of the HTR-Modular Reactor (SANA) Facility. The NACOK facility allows modeling of natural circulation in a pebble bed reactor under varying hot- and cold-leg temperatures to simulate air flow though a pebble bed in various experimental layouts. Resulting data have been used for computation fluid dynamics (CFD) code validation. The SANA facility allows study of the heat transfer mechanisms in a pebble bed core upon depressurization; however, the validity of the results for code benchmarking has been disputed because of a large difference in geometry between SANA and the PBMR and because SANA was designed based on the data required for codes used at the time, and therefore, flow in the pebble bed is neglected or approximated using the correlations obtained from the tests

The Idaho National Engineering and Environmental Laboratory (INEEL) Matched Index of Refraction Flow System is the world's largest of the type and researchers will use it to conduct basic and applied studies on complex turbulent and transitional flows, flows in porous media, and two-phase particulate flows using optical techniques, such as particle image velocimetry, laser Doppler velocimetry, particle tracking velocimetry, and a moving particle tracking system. Experiments run at the INEEL facility are isothermal and therefore do not reveal any buoyancy contributions from temperature variations. A variety of NGNP-sponsored experiments, including pebble bed exit flow testing and lower plenum fluid dynamics modeling will use the Matched Index of Refraction Flow System.

Russia's OKBM has operated several thermal-fluid and fluid dynamics facilities; however, all are currently nonoperational. These include the High Temperature Gas Test Facility, Main Circulator Test Facility, High Temperature Helium Test Facility, Control Rod Drive Mechanism Test Facility, Masex Test Facility, and TIGR Test Facility. Notable experiments run at these include an investigation of helium-air mass transfer through orifices and tubes and the influence of gas circulator operation on mass transfer (completed at Masex Test Facility in 1991) and investigations of the removal of residual heat during reactor cooldown and mass transfer at coaxial duct loss-of-integrity and chimney effect

III.2.2.1.8 Analytical Capabilities/Needs

The analytical capability to model and simulate HTGR accident scenarios and perform an independent evaluation of HTGR safety and mitigative features of a passively cooled gas reactor needs to be developed. Thermal-fluid analysis will be used not only for design-basis scenarios but also to support the review of PRA and structural analysis applications. This analytical capability will likely require one or more thermal-fluid computer codes capable of modeling the geometries of gas-cooled systems and the physical processes expected to occur. Because of their complexity, these codes need detailed assessment to demonstrate that they are appropriate for the proposed application.

The staff will need analytical capability on two different scales in its evaluation of GCRs. Systemwide calculations will be necessary to determine flows into and out of the core, major components, and surrounding environment (containment or confinement). Systems analysis codes will be needed to track multiple gas species (helium, air, water vapor, and possibly hydrogen) and calculate the dominant physical process, such as natural circulation, molecular diffusion, and graphite oxidation. Since the PBR and PMR designs both utilize a direct gas-turbine (Brayton) cycle, the staff will need models to simulate the power conversion unit and its interaction with the reactor core.

The analysis capability must also be able to provide details on the temperature and flow distribution in the core, the reactor vessel, and surrounding cavity cooling space and support structures. Determination of local hot spots within the core or plena requires a code that can calculate turbulent mixing and the formation of thermal plumes in natural convection flows. The RCCS operates using natural convection, the analysis of which also benefits from a code that can calculate boundary layer phenomena and mixing with ambient fluid.

The staff is currently using three codes with applicability to GCRs. None are completely qualified because of a lack of development and experimental data, but they could be used as a starting point for future work. These codes are discussed briefly below.

The Graphite Reactor Severe Accident Code (GRSAC), developed at ORNL, can simulate a wide range of accidents in gas reactors. Researchers have used it to simulate both the PBR and PMR designs, as well as benchmark transients run in the HTTR and HTR-10 integral test facilities. The forerunners of GRSAC, called ORECA and MORECA, developed in the 1975 to 1993 time-frame at ORNL, largely under NRC sponsorship, supported the staff’s licensing safety evaluation for Fort Saint Vrain and the preapplication review for the DOE modular high-temperature gas-cooled reactor (MHTGR). After 1994, MORECA became GRSAC and, through non-NRC funding sources (mainly the Defense Nuclear Agency), was further developed to model past accidents and postulated events in various non-HTGRs, such as Windscale, Magnox, and advanced gas-cooled reactors (AGRs). ORNL has added models appropriate for a pebble bed core. GRSAC provides for a detailed three-dimensional model of the core, plus models for the reactor vessel, shutdown cooling system, and RCCS. The core model in GRSAC allows for investigation of azimuthal temperature asymmetry, as well as axial and radial variations. The staff has obtained this code and has been using it to perform scoping calculations of the PBR and PMR.

The MELCOR code is a severe accident code developed at Sandia National Laboratory for the NRC to model the progression of accidents in light-water-cooled reactors. MELCOR models have been developed to simulate most aspects of a pebble bed reactor. Modifications include implementation of multifluid tracking capabilities, a graphite oxidation model, and a simple molecular diffusion model. Correlations were also added to model heat transport in a pebble bed and to include the effect of neutron fast fluence on thermal conductivity of the core. A study conducted by INEEL used MELCOR, which allows for general and flexible nodalization, to develop a detailed model of the reactor pressure vessel and RCCS.

Design and development organizations have used FLUENT and other CFD codes to examine the details of flows in AGRs. The ability of these codes to simulate turbulent mixing in complex geometries makes them well suited for analysis of flows in the upper and lower plena of GCRs where buoyant plumes and hot jets in the lower plenum may exist. Natural convection flow and heat transfer dominate cooling in the RCCS, and CFD may be needed to effectively examine the details involved in operation of gas reactor systems. The staff may also need a CFD capability to calculate the steady-state distribution of radionuclides on the internal surfaces of the pressure boundary system to provide the initial conditions for the calculation of the initial releases in the source term analysis.

III.2.2.1.9 Summary and Recommendations

The NRC staff will need to perform thermal-fluid evaluations of HTGR systems to (1) define regulatory safety limits and determine if those limits and an acceptable safety margin are maintained, (2) predict transient behavior in HTGR licensing-basis accidents, and (3) provide system performance information for PRA and inputs to other areas such as fuel performance and source term analyses.

Two types of codes will likely be necessary to perform thermal-fluid analysis for gas reactors. A reactor systems analysis code, such as GRSAC or MELCOR, can provide the overall, global systems behavior of the GCR. To analyze reactivity-initiated transients, such as water ingress events or ATWS, the staff may need to couple the systems analysis code with the Purdue Advanced Reactor Core Simulator (PARCS) neutronics code. A CFD code, such as FLUENT, will be needed to provide detailed local temperature and velocity distributions. Regardless of which code(s) are selected, significant code development and assessment will be necessary to improve and quantify models and correlations for gas reactor phenomena.

The staff will need data 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, the Thorium Hochtemperaturreaktor (THTR) in Germany, the HTTR in Japan, and the HTR-10 in China. These and other experimental programs, such as the air-ingress tests done in the NACOK facility at FZ-Jülich and in a similar facility at JAERI, as well as the pebble bed fluid-flow and heat-transfer tests performed at the SANA facility at FZ-Jülich, provide significant sources of measured data. However, additional data are 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 LOFC accidents, air and moisture ingress accidents with oxidation, and reactor cavity cooling. The NRC should initiate cooperative ventures with the international community to identify data needs, develop experimental facilities, and obtain the necessary data.

The code development, assessment, and experimental efforts will need to address several modeling and analysis issues:

- The capability to model flow and heat transfer in pebble beds must be confirmed and quantified. Existing models in GRSAC and MELCOR are simple and were not originally developed for nuclear applications. The appropriate constitutive relations for pressure drop, conduction in a packed bed, and modeling techniques sufficient to identify and capture local hot spots must be identified and implemented in the system codes.

- The ability to simulate natural circulation in both prismatic and pebble bed GCR designs must be demonstrated. Resistance to flow and bypass channels are unique to each design. Low-flow rates and conditions in which molecular diffusion dominate are difficult to simulate in systems codes, in which numerical diffusion can be a problem.

- Models and correlations for graphite oxidation and heat of reaction must be implemented into the codes and assessed. The heat release from graphite oxidation is approximately equal to decay heat and thus represents a significant contributor to heatup of the fuel.

- Molecular diffusion of air and water vapor through helium must be modeled. This will require development, implementation, and assessment of correlations for multicomponent diffusion coefficients and gas mixture properties.

- Graphite material properties and component emissivities must be determined and included in the code property routines.

These represent several of the major and most obvious research and infrastructure needs. The starting point for GCR thermal-fluid development should be a PIRT. Analysts should evaluate some initial PIRTs available from international studies to ensure that all important processes are included.

III.2.2.2 Nuclear Analysis

III.2.2.2.1 Background

The term “nuclear analysis” describes all analyses that address the interactions of nuclear radiation with matter. Nuclear analysis encompasses the analyses of (1) fission reactor neutronics, both steady-state 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, radiation detection, 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 HTGR reactor safety. Other sections of this document discuss nuclear analysis needs technically related to radiation protection, material safeguards, and out-of-reactor materials safety at the front and back ends of the HTGR fuel cycles (i.e., fuel enrichment, conversion, fabrication, storage, and shipping, and the storage, transport, and disposal of spent fuel).

III.2.2.2.2 Purpose

This section describes the nuclear analysis tools, data, and knowledge bases needed to support the staff’s safety review of HTGR applications. In identifying these R&D needs, the section also presents the issues related to nuclear analysis that affect reactor safety. The section begins with a brief discussion of the nuclear data libraries that are fundamental to all areas and applications of nuclear analysis and continues with a more detailed discussion of HTGR-specific nuclear analysis issues related to reactor safety.

All areas of nuclear analysis use the nuclear data libraries derived from files of evaluated nuclear physics data, such as the Evaluated Nuclear Data File, Volume B (ENDF/B) in the United States; the Joint European File in Europe; or the Japanese Evaluated Nuclear Data Library in Japan. The nuclear data files include fundamental data on radionuclide decay, as well as neutron reaction cross-sections, emitted secondary neutrons and gamma rays, and fission 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 be processed and used with advanced sensitivity and uncertainty analysis techniques. Such techniques can assist in the identification and application of appropriate experimental benchmarks for the HTGR-specific validation of nuclear analysis codes and methods.

Many of the processed nuclear data libraries in use today were developed in the 1980s or earlier. For example, the PBMR Pty. PBMR design relies on the German Very Superior Old Programs (VSOP) reactor physics code, with multigroup nuclear cross-section libraries derived in the 1970s and 1980s mainly from the evaluated physics data in ENDF/B-IV and ENDF/B-V. Until 2005, analysts were using similar pre-1990s cross-section libraries for preparing the LWR nodal physics data used by the NRC's reactor spatial kinetics code PARCS and for the lattice physics, criticality safety, fuel depletion, and shielding analysis sequences in the NRC's Standardized Computer Analyses 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 will require extensive reevaluation in the context of HTGRs and their fuel cycles and will continue to limit the implementation of modern nuclear analysis methods.

In response to a 1996 user need memorandum from NMSS, RES sponsored ORNL in an upgrade of its nuclear cross-section processing code system, A Modular Code System for Processing X-Sections (AMPX). Completed in 2004, the upgraded AMPX system is being used to create and refine new cross-section libraries that 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. AMPX can now likewise process cross-section libraries based on the latest evaluated nuclear data files in Joint European File-3 and Japanese Evaluated Nuclear Data Library-3. The recently completed AMPX upgrades and continued improvements to the Los Alamos National Laboratory's nuclear data processing code, NJOY, reflect significant progress in the production and testing of 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, including HTGRs.

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. Analyses use reactor neutronics codes to predict core-wide fuel burnup and the dynamic behavior of neutron-induced fission chain reactions in response to reactor control actions and system events. Under subcritical reactor conditions, where the self-sustaining fission chain reactions have been terminated by passive or active means, the decay of radioactive fission fragments and activation products becomes the dominant nuclear heat source.

The results from accident sequence analyses provide information that may be used in plant PRAs for assessing event consequences and their probabilities. Core neutronics codes, generally coupled with thermal-hydraulic (T/H) system codes, and in some cases with severe accident codes, will be needed to evaluate steady-state burnup and power distributions and the dynamic progression of operating events and accident sequences that involve reactivity and power transients. For accident sequences in which the self-sustaining fission chain reaction is terminated by active or passive means, the T/H and severe accident codes used in evaluating the thermal response of the subcritical system (e.g., maximum temperatures of fuel and reactor system components) must employ input data or algorithms that represent the intensity, spatial distribution, and time evolution of the decay heat sources. These data and algorithms must be the product of nuclear analysis.

The defining features of HTGRs include their use of fission-product-retaining CFPs; graphite as the moderator, reflector, and core structural material; and neutronically inert helium as the coolant. The PBR and PMR designs considered in this infrastructure assessment are fueled with low-enrichment uranium (LEU) (e.g., 4.6 to 19.9 percent ^{235}U) instead of the high-enrichment uranium (more than 90 percent ^{235}U) and thorium predominantly used in earlier HTGRs. These modular HTGR designs have long annular core geometries, with control and shutdown absorbers located in the graphite reflector regions and, in the case of some PMR designs, also in the core adjacent to the side reflectors. Maximum discharge fuel burnup levels in these LEU-fueled HTGR designs typically range from 80 to 150 GWd/t, significantly higher than the 45 to 60 gigawatt days per ton (GWd/t) fuel burnup levels reached in current LWRs with 5-percent maximum fuel enrichments. Therefore, in many respects, PBR and PMR designs have similar code modeling and validation issues for the prediction of reactor neutronics phenomena and decay heat generation.

With the use of hot graphite (e.g., 400 to 1000 °C) as moderator in HTGRs, the thermalized Maxwellian portion of the neutron spectrum generally spans higher energies than in LWRs. Thermal scattering into the 0.3 eV fission resonances of ^{239}Pu and ^{241}Pu and the 1.0 eV capture resonance of ^{240}Pu therefore becomes more significant, especially at the higher burnup levels of HTGR fuels. In fundamental terms, compared to light water, the power of reactor graphite to slow neutrons is about 20 times weaker and the thermal neutron absorption cross-section about 50 times smaller. In relation to LWRs, HTGRs have a neutron migration length typically about 5 to 10 times larger (e.g., ~30 centimeters (cm) versus ~6 cm) and a prompt neutron generation time about 50 times longer (e.g., ~1.5 milliseconds versus ~0.03 milliseconds). This means that (1) an HTGR will tend to have more tightly coupled spatial neutronics than would an LWR of similar dimensions and (2) a given super-prompt-critical reactivity addition would produce a much slower power rise (i.e., longer reactor period) in an HTGR than in an LWR. The fact that the solid graphite moderator is fixed in relation to the fuel means that changes in moderator density are small and have smaller reactivity effects than in LWRs.

Modular HTGRs also differ neutronically from LWRs in that significantly more neutron collisions occur throughout the epithermal energy range. Therefore, known uncertainties and imperfections in the evaluation and processing of epithermal resonance cross-section data and in the classical treatments of Doppler broadened and self-shielded resonance neutron absorption and scattering may be amplified in HTGRs in relation to LWRs, where the net integral effects of such epithermal data and analysis uncertainties are smaller and better quantified against integral data. The different neutronic heterogeneity effects in HTGR particle fuel versus LWR pellet-in-clad fuel give rise to further potential differences in the degree to which the effects of the various uncertainties and imperfections in resonance data and analysis methods tend to offset or cancel each other (as they seem to in LWRs) with regard to computed integral effects such as reactivity coefficients and the buildup and depletion of plutonium isotopes with burnup.

Other noteworthy neutronic characteristics of all modular HTGR designs include the following:

- The use of neutronically inert helium coolant effectively presents a large void fraction in the core and a resulting neutron streaming effect that must be treated in the diffusion theory core physics code models. Streaming in the large holes for control and shutdown absorbers is particularly important in the prediction of absorber worths.

- The inner and outer graphite reflectors provide sources of thermalized neutrons to the undermoderated annular core region, thereby giving rise to radial flux profiles characterized by pronounced thermal flux peaking and flux gradients at the core interface with the reflectors.

- The long neutron migration length in relation to radial annular core dimensions, material composition gradients, and computational node sizes gives rise to very high internodal neutron leakage and strong neutron spectral interactions among the core nodes as well as between core nodes and interfacing reflector nodes. These internodal spectral leakage effects are very much stronger than in LWRs and must be explicitly treated in the analysis of HTGR reactivity transients as well as power operations.

- Neutron and gamma ray energy deposition in the radial reflectors is substantial and can provide significant temperature feedback for the neutronic analysis of steady-state power profiles and reactivity transients.

Reactor neutronics and decay heat analysis issues unique to PBRs relate mainly to the use of multiple-pass online fueling in cores with unordered statistical packing of fuel pebbles with different burnups with locally varying bed porosity. PBRs have nuclear analysis issues associated with the potential for seismic compaction reactivity 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 or azimuthal gradients of pebble flow velocity resulting from the strong temperature dependence of pebble-to-pebble friction in hot helium (i.e., as seen in the THTR-300 pebble bed reactor). Section II.2.4 describes related research activities on the mechanics of pebble beds, including pebble flow, statistical packing, bridging, and seismic pebble bed compaction.

Physics analysis issues unique to PMRs relate mainly to the effects of burnable poisons, the possible combined presence of enriched “fissile” coated particles, unenriched “fertile” coated particles, and burnable poison particles in the fuel compacts, reactivity let down and control over batch-fueled burnup cycles, and the power-shaping effects of zoned fuel and poison loadings and batch-fuel shuffling. Near end-of-cycle in PMRs, there may be a particular need to quantify uncertainties in the positive reactivity contributions of the 0.3-eV fission resonance of ^{239}Pu and the precipitous drop in the capture cross-sections of ^{135}Xe and ^{149}Sm between 0.1 and 0.3 eV to the sign of the moderator temperature coefficient of reactivity at the normal operating temperature of the core graphite. The staff identified this issue in its MHTGR review activities during the late 1980s and early 1990s. Under certain conditions near end-of-cycle, when decay heat loads are highest, it may be possible that the moderator temperature coefficient becomes zero or positive during a positive reactivity insertion transient (such as rod withdrawal) until the core graphite heats up enough during the transient for the tail on the 1.0-eV capture resonance of ^{240}Pu to cause the sign of the moderator reactivity coefficient to go negative. For an LOFC occurring immediately after the reactivity transient, the initial fuel temperatures and core graphite temperatures could be much higher than those assumed for LOFC from normal full-power conditions. Higher values of core temperature and stored heat in the initial core conditions for the transient could lead to higher peak fuel temperatures during the core conduction cooldown.

Needed nuclear analysis infrastructure development includes the safety-related technical issues described below.

<u>Temperature and power coefficients of reactivity</u>. Analysts will need expertise, tools, and data to confirm that the safety analyses have treated appropriately the steady-state and transient reactivity feedback effects from core power variations and temperature variations in the fuel, moderator graphite, and reflector graphite regions. This includes parametric kinetics to provide the necessary understanding of how postulated variations of the respective reactivity feedback effects, including uncertainties, can impact reactor transient behavior in relation to safety margins in calculated peak fuel temperatures or other

appropriate safety-related metrics over a broad range of event scenarios. Based on sensitivity analyses and validation against representative experiments and tests, evaluations are needed to assess and account for computational uncertainties in the competing physical phenomena, including, for example, the positive contributions to the moderator temperature coefficients associated with ^{135}Xe and bred fissile plutonium.

The reported results from the benchmark validation of power reactivity coefficients against measurements in Japan's HTTR prismatic block test HTGR show inaccurate predictions, especially near full operating powers, that are significantly greater than 25 percent (i.e., the uncertainty limit targeted by the MHTGR designer in the early 1990s). Such discrepancies in reported code-to-data benchmark results indicate the need for further benchmark analyses and, potentially, for various improvements to the neutronics codes, their nuclear cross-section data libraries, and associated treatments of resonance absorption and scattering kinematics. Scoping studies will be needed to estimate the extent to which computed burnup- and state-parameter-dependent HTGR reactivity feedback coefficients (and other safety-related integral neutronic phenomena) can be changed by any or all of the following:

- applying estimated or bounding fluence-damage-induced changes to the neutron-scattering properties of graphite
- replacing the use of free-gas scattering kernels in fuel materials with approximated bound scattering kernels up to approximately 10 eV
- addressing known inaccuracies and inconsistencies in the measurement, evaluation, and processing of epithermal resonance cross-section data for important resonance absorber nuclides
- improving on the classical approximations used in Doppler broadening and self-shielding treatments of resonance neutron absorption
- replacing the nearly universally used but incorrect asymptotic assumption with more correct approximations of the kinematics of resonance neutron scattering
- applying available or estimated cross-section covariance data to perturbation-theory-based or direct sensitivity and uncertainty analyses of reactivity effects

The NRC should plan and perform activities in this area in coordination with domestic and international cooperative efforts on nuclear cross-section data evaluation and processing. These efforts include the DOE NGNP and AFCI-GNEP plans, the European Atomic Energy Community (EURATOM) HTR-N plans, and those recently pursued in related DOE international NERI projects and proposed followup activities involving S (alpha, beta) bound scattering data.

<u>Reactivity control and shutdown absorbers</u>. The tests and analytical evaluations for reactivity control and hot and cold shutdown may need to account for absorber worth variations over a range of normal and accident conditions throughout the burnup cycles in PMRs and in the transition from initial core to equilibrium core loadings. Particularly important is the evaluation of potential biases and uncertainties in predicted absorber worths through code validation experiments and modeling sensitivity and uncertainty studies. These efforts will need to consider the absorber worth variations caused by temperature changes in the core and reflector regions, stable and potential oscillatory xenon effects, variations or aberrations of pebble flow, potential changes in graphite neutron scattering caused by fluence damage (i.e., dose-dependent graphite scattering kernel), and the neutronic effects of accidental moisture or hydrogen

ingress.

Reactivity effects from moisture or hydrogen ingress. Although the absence of high-pressure, high-inventory water circuits in closed Brayton cycle systems may make the moisture-ingress issue less of a problem than in the steam cycle HTGR designs of the past, the effects from potential ingress of water, or of other hydrogenous substances, will likely still require evaluation for accident conditions in Brayton cycle HTGRs and in HTGRs used for hydrogen production and other process heat applications that involve hydrogenous chemicals. Analysts must consider the possibility of hydrogen diffusion through metallic heat exchanger barriers, in addition to barrier leakage and break scenarios. Neutronic effects to be evaluated include the potential for rapid reactivity addition from hydrogenous moderator entering the under-moderated core, the effects of the resulting neutron spectral softening on temperature coefficients of reactivity, the reactor kinetics effects of shortened neutron generation time resulting from faster neutron thermalization by hydrogen as compared to carbon, and the reduction of control and shutdown absorber worths caused by fewer neutrons migrating to the absorber element locations in the side reflectors or core periphery.

Maximum power pulses or transients. thermal-fluid-coupled spatial reactor kinetics analyses will be needed to evaluate load follow and control transients, as well as maximum reactivity power transients caused by credible upset events, such as overcooling, control rod ejection, rod bank withdrawal, shutdown system withdrawal or ejection, seismic pebble bed compaction, and moisture or hydrogen ingress. Of particular importance in the licensing review of HTGR designs is the need to identify, through safety analysis and risk assessment, any credible events that could produce a rapid power rise or pulse. The NRC should consider the estimated probabilities and power transient characteristics of such events in establishing any related plans or requirements for pulsed or transient accident testing and analysis of HTGR fuels (see Section III.2.3 of this report). For loss-of-cooling, passive shutdown events with failure of the active shutdown systems (i.e., ATWS), the delayed recriticality transient that occurs after 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 periphery of the core where the temperatures and xenon concentrations are lower.

Spatial xenon stability. The developers of modular HTGR designs cite maintaining axial xenon stability as a consideration that limits the maximum allowed height of the core (e.g., 11 meters (m) in the case of the PBMR Pty. PBMR). Analysis of azimuthal xenon stability may also be necessary to assess the potential for azimuthal decoupling of the annular core because of the isolating effects of the central reflector and any in-reflector control rods. Such evaluations will call for spatial kinetics simulations and the analysis of spatial eigenmode separations over a full range of power operating conditions and configurations.

Pebble burnup measurements and discharge criteria. PBMR Pty. states that selected FP 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 and fluence. Therefore, determining a suitable value for the discharge/recycle burnup criterion 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. Analysts will also need to consider burnup measurement calibrations and uncertainties. Furthermore, since a lower discharge burnup criterion will be applied to the initial charge of lower enrichment fuel pebbles than to the higher enrichment pebbles that are added in transitioning to an equilibrium core, the burnup measurement system must be able to distinguish between pebbles of different initial enrichments. Neutronics calculations will be needed to determine the higher neutron fluence experienced by lower enrichment pebbles in reaching the maximum burnup levels allowed in the transitional cores.

Pebble bed local hot spots. The results of melt-wire experiments conducted in the German AVR test reactor demonstrated the existence of unpredicted local hot spots during normal operation. Such hot spots affect the maximum normal operating temperatures of the fuel, which in turn can affect the failure fraction of TRISO CFPs during normal power operations and in eventual overpower transients and heatup accidents. These hot spots may arise from a combination of higher local power density (e.g., as a result of moderation effects near the reflector wall or from chance clustering of lower burnup pebbles), lower local bed porosity because of locally tight pebble packings, and reduced local helium flow resulting from the increase of helium viscosity with temperature. Whereas the slow evolution of LOFC heatup transients in a PBR will tend to wash out any effects of preaccident local flow starvation on subsequent peak fuel temperatures, the effects of higher local fission power densities and/or higher local fuel burnups will be sustained throughout the heatup transient as a result of higher local decay heat powers. Therefore, insights into the effects of decay-power hot spots, in particular, may be needed in evaluating the maximum fuel temperatures arising in pressurized and depressurized LOFC events.

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. Analysts may use computational studies with higher order methods, such as exact geometry and continuous-energy Monte Carlo neutronics codes, to investigate the distribution of power among assumed pebble clusters with different pebble burnups at various locations in the core. In the calculation of operating temperatures inside a pebble, the reduction of pebble power with pebble burnup may tend to be offset by the reduction of pebble graphite matrix material thermal conductivity with neutron fluence.

Decay heat power. Proposed HTGRs have maximum fuel enrichments and discharge burnup levels that are much higher and neutron energy spectra that are significantly harder than in current LWRs. Decay heat power generally increases with fuel burnup. The technical basis for decay heat power in LWRs (e.g., American National Standards Institute/American Nuclear Society 5.1) includes a database of calorimetry measurements and code-calculated fuel inventories of decaying FPs and actinides that have been validated in part against postirradiation radiochemical assays of LWR fuel. The calculated predictions of decay heat power in HTGRs will require a similar technical basis that is shown to be applicable to HTGR-specific fuel compositions and burnup conditions. The decay heat predictions used in HTGR safety analyses should account for validation uncertainties associated with the limited directly applicable measured data.

Pebble bed decay heat power distributions. As with pebble bed fission power densities (see related item above), each node in the core calculational model will contain pebbles with a broad range of decay heat power densities. Further computational studies may be needed to establish technical insights on acceptable modeling approximations (e.g., mesh averaging methods) and assumptions (e.g., local hot spots and power histories) for treating the spatial distributions of decay heat sources in PBRs.

Radionuclide decay before TRISO-coated particle fuel accident testing. To understand 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 long time intervals between fuel irradiation and testing. Such physical changes in the fuel particles 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, phase changes, 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. (This nuclear analysis issue relates directly to fuel analysis issues described in Section III.2.3 of this report.)

Physics of TRISO fuel irradiation in test reactors versus power reactors. The extensive use of various test reactors for the irradiation testing of HTGR TRISO fuels raises questions about the potential nonprototypicality of the neutron energy spectra, accelerated fuel burnup rates, and fuel temperature histories in the test reactors. Reactor-specific calculations of neutron fluxes and nuclide generation, depletion, and decay may be needed as 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. For example, ^{235}U and ^{239}Pu give substantially different yields of various FPs that potentially affect TRISO fuel performance. (This nuclear analysis issue relates directly to fuel analysis issues described in Section III.2.3 of this report.)

Another aspect of the fuel irradiation issue is the proposed further use of test reactor irradiations, followed by fuel radiochemical assays, to help validate the codes and methods used to predict HTGR fuel burnup isotopics. Here, in addition to the test reactor's different neutronic conditions, analysts should also consider the different burnup calculation modeling approaches and approximations necessitated by the different physical layouts in the test irradiations versus the HTGR core. Such considerations would further complicate the application of test reactor fuel irradiations and assays to the validation of HTGR fuel burnup codes as they are used in the safety-related analysis of reactor neutronics, decay heat power, and FP inventories potentially available for release.

Radiation shielding analyses for graphite and metallic damage fluence and radiation protection. Neutron irradiation substantially reduces the thermal conductivity of reactor graphite in a manner that varies with irradiation temperature. Therefore, code calculations of neutron fluence distributions in the graphite core and reflector regions will be important in the analysis of normal operations and in establishing initial conditions for the analysis of conduction cooldown events and other transients. Neutron irradiation further causes dimensional changes (mainly shrinkage) of reactor graphite. The resultant dimensional changes of graphite fuel and reflector blocks are expected to result in a widening of gaps between the blocks. These changes will need to be considered in thermal-fluid analyses (i.e., bypass coolant flows and conductive and radiative heat transfer) and in the shielding analyses for evaluating metallic component damage fluences and radiation protection. Fluence-induced damage to the reactor vessel was a potentially important design issue for the MHTGR. While the predicted fluence for the MHTGR reactor vessel was significantly less than that for LWR vessels, the possibility of embrittlement damage remained potentially significant because of the lower MHTGR vessel operating temperatures and the temperature dependence of fluence damage self-annealing. For current HTGR designs, fluence-induced vessel damage is reportly addressed in part by operating the reactor vessel close to LWR reactor vessel temperatures. Regardless, evaluation of graphite and metallic damage fluences, material activation, and radiation protection for HTGR designs will require validated radiation shielding code calculations.

III.2.2.2.3 Objectives and Associated Activities

The NRC safety R&D objectives are to establish and qualify the independent nuclear analysis capabilities and insights that may be needed to support the licensing evaluation of reactor safety analyses for PBR and PMR designs.

The Transport Rigor Implemented with Time-Dependent Operation for Neutronic Depletion (TRITON) lattice physics code and the PARCS spatial kinetics core simulator code constitute the current nuclear analysis component of the NRC's audit code suite for reactor systems analysis. The TRITON code, which is part of the SCALE-5 modular code system with its AMPX-processed ENDF/B-VI nuclear data libraries, performs detailed resonance shielding, lattice neutronics, and burnup isotopic calculations and processes the results into the burnup- and state-variable-dependent tables of spatially smeared and energy collapsed nodal nuclear data needed by PARCS. The PARCS code is currently coupled to the TRACE and RELAP5 system T/H codes for use in the steady-state and transient analysis of water-cooled reactor cores. The research objectives for HTGR nuclear analysis therefore include extending and assessing the capabilities of TRITON-PARCS as needed to perform coupled neutronic and thermal-fluid analyses of HTGR cores.

The primary neutron flux solver in TRITON is NEWT, a deterministic two-dimensional code that employs the extended step characteristic method to solve the discrete ordinates formulation of the neutron transport equation in many (e.g., 44 or 238) energy groups. The CENTRM code, which performs a deterministic continuous-energy solution of the one-dimensional neutron transport equation, provides resonance shielding for the NEWT calculation. The familiar ORIGEN-S code module provides the fuel-burnup nuclide transmutation solution in TRITON.

<u>Related technical information and R&D</u>. Listed and discussed below are past, ongoing, and planned R&D efforts that will benefit the development of NRC's HTGR nuclear analysis infrastructure.

- MHTGR preapplication review and research support efforts conducted by the NRC, as summarized in 1989 in NUREG-1338, "Draft Preapplication Safety Evaluation Report for the MHTGR," and in the draft update to that report completed in 1995. The 1995 draft update of NUREG-1338 references numerous supporting contractor and staff reports on various topics involving nuclear analysis (and many other technical areas) and includes a key set of supporting technical reports in its extensive appendices. The NRC project file for the MHTGR preapplication review (Project 672) also contains a collection of submittals by the preapplicant on a broad range of technical safety topics, including those involving nuclear design and analysis. Among the NRC research efforts supporting the MHTGR preapplication review were the development and use at ORNL of the MORECA code, which included early versions of the nuclear analysis (e.g., point kinetics, decay heat), as well as thermal-fluid, beyond-design-basis accidents (e.g., air ingress graphite oxidation), and FP release source term models now found in the current descendent code for HTGR analysis, GRSAC. (See also the related item below on GRSAC development.)

- Preapplication review interactions on the PBMR Pty. PBMR with Exelon during 2001–2002 and later directly with PBMR Pty. starting in 2005. The PBMR familiarization and review activities included several preapplicant presentations, white papers, and technical exchanges with the staff on a range of topics including issues related to PBMR nuclear design and analysis. In the

2001–2002 review activities with Exelon, the staff provided extensive written requests for additional information on a range of technical issues, including some involving nuclear design and analysis. Efforts to resolve applicable requests for information from that time have continued into the PBMR preapplication technical interactions that resumed in 2005.

- Completed RES-sponsored work including (1) upgrade of the ORNL AMPX code system for cross-section data library processing, (2) use of AMPX to develop initial sets of modern multigroup and continuous-energy nuclear cross-section libraries based on recent releases of ENDF/B-VI, (3) development of sensitivity and uncertainty analysis methods in the ORNL SCALE5 code system that use perturbation theory and available cross-section covariance data to evaluate the applicability of experimental reactor physics benchmarks and to help evaluate design-specific code biases and uncertainties from code benchmark results, (4) development of extended SCALE resonance treatment capabilities needed by TRITON (and also by related SCALE5 criticality calculation sequences) for analyzing the neutronic multiple heterogeneity of TRISO fuel particles and fuel elements in PBR and PMR HTGR lattices with multiple fuel and reflector regions, and (5) addition of r-theta-z geometry to the PARCS code to facilitate its use for steady-state and transient analysis of PBMR cores. In addition to RES-sponsored work, simplified SCALE4 resonance treatments for HTGR double heterogeneity, based on techniques for transforming spherical fuel particle and pebble geometries into neutronically equivalent cylindrical geometries, have been independently proposed and developed in South Korea.
- RES-sponsored work completed in recent years at ORNL to modify and improve the GRSAC code for analyzing operating, design-basis, and beyond-design-basis accidents in PBRs and PMRs and to train new GRSAC users from the NRC staff, DOE staff, several U.S. universities, and two national laboratories. Included in the recent GRSAC work for PBRs was the development of new code features for predicting how expected random or postulated spatial variations in pebble bed porosity and power density affect maximum coolant and fuel temperatures during normal operation. These GRSAC code features were recently applied to one of a set of IAEA benchmark problems (discussed below) with preliminary results showing that (1) expected spatial power and porosity variations through a PBR core can produce moderate hot-spot temperature increases (e.g., 15–20 °C) at steady-state full power, with the stronger contribution coming from porosity variations, and (2) expected random porosity variations through the core generally increase the computed overall pressure drop across the core compared to that calculated with conventional PBR models based on idealized uniform porosity.

- Recent and ongoing HTGR knowledge management efforts in RES, including work with ORNL to conduct expert interviews and develop a series of topical white papers prepared by senior technical experts with extensive experience in various areas of HTGR safety research and analysis. One of the expert white papers prepared to date includes a summary of issues related to nuclear analysis (as well as issues in various other technical areas) from the staff's past licensing review and oversight of Fort Saint Vrain and the staff's preapplication review and supporting research activities for the MHTGR design. Also included in the RES knowledge management efforts is staff participation in selected meetings on international and domestic R&D related to HTGR nuclear analysis, as noted in the next two items, as well as other HTGR-specific technical areas such as TRISO fuel performance, thermal-fluid analysis, and materials analysis.

- Recent RES staff participation in research coordination meetings of the IAEA's CRP-5, which, as discussed in the following subsection, includes the evaluation code-to-code qualification benchmarks and, most important, code-to-data validation benchmarks from measurements at the

Advanced Gas Reactor in the Kurchatov Institute in Russia (ASTRA) pebble bed critical experiment facility in Russia and from startup and safety demonstration tests conducted at the HTTR experimental PMR in Japan and the HTR-10 experimental PBR in China.

- Recent RES staff participation in a working meeting concerning the NEA PBMR-400 coupled neutronics/thermal-hydraulics transient benchmark of the Organization for Economic Cooperation and Development (OECD). Insights gained from observing and discussing the analyses of this code-to-code benchmark problem have given the staff a more detailed understanding of the additional code development and assessment efforts needed to give both TRITON and PARCS the proven functionality necessary for coupled neutronic and thermal-fluid transient analysis of PBRs and PMRs.

- Familiarization with existing codes and methods for HTGR neutronics and decay heat. Technical exchanges concerning the PBMR Pty. preapplication and staff participation in IAEA CRP-5 and related OECD NEA activities have given the staff a limited initial understanding of the German VSOP and TINTE codes used by PBMR Pty. The staff would require much more extensive familiarization with these reactor physics codes, associated decay heat algorithms, and licensing-basis analysis assumptions to complete review activities for the PBMR Pty. PBMR design certification. The NRC should factor into its overall R&D plans the insights and questions that would arise from increasing familiarization with such established HTGR nuclear analysis codes in support of this and related areas of the HTGR technical safety review.

- Scoping studies on HTGR reactivity coefficients and kinetics. By using available codes (e.g., GRSAC with point kinetics), simple parametric point kinetics analyses can provide insights into the effects of postulated variations or uncertainties in temperature reactivity feedback coefficients of the HTGR fuel, moderator, and reflectors on different transients. The staff is not yet aware of any completed or planned study results by others that the NRC could obtain.

- Higher order calculations to qualify reactor physics codes and methods for HTGRs. Using available higher order methods, such as exact geometry and continuous energy Monte Carlo codes (e.g., Monte Carlo “N” Particle (MCNP)/Monte Burns, MVP, MONK, SCALE/CE-KENO), exploratory analyses are being performed on selected nuclear analysis issues as described in this section. This approach entails the use of exact models and methods with progressive simplifications to understand detailed phenomena and evaluate the more approximate lattice physics and diffusion theory methods and models needed for coupled steady-state and transient reactor physics analysis. Higher order calculations have seen limited application in the NRC’s work at ORNL to qualify the HTGR-applicable double heterogeneity treatments recently developed in SCALE. Several studies outside the NRC have used MCNP in attempts to develop exact reference code solutions for the analysis of particle fuel double heterogeneity in HTGRs. These studies have shown that modeling simple cubic arrays of fuel particles introduces only very minor distortions (e.g., <0.4 percent) in k_{eff} in relation to more realistic models with more randomized particle distributions. While analysts have also made advances in applying temperature-dependent cross-section data in codes like MCNP, such studies have focused on k_{eff} results and have not yet evaluated particle modeling effects on computed reactivity coefficients. Monte Carlo code studies to date have also done little to develop reference models that represent the unordered statistical packings of fuel pebbles within a pebble bed and the more ordered pebble packings along the reflector walls that result in the well-known radial variations in effective porosity near the walls. One approach to achieving the latter reference models would entail using a pebble bed flow simulation code like PFC3D to provide input locations of fuel pebbles within a Monte Carlo code model.

Related domestic and international cooperation. Important areas of past, ongoing, and potential future domestic and international cooperation on HTGR nuclear analysis R&D include the following:

- Evaluation and documentation of the HTGR physics benchmark measurements performed in the early 1990s on the HTR configuration of the PROTEUS zero-power critical experiment facility at the Paul Scherrer Institute in Switzerland. Many of the HTR-PROTEUS benchmark data were analyzed as part of an IAEA cooperative research project and documented in TECDOC-1242, “Critical Experiments and Reactor Physics Calculations for Low Enriched HTGRs,” issued in October 2001. In 2004–2005, DOE supported INL and Argonne National Laboratory in their work with the Paul Scherrer Institute through the OECD NEA International Reactor Physics Experiment Evaluation Project (IRPhEP) to further evaluate and document additional data from the high-temperature reactor configuration of the Proteus Critical Experimental Facility (PROTEUS) experiments for inclusion in NEA-1765, “International Handbook of Evaluated Reactor Physics Experiments.”

- Evaluation and documentation of existing and proposed HTGR physics benchmark data from the ASTRA zero-power critical experiment facility at the Kurchatov Institute in Russia. Recent ASTRA experiments sponsored by PBMR Pty. are presently being evaluated in IAEA CRP-5. A TECDOC report to be completed in 2007 will document the results of that evaluation. Parallel NEA-coordinated efforts are underway and planned in IRPhEP to incorporate further evaluation of these experiments in the NEA-1765 handbook. PBMR Pty. has tentatively planned further experiments at ASTRA, and the Commissariat a l’Energie Atomique RAPHAEL project has tentative plans for further experiments at ASTRA to include configurations with a solid central reflector. ASTRA experiments to date have been conducted at room temperature only. In addition, the HTR-N program within the HTR Technology Network of EURATOM has proposed further cooperative work at ASTRA to perform temperature coefficient and hot condition rod worth benchmarks by nonnuclear heating of the test zone possibly to temperatures as high as 700 °C.

- Evaluation of relevant historical HTGR physics benchmark data. In the “Next Generation Nuclear Plant (NGNP) Research and Development Program Plan,” issued in January 2005, INEEL reported that an assessment was completed in 2004 of potentially relevant sources of HTGR physics benchmark data for NGNP. These sources included data from the British Magnox, AGR, and early DRAGON HTGR programs, as well as 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. Also included were the HTR-10 in China and the HTTR and VHTRC test facility in Japan. The assessment concluded that the HTR-10, HTTR, and VHTRC were the most promising facilities to provide physics benchmark validation data for the anticipated LEU-fueled PBR and PMR design concepts for the NGNP reactor. The fact that the early HTGRs operated in the United Kingdom, the United States, and Germany used primarily non-LEU fuels makes their test and operating data less applicable to code validation for modular HTGR designs. Other graphite-moderated reactors, including the first Chicago pile, the Hanford production reactors, and the British Magnox power reactors, have used much higher ratios of graphite to fuel than those in HTGRs to achieve the much softer neutron energy spectrum needed to allow fueling with unenriched natural uranium.

- <u>Evaluation and documentation of existing and new HTGR physics benchmark data from the HTR-10 test reactor in China.</u> China's INET has contributed to IAEA CRP-5 a number of code-to-data benchmarks from the startup tests and safety demonstration testing program at HTR-10. These include (1) steady-state neutronic benchmarks of measured initial criticality and measured control rod worths in the initial cold core and fresh full-power core, (2) a coupled neutronic and thermal-fluid benchmark of steady-state temperature distributions measured at full power at 22 locations around the pebble bed core periphery, coolant inlet plenum, outlet plenum, reflectors, and fuel discharge chute, and (3) two coupled neutronic and thermal-fluid transient benchmarks of measurements from two HTR-10 safety demonstration tests, "Loss of Primary Flow Without Scram" and "Control Rod Withdrawal Without Scram." TECDOC-1382, "Evaluation of HTGR Performance: Benchmark Analysis Related to Initial Testing of the HTTR and HTR-10," issued in November 2003, evaluated and documented the initial criticality and control rod worth experimental benchmark results. The second and final CRP-5 TECDOC report, which will be issued in 2007, will document completed or ongoing evaluations of the remaining HTR-10 benchmarks. In 2005, NEA started coordinating specific IRPhEP efforts to further evaluate and document selected HTR-10 physics benchmarks for future incorporation into the NEA-1765 handbook. In 2004, the EURATOM HTR Technology Network program reported that it was including INET as a partner in its activities and that, within its HTR-N project, it was arranging benchmark cooperation on further planned HTR-10 safety demonstration tests and a proposed HTR-10 melt-wire monitor experiment similar to the one performed in Germany's AVR. Further consideration has also been given in this NEA cooperative context to performing state-of-the-art code analyses to better understand the unexpected AVR melt-wire test results and to determine to what extent such code analyses can produce computed spatial hot-spot or postulated short-term global heatup results that are consistent with the AVR melt-wire results.

- Evaluation and documentation of existing and new HTGR physics benchmark data from the VHTRC PMR heated critical experiment facility and the HTTR in Japan. JAERI, which is now the Japan Atomic Energy Agency (JAEA), contributed several code-to-data benchmarks to IAEA CRP-5, including (1) neutronics code benchmarks against measurements of several HTTR initial critical configurations, excess reactivity, scram reactivity, and isothermal temperature coefficients, and (2) measured HTTR thermal-fluid benchmarks of transient loss of offsite power and of steady-state operation of the vessel cavity cooling system. TECDOC-1382 evaluated and documented all of these HTTR benchmarks. An earlier IAEA cooperative research project evaluated neutronic benchmarks from VHTRC experiments with nonnuclear heating up to 200 °C and documented them in TECDOC-1249, "Critical Experiments and Reactor Physics Calculations for Low-Enriched HTGRs," issued in October 2001. The HTTR and VHTRC physics benchmark experiments evaluated by IAEA are also recent additions to the NEA-1765 handbook. The staff is not aware of any past, active, or planned international cooperative activities involving (1) the HTTR power coefficient benchmark mentioned in the above discussion of reactivity feedback coefficients (this experimental benchmark appears to have been evaluated and analyzed only in Japan), (2) other recent or continuing testing programs in HTTR, or (3) additional experiments in VHTRC.

- Code-to-code benchmarks for coupled HTGR transient analysis and code-to-data benchmarks for HTGR reactor physics validation. Starting in 2005, the NEA has been coordinating the "PBMR-400 Coupled Neutronics/Thermal-Hydraulics Transient Benchmark" as a simplified code-to-code standard benchmark problem for coupled code testing and qualification. Among the reactor spatial kinetics codes being applied to the benchmark problem is the NRC's PARCS code, which is exercised here in coupling with the German pebble bed thermal-fluid code THERMIX-DIREKT. Key among the simplifications in this benchmark is the use of prescribed common tables of two-group static and dynamic nodal macroscopic cross-section data along with a simplified two-dimensional system geometry model with prescribed material thermal properties. A similar PBMR-400 code-to-code benchmark problem being analyzed in IAEA CRP-5 lacks these simplifications, so, taken together, the two benchmarks will provide a basis for identifying the sources of differences among coupled code results. In 2003 and 2004, the NEA IRPhEP started the previously mentioned review and evaluation of existing HTGR neutronics benchmarks for inclusion in its international handbook of reactor physics experiments. In late 2005, NEA began planning in coordination with the EURATOM HTR-N project to evaluate experimental benchmarks from the former testing programs of the AVR pebble bed test reactor and the KAHTR pebble bed critical experiment facility at Jülich.

- HTGR high-burnup isotopics for neutronics and decay heat validation. In 2004, the HTR-N project of EURATOM reported plans to perform radiochemical isotopic assays on a fuel pebble irradiated to high burnup in the high flux reactor (HFR) in the Netherlands. As noted previously, the neutron-physical and computational modeling nonprototypicality of such test reactor irradiations in relation to actual fuel irradiations in HTGRs should be considered in applying such data to the validation benchmarking of HTGR fuel burnup isotopic calculations as they are used in the analysis of reactor neutronics, decay heat power, and radionuclide inventories.

III.2.2.2.4 Application of Research Results

Fundamental to reactor safety analysis is the ability to predict the fission and decay heat sources for normal operation, AOOs, DBEs, and BDBEs. The research activities described above, combined with needed NRC-sponsored R&D, will support the staff's capability to independently assess HTGR safety performance and nuclear analysis issues associated with HTGR designs. Through the nuclear analysis R&D activities, the staff will develop its technical insights in these areas, and the NRC will establish and qualify its independent analysis tools and capabilities. The R&D activities include the investigation and analysis of validation issues and modeling approximations to inform the staff's evaluation and treatment of potential biases and uncertainties in the computed steady-state and transient HTGR fission power and decay heat sources.

III.2.2.3 Accident Analysis

III.2.2.3.1 Background

For LWRs, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," issued in 1991, and subsequent reactor risk studies performed by the NRC and industry showed that accidents involving severe core damage, coupled with containment bypass or containment failure, dominated public risk from reactor operation. These accidents result from sustained loss of core cooling and can release substantial quantities of radioactive FPs into the environment. Analysts needed the ability to model the progression of severe accidents and estimate releases of FPs into the environment to quantify LWR risk and to address LWR severe accident issues. However, for the proposed HTGR designs, which do not include a pressure-retaining low-leakage containment building, the events expected to involve the largest accident source term, radionuclide release, and offsite radiological dose and risk are those events that result in the loss of coolant pressure boundary. For HTGRs, as is the case for LWRs, loss of pressure boundary events may occur in the AOO frequency region, the DBE frequency region, and the BDBE frequency region. Further, among the HTGR loss of pressure boundary events, those that can lead to chemical attack, such as events resulting in air or water ingress, are potentially the most severe HTGR accidents in terms of core structure degradation and FP release. They are also the most challenging to analyze because of the more complex accident phenomena. As described below, the NRC has developed several codes to model FP transport and release accidents, including those involving complex accident behavior and progressions.

The NRC's LWR severe accident codes, based on many experiments performed in the 1980s following the Three Mile Island 2 accident, include MELCOR, SCDAP/RELAP5, CONTAIN, VICTORIA, and IFCI. As NRC's consolidated LWR accident code, MELCOR can model most aspects of a severe accident including thermal-fluid analysis, core melt progression, FP release and transport in the reactor system, and containment. For LWRs, the United States and other nations have performed many experiments to develop 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. The TEXAS code is used to analyze fuel coolant interaction phenomena.

As part of NRC's review of HTGRs, the development of FP transport and source terms will be important to the assessment of several policy issues, such as the containment functional performance requirements, the required size of emergency planning zones, and, potentially, the selection of design-basis accidents and beyond-design-basis accidents. There is a need for data and modeling methods for HTGR materials and configurations. Research will be needed to support both the development of infrastructure to perform confirmatory analysis and to identify and resolve many of the policy issues based on source term.

III.2.2.3.2 Purpose

HTGR events that lead to FP release from the reactor pressure boundary must be modeled. Source term (i.e., FP release and transport) analysis methods using codes, such as MELCOR, will be needed to estimate the magnitude, composition, and timing of FP release from the reactor pressure boundary, from the containment or confinement system, and to the environment.

HTGR DBE and BDBE analysis and source term analysis will also be needed to support the development of limiting sequences and to confirm the applicants' safety analyses. Therefore, FP release and transport data and accident progression analysis codes and the expertise to apply them will be needed both to estimate consequences of events that contribute significantly to the integrated plant risk and to evaluate specific HTGR safety and technical issues.

For HTGRs, and other non-LWRs that significantly differ from operating and advanced LWRs, event sequences and the FP transport and release process (e.g., graphite dust release from a PBR pressure boundary) can also differ. In HTGRs, FPs may be released from coated particle defects and heavy metal contamination from manufacture, diffusion during normal operation and accidents, release from CFP failures during normal operations and accidents, and by lift-off of the FP plated out on cool metallic surfaces during normal operation.

The risk from FP releases from an HTGR pressure boundary may be associated with AOOs, DBEs, and BDBEs. The FP release transport from the fuel for these event categories can occur as a result both of diffusion through intact CFPs and release from failed CFPs. Technical expertise in the area of fuel FP transport and behavior during normal operation and licensing-basis events is needed to assess HTGR event consequences and overall plant risk. 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 on surfaces within the pressure boundary and on surfaces within the containment or confinement system. Aerosol deposition occurs through a variety of mechanisms, such as gravitational settling, thermophoresis, and diffusiophoresis. Data and modeling on FP release and transport resulting from the effects of fuel oxidation and convective flow within core and out of the pressure boundary will be needed.

III.2.2.3.3 Objectives and Planned Activities

MELCOR has most of the capabilities needed to analyze beyond-design-basis accident issues. However, for HTGRs, MELCOR needs modification because of the differences between LWR and HTGR designs. These include differences in the fuel, core, and reactor internal structure design and differences in the materials for the fuel, core, and core support structures and coolant. To support code modifications, a PIRT will be needed to delineate the important HTGR accident phenomena and factors, including FP release and transport phenomena to be modeled by the code, as well as the experimental data required for model development and assessment. Proposed modifications identified at this time are described below, together with an activity to assess MELCOR in terms of the available experimental data, existing MELCOR codes that have been modified for HTGR accident analysis, and other codes. For example, INL has modified the MELCOR code for analyzing PBR and PMR accidents, including air ingress accidents, and ORNL has developed and used the GRSAC code for HTGR accident analyses, including air ingress accidents. The NRC has obtained the INL HTGR version of the MELCOR code and the ORNL GRSAC code and will use them to evaluate the required code capabilities and modeling needs for HTGR accident analysis.

<u>Modify the code to incorporate needed models and data</u>. Before application to HTGRs, the MELCOR code will require modification to incorporate available models/data. Modifications such as those described below will allow modeling of the FP release from the core and deposition in the reactor coolant system and containment.

- 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 AGR fuel (e.g., spherical fuel pebbles, block/prismatic fuel configurations). The modifications will include the effects of fuel and core temperature, air or steam oxidation, and burnup on FP release and transport.

- 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 oxidation model should be able to predict self-sustaining oxidation (i.e., graphite burning). Oxidation models for both PBR fuel (i.e., graphite matrix material) and core structural, moderator, and PMR fuel blocks (i.e., nuclear-grade graphite material) will be needed. In addition, the model should consider smoke and particulate formation.

- Update materials properties models. Fuel and structural material components in MELCOR must include graphite. Key properties include conductivity, specific heat capacity, and diffusion coefficients. The models should consider graphite/fuel degradation and relocation, as well as the strength and integrity of core supporting structures. The core modeling capabilities should allow for both PBRs and PMRs.

- Improve numerics. MELCOR numerics will need to use longer time steps to carry out reasonable execution times for slowly developing accidents. This may involve changing the numeric solver for MELCOR to implement the semiexplicit two-step algorithm.

Evaluate the need for additional FP deposition/transport experiments and models. When model implementation in the MELCOR code is complete, analysts should evaluate the code against available integral FP transport experiments. Also, analysts should prepare input decks for selected advanced reactor designs and demonstrate code capabilities for selected performance scenarios.

Assess the code against available experimental data and other codes. To achieve this objective, a literature review is needed to identify HTGR experiments on FP release during high-temperature transport and deposition conditions within the reactor pressure boundary and containment or confinement under accident conditions. Because the release of non-FP aerosols from the core increases FP aerosol deposition, 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 staff will assess the need for additional experiments.

Apply code to specific advanced reactor design. The results of the above R&D will be a version of the MELCOR integrated severe accident code capable of analyzing the progression of AOOs, DBEs, and BDBEs (including severe accidents) in either PBRs or PMRs. This version of MELCOR could be used with input from other codes that establish initial conditions and boundary conditions (e.g., initial FP distribution within the pressure boundary surfaces, initial failed versus intact coated particles) to independently confirm an applicant’s source term and FP release calculations, identify the potential need for safety enhancements or other regulatory action, provide guidance for NRC reviewers, and provide the technical basis for safety analysis requirements and acceptance criteria. The HTGR version of the MELCOR code must be able to effectively model the key HTGR accident progression phenomena for event sequences that could potentially result in substantial release of FPs from the core (e.g., sustained air ingress, graphite burning, defective or degraded fuel), including event sequences resulting from postulated malevolent acts.

The results of the database work will be used to develop and assess FP release, transport, 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 support the staff’s independent evaluation of the applicants’ designs with respect to AOOs, DBEs, and BDBEs (including severe accidents).

Section III.2.3 of this report describes the national and international activities associated with HTGR fuel R&D. As described in Section II.2.3, only historical German data for PBR-type coated particle fuel are currently available for use in developing and assessing MELCOR models for HTGR fuel FP release, behavior, and transport in the reactor system under postulated accident conditions. Fuel FP release and transport data from intact and failed particles under accident conditions for modeling HTGR-specific applications will require HTGR-specific fuel fabrication data together with fuel irradiation and accident condition testing data. These data will be available in the future upon completion of the HTGR design-specific fuel development and qualification programs.

III.2.2.3.4 Application of Research Results

The NRC will apply this R&D to develop and validate analysis tools needed to evaluate the behavior of the HTGR designs under postulated accident conditions, as well as any resulting releases and transport of radioactive FPs within and outside the reactor pressure boundary. The agency will need this capability to support its review of HTGR applications and its evaluation of policy and safety issues, such as containment functional performance requirements and emergency preparedness requirements for HTGRs.

III.2.3 Fuels Analysis

III.2.3.1 Background

An HTGR core will contain several billion CFPs dispersed within several hundred thousand graphite fuel elements (i.e., fuel pebbles, fuel compacts). These CFPs are the primary safety barrier to the release of radionuclides to the environment. Thus, for HTGRs, the CFPs provide the principal containment function preventing release of radionuclides to the environment and are relied on to effectively perform this function during normal operation, AOOs, DBEs, and BDBEs. That is, HTGRs ensure radiological containment by retaining and containing FPs where they are generated, within the CFPs.

The basis of the HTGR safety case and safety analysis is the expectation and the requirement that CFPs will have a very low failure rate within the licensing-basis envelope. Accordingly, an HTGR applicant must assure with high confidence that only a very small fraction (i.e., 10^{-5} to 10^{-4}) of the CFPs within the core will fail as a result of the combined effects of manufacturing defects, operational service conditions, and any accident conditions. The applicant must demonstrate in the reviews of HTGR fuel design, fabrication, irradiation testing, and safety that the fuel containing the CFPs which will be loaded into the HTGR core will meet this expectation and requirement with a high confidence level. Because HTGR fuel qualification test programs are supposed to demonstrate that fuel performance requirements are met, HTGR designers have proposed that HTGRs not be required to have a conventional high-pressure, leaktight containment.

Worldwide, the design, manufacture, testing, and operating experience with HTGR coated particle fuels is still relatively limited compared to that with LWR fuels. Additionally, CFP performance analysis models and methods are not yet fully developed and benchmarked for new coated particle fuels. Experience has also shown that CFP design and fabrication processes are major determinants of in-reactor fuel performance. Accordingly, HTGR fuel qualification is primarily based on irradiation testing and accident condition testing of prototypical production fuel to demonstrate the in-reactor performance (i.e., CFP failure fraction) and FP release behavior of the production fuel. Test conditions are intended to conservatively envelop the core conditions, especially with respect to those parameters that have a strong effect on the degradation and failure of CFPs. Fuel qualification tests are intended to qualify the fuel design and manufacture and must be conducted even if the performance of a particular fuel design has been demonstrated previously. Performance testing will be needed to provide data on CFP performance and FP release data over the range of operating, transient, and accident conditions defined by the licensing basis.

HTGR fuel designers and vendors have not proposed to assure safety via specific design or safety limits on the fuel. Rather, fuel designers and vendors propose that safety be assured via event frequency-based radiological dose limits for the plant and that safety analyses codes, which calculate fuel failure as a basis for FP release and accident source term and dose, incorporate internal checks to confirm that the calculated fuel service conditions (e.g., burnup, fuel temperature) not exceed the maximum values for which the fuel was qualified.

HTGR applicants are expected to propose that the safety analysis use a mechanistic scenario-specific accident source term rather than a conservative bounding accident source term to calculate accident consequences. The source term is based on the calculated release of radioactivity from all FP sources within the fuel element. These include releases from intact and failed particles, as well as releases from heavy metal contamination located outside coated particles. The failed particle fraction is the sum of the initial CFP defect fraction from manufacture, the CFP failure fraction that occurs during plant operations, and the CFP failure fraction that occurs as a result of a transient or accident. The overall FP release from the fuel in the core will be based on the sum of releases from all intact and failed CFPs in the core plus releases from the fission of heavy metal contamination outside of the CFPs. The mechanistic source term will include the transport, retention, and release of FPs within the fuel element, transport, retention, and release within structures and surfaces within the primary pressure boundary, and transport, retention, and release from the reactor confinement structure.

The source term calculation will require a sound technical basis, which depends on a sufficient database and modeling of fuel FP transport and release. In this regard, FP transport through intact CFP layers is specific to the design and manufacture of the fuel. Because of the limited operating experience and database for FP transport, testing of HTGR production fuel and fuel materials is needed to develop and benchmark the FP release and transport models to be used in the mechanistic accident source term calculations over the range of applicable HTGR plant operating and transient conditions and postulated accident conditions. The development of fuel-specific FP transport testing, including quantifying uncertainties, is essential for HTGR plant licensing.

Extensive HTGR CFP irradiation (operational) testing has been conducted in Germany, the United Kingdom, the United States, Japan, Russia, the Netherlands, and China to understand and characterize the behavior and safety performance of CFPs during normal operation, operational transients, and accidents. The fuel development and qualification irradiation testing philosophy of the international programs has generally focused on demonstrating acceptable and predictable fuel performance within the specified design, manufacture, operating conditions, and accident condition envelope applicable to a particular HTGR fuel design. Far less testing has explored conditions beyond the licensing-basis envelope to quantify the margins of conditions wherein larger and potentially unacceptable increases in particle failure rates begin to occur for qualified production fuel. Such failure rate increase data will be needed to quantify these margins.

In-reactor CFP performance is highly dependent on both coating product specifications and coating process specifications. The current state of CFP manufacturing technology and fuel characterization techniques does not allow fuel irradiation performance to be assured solely on the basis of fuel product specifications. Accordingly, at this time, selected fuel fabrication process characteristics are also specified to provide CFP properties that result in acceptable irradiation and accident performance. These critical fuel process specifications are generally considered proprietary information. Finally, the design of certain components of the chemical vapor deposition coating furnace are also known to be critical to the characteristics and irradiation performance of the manufactured fuel. Ensuring that the design of these components remains fixed and that the critical components are properly maintained is essential to the qualification, quality, and performance of the manufactured fuel over the fuel supply lifetime. These aspects of CFP manufacture are significant embedded elements of the fuel that is produced for the fuel qualification program. A full understanding of the specifications for fuel process parameters, fuel product characteristics, and fuel manufacturing equipment components that are critical to ensuring in-reactor fuel performance is needed to provide the technical basis for any regulatory controls (e.g., license conditions for the fuel fabrication facility, inspection procedures) established for HTGR fuel manufacture.

Fuel service conditions that are known to have an important effect on CFP performance (e.g., failure, FP release) include fuel temperature, kernel burnup, coating layer fast fluence, and particle power. Factors affecting CFP performance during accidents (following the degradation effects of operations) include particle temperature and oxidation. Events that can increase fuel particle temperature include core heatup events (e.g., caused by loss of normal core cooling), core power increase events, and significant local reactivity insertion events. Chemical attack (e.g., oxidation because of air ingress) of the fuel element and, potentially, the CFPs directly can also affect CFP accident performance.

The mechanistic prediction of CFP performance and the mechanistic prediction of the accident source term will require capabilities in a number of interfacing technical areas. These include (1) nuclear analysis prediction of fuel burnup, fast fluence, thermal fluence, and fuel particle power; and (2) thermal-fluid analysis of core operating temperature distributions, core accident temperature distributions, and core multimedia flow distributions (for fuel oxidation during postulated air ingress events). The FP release rates from the fuel during normal operation and licensing-basis accidents are key inputs to the accident source term analysis, documented in Section III.2.2.3 of this infrastructure assessment.

There is a need to assess whether fuel design limits and/or safety limits will be required to ensure that the failure rate and the FPs released from the CFPs are within the safety analysis basis. The assessment should consider the need for such limits during normal operation, design-basis accidents, and beyond-design-basis accidents. If limits are imposed, the applicant will have to demonstrate that it meets the limits. If a fuel safety margin is applied, it would be in addition to the fuel design margin. The fuel safety margin would provide margins to the consequence limits even if the conditions were outside the design-basis limiting conditions (e.g., fuel design and manufacturing specifications, maximum allowed fuel operating temperature, maximum allowed fuel burnup). The applicant would develop any safety limit aspects. However, a complete assessment of the safety limit would likely require NRC research since HTGR designers and applicants generally do not address conditions that go substantially beyond the licensing-basis conditions.

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 may also be needed 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 appear below. This discussion recognizes the considerable worldwide research on HTGR fuels with CFPs that has been conducted over the last 30 years or is currently ongoing. The NRC's HTGR fuel performance analysis R&D needs should capitalize on this body of work in establishing the knowledge, data, and tools needed to form the technical basis for fuel-related policy decisions and the technical review of HTGR applications. The past, current, and planned HTGR research provides a base and context for assessing which R&D activities should be pursued to fill infrastructure gaps.

The JAEA believes that silicon carbide (SiC) will not perform adequately as the primary barrier to FP transport from the fuel kernel because of the higher required fuel burnup and higher core/fuel operating temperature of the VHTR. The JAEA believes that the more severe VHTR service conditions will unacceptably increase SiC thermal degradation and SiC corrosion because of the increased release of palladium from the kernel.

<u>Past HTGR Fuels Research</u>. SECY-03-0059 documents past research into the safety of HTGR fuels.

<u>Current HTGR Fuels Research</u>. The following is a summary of the current national and international research efforts on HTGR fuels.

U.S. DOE Fuels Program. DOE is currently implementing an advanced HTGR fuels technology R&D program plan, referred to as the Advanced Gas Reactor Fuel Development and Qualification Program. The fuel development program plan directly supports the design, development, and licensing of the DOE very high temperature gas reactor (i.e., the NGNP). The INL is leading and coordinating the AGR fuel development program. An objective of the program is to achieve a fundamental scientific understanding of TRISO-coated particle fuel behavior and performance. The program includes the conduct of fuel irradiations and accident condition tests to obtain data for fuel behavior modeling (e.g., particle integrity and FP transport) and to advance the understanding of the relationship between fuel manufacturing processes, fuel product characteristics, and fuel performance. Another objective of the program is to produce high-quality, high-performing TRISO-coated particle fuel on a consistent basis, similar to that achieved in Germany in the 1980s.

The AGR fuel program is also intended to provide a baseline of fuel qualification data and analytical tools to support NGNP demonstration plant licensing and operations analysis. A major objective of the program is to demonstrate that the fuel, which is designed, fabricated, and tested by the program, has the required safety performance for core conditions that are prototypical of the NGNP. The program also seeks to extend acceptable fuel performance to high burnup (>220 K MWD/MTU), core exit temperatures above 1000 °C, and fuel temperatures as high as 1300 °C. If SiC restricts operations (i.e., burnup, temperature) because of its performance capabilities, DOE may pursue alternative coatings, such as zirconium carbide (ZrC). The AGR fuel program is also intended to reduce uncertainties associated with NGNP fuel manufacture and qualification.

The AGR fuel technology R&D program includes fuel manufacture, fuel and materials irradiation, fuel accident condition testing, fuel postirradiation examination (PIE), fuel performance modeling, and FP transport modeling. The FP transport data and models are intended to support the technical basis for the mechanistic accident source term calculation for NGNP licensing. The fuel qualification program plan involves irradiation and accident conditions that bound VHTR operating and accident conditions for either a PBR or a PMR. The VHTR service conditions for burnup, fast fluence, temperature, power density, and peaking factors exceed both the earlier German HTR service conditions and the fuel qualification requirements for the PBMR Pty. PBMR. At this time, the reference TRISO-coated particle fuel for the NGNP VHTR involves a UCO kernel rather than a UO_2 kernel. However, UO_2, the German, and PBMR Pty. PBMR reference kernel composition remain under consideration for the initial core fuel load. The DOE irradiation testing program plan includes several fuel designs (variants) to yield a better understanding of the relationship between the fuel fabrication process specifications, the fuel product specifications (particle attributes), and fuel irradiation performance and accident condition performance. DOE has reported that an established capability to conduct all required fuel product characterization techniques (e.g., anisotropy, crystal/grain size, porosity). The first planned irradiations, which will be conducted at the Advanced Test Reactor in Idaho, are scheduled to begin in late 2006, with the final fuel qualification irradiations scheduled to be completed by 2015. The program is intended to develop fuel performance data and information needed to support NRC licensing of a VHTR.

European Commission Fuel Program. The European Union is sponsoring an HTGR fuel technology development program referred to as the high-temperature gas-cooled reactor fuel technology (HTR-F). The program supports an integrated VHTR R&D project. The objectives of HTR-F initially were to reestablish past German knowledge in the areas of TRISO CFP design, fabrication, and testing. The program R&D plan includes assessing the performance of previously manufactured German fuel pebbles with TRISO CFP and UO_2 kernels at very high burnup; developing an analytical code capability and supporting materials property data for predicting TRISO CFP behavior under irradiation and accident conditions; and retrieving, evaluating, and cataloging historical fuel irradiation data with the aim of constructing a searchable TRISO CFP database. R&D elements include fuel manufacturing technology, quality control methods, irradiation testing, accident condition testing, PIE, fuel performance analysis modeling, and FP release modeling. Near-term accident condition (heatup) tests involve previously irradiated German archive fuel pebbles and use accident testing equipment previously utilized at the Jülich Research Center. The planned tests include fission gas and fission metal release measurements for simulated accident heatup up to 1600 °C and safety margin tests up to 1800 °C. To address a fuel testing method issue documented in SECY 03-0059, accident condition tests will compare the particle failure fractions for a traditional “ramp-and-hold” fuel heatup with particle failure fractions for an accident transient time-temperature heatup curve. Additionally, fresh German archival fuel pebbles will be irradiated to very high burnup at the Petten HFR to obtain information on the burnup capabilities (i.e., margins) of the German reference fuel. Recently, the program objectives were expanded to include fuel performance improvements, including developing CFPs capable of meeting the more challenging VHTR core service conditions, including the development of an advanced ZrC coating as a replacement for the SiC coating.

French Fuel Program. The Commissariat a l'Energie Atomique and AREVA have an ongoing laboratory-scale HTGR CFP fabrication R&D program to establish a French HTGR fuel fabrication capability. The program builds on the German reference TRISO fuels knowledge and performance experience and involves fuel particle fabrication, quality control methods, fuel irradiation and material irradiation, fuel accident condition (heatup) testing, PIE, and fuel performance modeling. A goal of the program is to qualify coated particle fuel for VHTR service conditions. Other objectives of the program are to produce TRISO-coated particle fuel; develop the capability to construct a production-scale fuel fabrication facility; develop the capability to adjust fuel fabrication parameters to optimize the performance of reference UO_2 kernel fuel; investigate UCO kernel fuel as an alternative to UO_2 kernel fuel; investigate the use of a ZrC layer as the principal FP retention barrier as an alternative to an SiC layer; and conduct irradiation testing, accident condition testing, and PIE as input to fuel performance assessment and fuel performance model development. Fuel irradiations will be conducted at the French OSIRIS materials test reactor to verify fuel performance and FP retention characteristics. The French are also developing a fuel performance code, referred to as the Advanced Thermal Analysis Software (ATLAS), to support the VHTR safety analysis. ATLAS is a finite-element code which uses a statistical approach to determine the failed particle fraction, based on the fuel irradiation history, accident conditions, and statistical distributions of the particle characteristics (e.g., dimensions, properties) from manufacture. Researchers will also use the ATLAS code to calculate the FP release from failed particles during normal operations and accident heatup conditions. The calculated releases are to be input to an accident analysis code to calculate FP transport and release from the core (i.e., source term). Fuel samples will be irradiated at a material testing reactor to provide irradiated fuel performance data for fuel fabrication development, to qualify fuel, and to support the development and validation of fuel performance and fuel FP transport models in the ATLAS code.

German Fuel Program. In Germany, current HTGR fuel technology R&D is limited. Activities involve participation in the European Union HTR-F project and fuel technology development with other countries (e.g., PBMR Pty., China, and Korea). Other activities include participating in IAEA CRP-6 CFP performance analysis code benchmark calculations and documenting a description of the Jülich Research Center PANAMA fuel performance code.

Japanese Fuel Program. The Japanese have successfully manufactured HTGR UO_2 coated particle fuel for fuel qualification testing and initial fuel loading of the JAERI HTTR. Currently, the Japanese are acquiring fuel operating experience on a significant scale via fuel irradiation and burnup accumulation within the HTTR core. JAEA is now supporting HTGR fuel technology R&D. Current fuel-related R&D is focused on advanced fuels for VHTR applications. For this application, TRISO CFP R&D is aimed at developing a ZrC layer to replace the SiC, which JAEA believes is not suitable for the higher fuel operating temperatures within the VHTR core. The agency has already conducted irradiations to limited burnup with follow-on PIEs to assess ZrC performance. Irradiations to higher burnups are planned for 2007. JAEA is conducting its ZrC R&D, including irradiation and code modeling, in cooperation with the DOE international NERI projectsthrough ORNL (irradiation, PIE) and INL (modeling). JAEA is also initiating an R&D program for its prismatic block fuel element to replace the reference graphite fuel pin in block concept with a fuel compact in block design. JAEA is also planning irradiation tests to develop improved materials property data for SiC and pyrolytic carbon (PyC) layers under high neutron irradiation.

Korean Fuel Program. Korea has implemented a broad-scope HTGR fuel technology R&D program to support its Nuclear Hydrogen Production Technology Development and Demonstration Project. This project will use a VHTR. Fuels R&D includes ongoing fuel fabrication processes and methods development for coated particle fuel including manufacture of fuel kernels, development of PyC and SiC layer coating processes, manufacturing quality control methods, irradiation and accident condition experiments, and fuel performance analysis models and methods. The implementation schedule for R&D through technology demonstration extends over the next 10 years. Korea is also pursing advanced particle fuels development involving ZrC. Laboratory-scale fuel R&D is currently underway.

Chinese Fuel Program. The People's Republic of China has a relatively low-power operating pebble bed reactor (HTR-10) with fuel manufactured by INET. In advance of initial HTR-10 operation, INET has been conducting fuel qualification irradiation testing and accident condition testing with increasing burnup at a materials test reactor in Russia. The fuel for the qualification tests and the fuel in the initial core load met or exceeded all fuel quality specifications. The ongoing fuel qualification irradiation program in the Russian test reactor also involves periodic heatup of the in-reactor test fuel pebbles to simulate accident heatup conditions. HTGR fuel technology development is focused on plans for a reference fuel design for a commercial HTGR power reactor, increased fuel production capacity to supply the first HTR-PM demonstration pebble bed reactor module, and large-scale fuel production to supply fuel to support the anticipated rapid increase in the deployment of HTR-PM modules in China for electric power generation. HTGR fuel technology R&D for the HTR-PM includes development of a process to produce a high-quality, high-performing ZrC coated layer to replace the SiC layer in TRISO CFPs and development of a process to coat pebble fuel elements with SiC/SiO_2. The latter is intended to significantly reduce fuel pebble oxidation during a postulated core air ingress accident. Plans are also underway to conduct irradiation testing of the HTR-PM reference fuel.

Indonesian Fuel Program. The National Nuclear Energy Agency of Indonesia began its HTGR fuels program in 1997. The program involves TRISO fuel fabrication technology development in the areas of fuel kernel production and SiC coating process studies (using conventional chemical vapor deposition particle coating methods) and fuel particle performance modeling. The long-term goal of the program is to develop a capability to produce high-quality and high-performance TRISO-coated particle fuels.

Indian Fuel Program. India has established a TRISO-coated particle fuels R&D program for operation in a compact HTR. The fuels technology development program is currently focused on achieving production capabilities for UO_2, ThO_2, $(U,Th)O_2$, and $(U,Pu)O_2$ kernels and the optimization of the coating process for ZrC coatings.

Massachusetts Institute of Technology Fuel Program. The Massachusetts Institute of Technology 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 collaborative research areas include the calculation of temperature distributions inside pebble fuel elements; fuel performance models to predict CFP thermal-mechanical behavior, including failure of CFPs and finite element models of CFPs; and failure models based on fracture mechanics to predict CFP failure probability.

PBMR Pty. Fuel Program. The PBMR Pty. reference fuel design is the same as the German reference fuel design. Near-term HTGR TRISO fuel technology R&D includes kernel and particle coating fabrication processes, manufacturing quality control methods, advanced fuel characterization and PIE techniques, irradiation qualification testing, and development of a fuel performance analysis code. The company is conducting fuel manufacturing R&D in a fuel development laboratory and will use the results to develop the fuel fabrication methods for the planned large-scale PBMR Pty. fuel production facility. PBMR fuel is to be manufactured using feed materials, processes, and equipment which are equivalent to those used for the manufacture of the German reference fuel. Plans are underway to conduct fuel irradiation tests using unirradiated German reference archive fuel and subject it to operating conditions and accident conditions expected for the PBMR design. Two irradiations with production fuel spheres are planned. Four spheres will be irradiated in the Russian IVV-2M reactor and five spheres at the Petten HFR. These tests, scheduled to run in parallel starting in the latter half of 2006, are intended to contribute to an empirical database which demonstrates that the German fuel elements made with the German fuel manufacturing process will perform satisfactorily in conditions simulating PBMR operating and 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, the company is planning to develop and establish the process, equipment, and production facilities to be used to manufacture the production fuel for the PBMR demonstration reactor module and later commercial PBMR modules. PBMR Pty. had planned to have fuel from its pilot fuel fabrication facility available for irradiation testing in 2006.

International Atomic Energy Agency Fuel Program. IAEA has established CRP-6, “Conservation and Application of High Temperature Gas Cooled Reactor (HTGR) Technology: Advances in HTGR Fuel Technology,” to encourage the development and sharing of information on HTGR coated particle fuel technology. The scope of activities includes HTGR fuel R&D in the areas of fuel fabrication, fuel characterization and advanced quality control techniques, techniques for operational fuel performance monitoring, fuel irradiation testing and fuel accident condition testing, code-to-code and code-to-data operational benchmark calculations and accident condition benchmark calculations, and spent fuel management. Additionally, CRP-6 includes the compilation of national nuclear regulators’ safety perspectives related to HTGR fuel and related R&D.

III.2.3.2 Purpose

The purpose of the regulatory research infrastructure assessment in the area of HTGR fuel performance analysis is to assess the knowledge, data, and tools that the NRC would need to review an HTGR application in the area of fuel performance and fuel FP release. The staff requires this infrastructure development to support its review of applications such as the NGNP VHTR, the PBMR Pty. PBMR, and the GA GT-MHR. The infrastructure development addresses both PBR and PMR fuel forms and UO_2 as well as UCO fuel kernels. The assessment scope is for TRISO-coated particles with an SiC layer and addresses core operating and accident conditions representative of either a commercial advanced HTGR or demonstration VHTR application. The fuel infrastructure assessment includes design and fabrication, irradiation testing, accident condition testing, fuel operational performance and accident performance, FP transport modeling and related data needs, and performance (behavior, integrity) modeling and related data needs.

The purpose of the assessment is to identify the HTGR fuels R&D needed to provide the technical basis for the staff to review the fuel aspects of an HTGR application and make sound regulatory decisions. The staff needs this R&D so that it will have the requisite capability to review TRISO CFP technology. This capability includes the knowledge, data, and tools needed to independently and authoritatively assess an applicant's technical and safety basis for fuel quality, safety performance, and FP transport. The staff must have an independent capability to predict fuel performance and FP release during normal operation, design-basis accidents, and potential severe accidents and to support an independent prediction of event-specific and design-specific accident source terms. To acquire the needed review capability, the NRC will need to capitalize and utilize past, ongoing, and planned worldwide CFP R&D program activities.

III.2.3.3 Objectives and Associated Activities

The overall goal of the fuels analysis infrastructure assessment area is to identify the R&D that the NRC will need to develop the technical basis for licensing requirements, regulatory requirements, or regulatory guidance or SRPs for HTGR fuels; assess whether HTGR fuels raise safety or technical issues; quantify HTGR fuels safety margins and failure thresholds; quantify significant HTGR fuels safety issues involving large uncertainties; develop the capability to independently confirm important HTGR fuels analysis results; and support the analysis of HTGR policy issues linked to HTGR fuel performance.

III.2.3.3.1 Fuel Design and Fabrication

TRISO CFP irradiation performance and accident performance (i.e., CFP failure probability, FP release) depend in large part on the CFP design and manufacture. TRISO CFPs fabricated in Germany in the 1980s, using German fabrication equipment and process specifications, had rates of irradiation-induced failure several orders of magnitude lower than similar TRISO CFPs fabricated in the United States in the 1990s using U.S. fabrication equipment and process specifications. A critical comparative analysis of these and other experiences has led to the common understanding that the fabrication process and process specifications determine the properties and property variations of the CFP constituents, and these in turn drive the in-reactor performance of the fuel particles. As a result, current worldwide CFP fabrication technology R&D for application to near-term HTGRs seeks to repeat the German performance experience by replicating the German fuel manufacturing process methods and specifications. Additionally, significant basic CFP technology R&D is aimed at reverse-engineering the German particles to understand the structure, morphology, and properties of the particle constituents with the objective of replicating, controlling, and eventually forward-engineering and custom-designing CFP characteristics with custom fuel fabrication processes in order to attain high performance for the more challenging service conditions of a VHTR.

Since CFPs provide the main barrier to FP release in an HTGR and because fuel fabrication determines fuel performance in-reactor and during accidents, the staff will need to acquire sufficient knowledge and information of HTGR fuel fabrication process and quality controls to ensure adequate regulatory oversight and controls of HTGR fuel fabrication facilities. Also, compared to the manufacture of LWR fuel, HTGR fuel manufacture may require enhanced regulatory oversight and controls to ensure the requisite fuel characteristics and quality over the life of the fuel supply. This might involve fuel fabrication technical specifications and vendor core inspections. The need for enhanced regulatory oversight and controls is a significant safety issue and a potential Commission policy issue. Alternative measures might involve reactor coolant activity monitoring and periodic end-of-life fuel accident simulation testing. However, these alternatives have technical, safety, and regulatory advantages and disadvantages.

The objective of infrastructure development in the area of HTGR fuel design and fabrication is to provide NRC staff with the insights, information, and indepth knowledge of contemporary HTGR fuel fabrication, including the critical process parameters, critical product parameters, and quality control measures that are vital to achieving the required fuel quality and fuel performance over the life of the fuel supply for the plant.

A knowledge infrastructure for HTGR fuel fabrication is needed to ensure appropriate regulatory oversight of HTGR fuel fabrication facilities, including input to potential technical specifications for fuel manufacture, inspection programs for HTGR fuel manufacturing facilities, and HTGR safety reviews.

The objective of fuel design and fabrication infrastructure development is to provide the staff with needed expertise in this area, including the following:

- the important fuel manufacturing process parameters and fuel product parameters and their associated specifications and their relationship to fuel performance during in-reactor irradiation and accidents

- the methods used to characterize and measure fabricated fuel properties, including those associated with the kernel, coating, and matrix and fuel element

- the manufacturing process controls and product controls that keep variation of the fuel characteristics within allowable tolerances, including the chemical vapor deposition coater characteristics (e.g., size, levitation gas, coating gas distributions)

- the statistical methods, product sampling analysis methods, and statistical acceptance criteria

III.2.3.3.2 Fuel Irradiation Testing

The in-reactor performance of CFPs depends on the fuel design and manufacture as well as fuel irradiation conditions. Accordingly, fuel irradiation performance data generally apply to a specific fuel design and manufacture. However, fuel irradiation performance data for a specific fuel design and manufacture can also provide generic behavior and performance insights for closely related fuel designs involving similar manufacture. Application of performance test data and insights from the irradiation of a particular fuel design and manufacture to another fuel design and manufacture must be fully assessed.

Fuel qualification testing by HTGR design organizations involves irradiation test conditions that envelop plant licensing-basis irradiation conditions. Historically, design organizations have not conducted qualification and irradiation tests significantly beyond the bounds of the predicted fuel licensing-basis irradiation conditions for such parameters as the maximum fuel temperature, burnup, and fast fluence applicable to the core design and fuel management scheme. To understand and quantify fuel performance margins (e.g., change in CFP failure rate, change in FP release), fuel irradiations significantly beyond the maximum fuel licensing-basis conditions (i.e., maximum design conditions) will be needed.

Additionally, FP transport (e.g., radionuclide diffusion) through intact particle layers depends on the microstructures of the individual layers of the coated particle. Diffusion rates are generally equivalent for similar layer microstructures, and microstructures are generally similar for equivalent manufacturing processes. However, this will require confirmation via fuel irradiation testing for each reference fuel design and fuel manufacturing process.

The fuel irradiation testing aspect of the infrastructure assessment assumes that the vendor will conduct all fuel testing necessary to support the license-basis conditions. Such fuel testing would be expected to address all conditions of the licensing basis (e.g., core maximum operating temperature, fuel design burnup, fast fluence, particle power). These tests will need to include sufficient quantities of CFPs (i.e., fuel elements) to establish an adequate statistical database and will need to confirm the expected CFP failure mechanisms for the fuel design manufacture and operating/irradiation conditions via PIEs. It is expected that irradiation testing will be conducted with fuel fabricated by the fuel production equipment, processes, specifications, and quality control methods that will later be used for the manufacture of the production fuel supply over the plant lifetime.

The NRC will also need expertise in the conduct of HTGR fuel irradiation testing equipment, procedures, and practices, as well as an understanding of testing limitations and potential areas of testing oversights and omissions. This expertise will give the staff a sound basis for judging the acceptability of vendor fuel qualification program equipment, methods, quality assurance practices, and the like.

The objective of fuel irradiation testing infrastructure development is to provide the staff with the needed data and insights in this area, including the following:

- irradiation performance (i.e., particle failure rate) and qualitative behavior (e.g., failure mechanisms) of the CFPs and constituent component materials for use in developing coated particle fuel performance analysis models and benchmarking CFP performance analysis codes via PIE of the irradiated fuel

- values of physical and mechanical properties (e.g., layer strength, creep, elastic modulus) of irradiation-dependent fuel particle coating layers and changes in these physical and mechanical properties for use in developing CFP models and benchmarking CFP performance analysis codes,

- FP transport (e.g., diffusion) and release within and from the fuel to develop and benchmark coated particle fuel FP transport (e.g., diffusion) models

- irradiation performance (i.e., particle failure rate) over fuel service operating conditions (e.g., temperature, burnup, fast fluence) and fuel performance margins (i.e., increase in particle failure rate) for conditions beyond the core operating design limits

- issues associated with fuel irradiation testing (e.g., the effects of accelerated testing[1] on particle behavior/failure) and data measurement uncertainties to be applied to the staff's review of vendor fuel irradiation test programs and fuel performance safety analyses

- uncertainties in predicted fuel performance and FP transport because of data and modeling uncertainties

Specific Fuel Irradiation Testing R&D Needs. The following are areas for further R&D related to specific fuel irradiation testing needs:

- Fuel irradiation test data for reference HTGR fuels on FP transport through the coatings of intact particles will support the development of fuel FP transport models and benchmark HTGR fuel FP transport codes.

- Fuel irradiation test data for reference HTGR fuels on FP transport from failed particles (e.g., FP gas release from kernels) will support the development of HTGR fuel FP transport models and to benchmark HTGR fuel FP transport codes. The CFPs would be designed to fail early in life (e.g., there would be no buffer layer, or an SiC layer would have a small drilled hole).

- Fuel irradiation test data for reference HTGR fuels on FP transport through matrix material of compacts/pebbles will support the development of HTGR fuel FP transport models and will benchmark fuel FP transport codes.

- Fuel irradiation test data for reference HTGR fuels on CFP failure rates versus burnup, temperature, and fast fluence within operational limits will be used in developing CFP performance models and benchmarking CFP performance analysis codes and to provide fuel for accident condition testing and PIE.

- Fuel irradiation test data for reference HTGR fuels on CFP failure rates versus burnup, temperature, and fast fluence that exceed operational limits will be used to understand and quantify operating margins, develop CFP performance models, benchmark CFP performance analysis codes, and conduct accident condition testing and PIE. Irradiation conditions (temperature, burnup, fluence) would be sufficient to cause CFP failures and include power and thermal cycling associated with in-core fuel life cycle.

- Fuel irradiation test data for reference HTGR fuels will be used to evaluate CFP failure rates versus burnup, temperature, and fast fluence for irradiations that simulate power/temperature cycling of fuel particles because of design-specific, time-dependent fuel movement or core management practices.

- PIE information on reference fuels will provide insights into the following:

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1 This issue is documented in SECY-03-0059. Recent studies with a mechanistic CFP performance analysis code showed that, depending on the particle design and the degree of acceleration involved, accelerated irradiation testing may be highly conservative or may be nonconservative. The study concluded that fuel particle performance models are needed to obtain an accurate understanding of the effects on performance of a specific level of acceleration on a specific fuel design. Alternatively, close to real-time irradiations could be needed to resolve this testing issue.

- the applicable fuel particle failure mechanism(s)

- FP gas and metal diffusion in kernels for FP transport models

- FP gas and metal diffusion in coatings for FP transport models

- FP diffusivities and sorptivities in graphite matrix materials for FP transport models

III.2.3.3.3 Fuel Accident Condition Testing

Accident condition (e.g., heatup, oxidation) fuel performance (i.e., particle failure fraction) depends on the fuel design and manufacture and the prior irradiation conditions (i.e., fuel preconditioning), as well as the specific accident conditions. Accordingly, data on fuel performance during accident condition testing are generally specific to a particular fuel design, manufacture, and irradiation history. Accident condition testing can obtain data on the particle failure rate during the accident, the FP transport in the fuel during the accident, and the particle behavior and failure mechanism(s), via PIE, for fuel modeling.

The objective of the fuel accident condition (i.e., safety) testing infrastructure development is to provide the staff with needed data and insights in this area including the following:

- fuel performance (i.e., particle failure rate) during accident conditions (temperature rise, oxidation,[2] reactivity insertion[3]) to support the development and benchmarking of coated particle fuel performance analysis models and codes

- fuel FP transport and release during accident conditions to support the development and benchmarking of fuel FP transport models

- fuel performance (increase in particle failure rate) for postulated accident conditions (e.g., temperature rise) that exceed the most severe predicted accidents to determine coated particle fuel performance "margins" and to support the development and benchmarking of coated particle fuel performance models and codes

- fuel behavior and failure mechanisms via PIE of fuel subjected to accident conditions

- accident condition heatup testing methods issues (e.g., "ramp and hold" temperature profile versus accident "transient" temperature profile[4]) and uncertainties (e.g., measurement) that are relevant to fuel qualification test program reviews and fuel performance safety reviews

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2 This issue is documented in SECY-03-0059. Limited oxidation testing of irradiated fuel pebbles was conducted in Germany. The DOE AGR fuel development program plan includes air and steam ingress safety testing of irradiated fuel.

3 This issue is documented in SECY-03-0059.

4 This issue is documented in SECY-03-0059. To help resolve this issue, the European Union HTR-F program plans to conduct heatup tests with the ramp and hold method and the transient temperature increase method on German archive fuel pebbles from the same fuel lot and with the same irradiation history.

Specific Fuel Accident Condition Testing R&D Needs. The following are areas for further R&D related to specific accident condition testing needs:

- Data on CFP failure rates versus accident temperature for fuel heatup conditions up to the maximum predicted design temperature for the most severe heatup accident will be used to develop CFP performance models, benchmark CFP performance analysis codes, and conduct PIE.

- Data on CFP FP release failure rates versus accident temperature for fuel heatup conditions up to the maximum predicted design temperature for the most severe heatup accident will be used to support the development and benchmarking of fuel FP transport models.

- Data on CFP failure rates versus accident temperature for fuel heatup conditions above the maximum predicted design temperature for the most severe predicted heatup accident will be used to assess fuel performance safety margins and develop and benchmark CFP performance models and performance code. Peak accident heatup temperature would be sufficient to cause CFP failures.

- Data on CFP failure rates for oxidation conditions that simulate the most severe air ingress accident heatup event are needed.

- Data on CFP failure rates for traditional “ramp and hold” accident condition heating test method versus transient simulation accident condition heating test method are needed.

- PIE information on fuel with failed CFPs will be used to understand or confirm particle behavior and the mode(s) of failure.

III.2.3.3.4 Fuel Performance Analysis Modeling

Coated particle fuel failures during normal operation and accidents are the major contributors to the HTGR accident source term. Historically, HTGR accident source term calculations have been based on empirical models of experimental data. These models seek to correlate particle failure rate as a function of factors such as fuel temperature, burnup, and fast fluence as well as local accident conditions. Each fuel (i.e., each design and manufacturing process) involves separate irradiation and accident condition failure data and empirical models.

Mechanistic fuel performance analysis codes are also being developed to simulate particle behavior, failure, and FP release for general application to a variety of fuel designs and manufacture. Codes for mechanistic fuel performance analysis have the potential to be an independent tool for assessing the performance of alternative fuel designs and manufacture (fuel properties) irradiation and accident conditions. Analysts are using these codes for design and design analysis, as well as for safety analysis and safety evaluations. These fuel performance analysis codes can assess the design of fuel qualification test programs, fuel operation different or outside the fuel qualification program conditions, and the performance of fuel manufactured outside the fuel fabrication acceptance specifications (e.g., dimensions and properties). Finally, some HTGR design organizations are seeking to link the output of neutronic thermal-fluid core safety analysis codes to the input of a mechanistic fuel performance analysis code to calculate core-wide particle failure rates for the mechanistic accident source term calculation.

The objective of fuel performance analysis modeling infrastructure development is to give the staff an analytical tool to support its review of an HTGR application. This tool should be capable of conducting the following tasks:

- calculate HTGR coated particle fuel performance (e.g., behavior, particle failure) for all licensing-basis conditions

- calculate the structural, thermal, and chemical interactions within the CFP including internal pressure, particle layer stresses and strains, external oxidation, applicable mechanism(s) of particle failure, and particle failure probability versus changes in CFP design (e.g., UO_2 or UCO kernel), manufacture (e.g., manufacturing variability of product dimensions and properties, defects), irradiation time, and local environmental and accident conditions

- evaluate fuel performance in the absence of extensive data on design and manufacturing, specific irradiation test, accident condition test, and PIE

Specific Fuel Performance Analysis Modeling R&D Needs The following are areas for further R&D related to specific fuel performance analysis modeling needs:

- A review of potentially available alternative HTGR coated particle fuel performance analytical codes that take a "first principles" approach to modeling coated particle fuel behavior and particle failure, together with the NRC HTGR fuels PIRT for key phenomena to be considered on the fuel performance assessment model is needed. This review should include selection of the fuel analytical performance code platform with the greatest potential for success in meeting NRC independent capability needs to achieve the above-stated objectives.

- A literature search for published information and data on the properties of HTGR coated particle fuels should be conducted.

- Design- and manufacture-specific fuel particle coating layer properties versus irradiation from fuel irradiation tests should be evaluated.

- Integral code-to-code and code-to-data benchmark data need to be developed.

- Fuel irradiation testing data and accident condition testing data are needed to develop and refine behavior models of the selected analytical fuel performance code.

- Sensitivity analyses will be used to assess the effects of varying significant coated particle fuel phenomena on particle performance

- Fuel performance benchmark calculations of historical fuel irradiation and accident condition tests using best available inputs for physical and chemical properties should be performed.

- Predictive fuel performance (e.g., particle failure) calculations of HTGR applicant-planned fuel irradiation and accident condition tests using best available physical and chemical properties data will be compared to actual fuel irradiation and accident condition test results.

III.2.3.3.5 Fuel Fission Product Transport Modeling

For HTGRs, radiological containment comprises a number of concentric mechanistic barriers to the release of radionuclides to the environment. Some mechanistic barriers are designed to be high-integrity, relatively impervious physical barriers (e.g., the SiC layer within the CFP and the reactor coolant system pressure boundary) while mechanistic barriers contribute to radiological containment by inhibiting, slowing down, or delaying the transport of radionuclides to the environment. However, for HTGRs, the most important mechanistic barrier is, by far, the CFPs within the fuel element graphite matrix.

The case for HTGR safety depends on the premise that the mechanistic barriers associated with the fuel are highly effective in preventing significant radionuclide transport and release to the other outer concentric mechanistic barriers. Quantifying the performance of the fuel mechanistic barriers to the transport of radionuclides is the single most important element in determining the magnitude of the HTGR mechanistic accident source term. Since the accident source term can involve both prompt and delayed radionuclide transport and release, the effectiveness of the fuel mechanistic barriers must be understood and quantified in terms of the fuel radionuclide release and transport during normal operation as well as release and transport during licensing-basis accidents. It is expected that FP transport properties will depend on the CFP constituent layer microstructures which in turn depend on the specific fuel manufacturing process parameter values. To the extent that these manufacturing parameter values are significantly different from the historical manufacturing parameter values (i.e., those of the German reference fuel), it is likely that the radionuclide transport (e.g., diffusion) rates will also be significantly different. This will require fuel-specific fuel FP transport data and fuel-specific fuel FP transport modeling. The staff will need these data and models for its review of HTGR design-specific accident source term calculations and for its independent calculation of design-specific offsite and onsite doses for licensing-basis events.

The objective of fuel FP transport modeling infrastructure development is to give the staff the data needed to support its review of the technical basis for the event-specific mechanistic source term calculation and an independent analytical capability in this area. Infrastructure needs include the following:

- FP transport data for the fuel kernel, particle coatings, and matrix graphite to support the development and benchmarking of coated particle fuel FP transport models and methods, for all licensing-basis conditions, including intact and failed particles

- FP transport data for the fuel kernel, particle coatings, and matrix graphite to quantify uncertainties in the fuel FP transport models and data, including intact and failed particles

<u>Specific Fuel FP Transport Modeling R&D Needs</u> The following are areas for further R&D related to specific fuel FP transport modeling needs:

- Fuel irradiation test data for reference HTGR fuels on FP transport through the coatings of intact particles will support the development of fuel FP transport models and will benchmark HTGR fuel FP transport codes.

- Fuel irradiation test data for reference HTGR fuels on FP transport from failed particles (e.g., FP gas release from kernels) will support the development of HTGR fuel FP transport models and will benchmark HTGR fuel FP transport codes. The CFPs would be designed to fail early in life (e.g., there would be no buffer layer, or the SiC layer would have a small drilled hole).

- Fuel irradiation test data for reference HTGR fuels on FP transport through matrix material of compacts/pebbles will support the development of HTGR fuel FP transport models and will benchmark fuel FP transport codes.

- PIE information on reference fuels will provide insights on the following:

  - FP gas and metal diffusion in kernels for FP transport models

  - FP gas and metal diffusion in coatings for FP transport models

  - FP diffusivities and sorptivities in graphite matrix materials for FP transport models

III.2.3.3.6 Fuel Performance Monitoring

HTGR plant designs are expected to include operational fuel performance monitoring systems. These systems indirectly monitor fuel performance by measuring coolant activity. The coolant activity monitoring systems are intended to provide defense-in-depth against higher than expected increases in FP releases from the fuel during normal operation or transients. These systems are intended to detect elevated fuel particle failure rates during normal plant operations which could indicate that the fuel performance is approaching or is outside the bounds of the licensing-basis assumptions about fuel condition. Coolant activity monitoring systems can also ensure that the fuel is operating within the integrity/degradation parameters seen in the fuel qualification irradiation tests.

HTGR coolant activity 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 from 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 planned HTGR coolant activity monitoring systems have the capability to detect significant latent fuel particle failure conditions (i.e., weak fuel or weakened fuel). If latent fuel particle failure conditions occur, they might not result in elevated fuel particle failures. In such a situation, the coolant activity monitoring systems might not be a reliable means to prevent higher than expected particle failure fractions during a postulated accident. Latent failure conditions might exist as systematic undetected errors in either the quality of the manufactured fuel. Such latent failure conditions might result from an out-of-specification particle layer coating rate (not detected during manufacture) resulting in reduced SiC layer coating strength, or SiC layer failures that occur during operations and remain undetected because the inner and/or outer PyC layers remain intact as a result of undetected elevated particle temperatures. Coolant activity monitoring systems might not detect such latent failures. If the failures are undetected and no actions are taken before an event, a core-wide particle failure fraction above the predicted level might result.

The objective of the fuel performance monitoring infrastructure development is to improve the staff’s ability to assess the efficacy and effectiveness of HTGR coolant activity monitoring systems in detecting fuel degradation in support of the staff’s review of an HTGR application. Research needs include the following:

- evaluating the capability of coolant activity monitoring systems to detect weak fuel caused by out-of-specification manufacture or fuel weakened by operating conditions above that allowed by the design and assessing 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

- determining whether fuel that is weak or weakened because of specific credible manufacturing or operating conditions would be detectable by elevated coolant activity monitoring systems during normal operations to preclude unacceptable fuel failure rates during heatup accidents

Specific Fuel Performance Monitoring R&D Needs. The following are areas for further R&D related to specific fuel performance monitoring needs:

- Available historical irradiation test data, operational data, and accident simulation test data should be reviewed for evidence of the capability of core condition monitoring systems to detect weak or weakened fuel.

- Fuel irradiation tests with conditions significantly higher than design operating conditions followed by accident condition heatup tests should be conducted. Fission gas release measurement data for both tests will be the principal basis for assessing the potential for inducing and detecting weakened fuel that would fail during an accident.

- Sensitivity analyses with a fuel performance code will be used to assess whether fuel that is weak (in various ways) because of manufacturing errors would consistently result in detectable increases in fuel failure rates or whether there are conditions of weakness because of manufacture that would not be evident as increased failures until the accident condition.

III.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 provide essential insights on the FP source term for normal operation and accident conditions. The source term information is key to systems analysis, accident analysis, and consequence analysis and will play a significant role in supporting regulatory decisions in several 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 decisionmaking on fuel performance, including the acceptability of an applicant's fuel irradiation program.

III.2.4 Materials Analysis

III.2.4.1 Background

A key HTGR safety R&D area is the behavior of metallic and graphite components that provide structural, barrier, or radionuclide retention functions during normal and off-normal conditions. A sound technical basis must be available for evaluating expected lifetime and failure modes of RPV materials and components whose failure would result in a loss of core geometry and/or an ingress of air, water, or steam into the reactor coolant pressure boundary. High-temperature materials are required to maintain core geometry, adequate core cooling, access for reactivity control and shutdown systems, and, in the case of a PBR, a defueling route. The materials analysis section emphasizes the need for research to establish a technical understanding of the metallic and graphite components under high-temperature operating and accident conditions. The integrity of the pressure boundary and structural components is linked to nearly all other safety research areas and, in fact, determines the useful life of the plant.

The design and licensing approaches for modular HTGRs and the DOE NGNP VHTR will rely extensively on the plant-specific PRA. Information and data from the materials performance area are needed as input to the plant PRA. Because failure probability data for modular HTGR or VHTR design-specific components under HTGR-specific service conditions are not yet available from operational experience, the needed information must be developed or validated from materials research on potential degradation processes and quantifications of their rates of progression under realistic service conditions. The evaluation of component service life, safety margins, and behavior under accident conditions requires information on spatial and temporal variations as well as the typical values of inputs such as temperature, pressure, helium coolant composition, and fluence determined by reactor systems and fuel analyses. Outputs of the materials component analyses should provide insights for stable configurations of the core, available operating time, and temperature, pressure, fluence, and helium coolant impurity limits. This research area should integrate research areas such as fuel integrity, neutronics, and reactor system analysis.

Two aspects of the HTGR and VHTR designs raise the potential for the need for research to support improved inservice inspection (ISI) programs and for continuous monitoring methods. First, more components are enclosed in pressure vessels, making access for inspection difficult. Second, there are longer operating cycles between the scheduled short-duration planned outages when ISIs would be conducted. These circumstances indicate the need to evaluate the effectiveness of less frequent ISIs in ensuring the timely detection of cracking and degradation of components as well as the potential for excessive growth of cracks before the next ISI. The NRC staff expectation is that continuous online monitoring techniques will be proposed as a means to ensure structural integrity, the timely detection of coolant pressure boundary leaks, and the validity of assumed pressure boundary failure frequencies used in the plant-specific PRA. The technical basis for the effectiveness of these techniques will need to be established.

III.2.4.2 Purpose

The NRC staff must develop an independent assessment capability and expertise in the area of high-temperature materials analysis. This capability and expertise are necessary to effectively evaluate and establish regulatory technical bases for HTGR and VHTR safety and safety margins. HTGR and VHTR designs differ significantly from LWR designs in terms of the materials used (such as high-temperature metals and graphite), higher coolant temperatures, and single-phase coolants. HTGR and VHTR materials may also exhibit different degradation mechanisms (such as creep), and the behavior of metallic components in their service environments will also differ from that of LWRs. Research is also needed to develop data for graphite structure design analysis, including the margins, performance, and degradation behavior of graphite structures for licensing-basis conditions, including licensing-basis accidents.

HTGR and VHTR graphite blocks will provide neutron moderation and reflection; structural support; channels for the fuel elements, helium coolant flow, control rod, and shutdown rod insertion; and neutron shielding. Graphite structures and components also will furnish a major heat sink and heat transfer pathway when a loss of normal core cooling occurs. During reactor operation, many physical and thermal properties of graphite change significantly as a result of the effects of temperature, the environment, and irradiation. Significant internal shrinkage, bowing, and stresses can develop that can cause component cracking, change the shape of the graphite core blocks, and/or alter the core geometry. In addition, when graphite is irradiated to a sufficiently high radiation dose, swelling occurs that rapidly reduces strength, causing the component to lose its structural capability. In the event of a loss of pressure boundary accident with air ingress, graphite oxidation can produce additional changes in its physical and mechanical properties.

Irradiation damage will also result in a decrease in graphite conductivity during reactor operation. Over time, this decrease in conductivity will increase the graphite core and fuel operating temperatures. In addition, the loss of graphite thermal conductivity can be recovered during slow heatup events as a result of the effects of high-energy graphite lattice annealing. If credited, the graphite conductivity recovery would substantially reduce the peak core and fuel temperatures during a heatup accident and thereby significantly decrease the fuel FP release during the accident. In the early 1990s, the staff considered crediting the beneficial effect of graphite lattice annealing in the heat removal models that it was using to review the MHTGR accident analyses. The loss and subsequent recovery of graphite conductivity will need to be fully quantified in models though appropriate graphite material irradiations and accident condition heatup simulations.
Through the 1980s, research progressed on the high-temperature design (creep, fatigue) of metal components for the liquid metal fast breeder reactor (LMFBR). This research formed the basis for some American Society of Mechanical Engineers (ASME) design requirements and code cases for high-temperature components. The staff will need to review and evaluate both this research and subsequent research since the 1980s and 1990s—particularly with respect to the temperature, coolant environment, and materials used—to determine the applicability of the earlier research and resulting ASME Boiler and Pressure Vessel Code (ASME Code) requirements to HTGR designs.

The NRC staff needs to develop an independent capability and expertise to understand materials behavior under normal operating conditions, design-basis conditions, beyond-design-basis conditions, and conditions that result in significant component degradation and failure. This information will be necessary to evaluate safety margins and failure points and to quantify uncertainties. To support the evaluation or conduct HTGR PRAs, the staff will need information and data on the probability of safety-significant reactor components failing. Because of the lack of operating experience for specific designs, this information should be developed analytically, using probabilistic fracture mechanics. Thus, potential degradation mechanisms of metallic and graphite components must be identified, and the degradation progression must be quantified for HTGR and VHTR operating conditions. Potential technical issues that should be addressed are (1)availability and applicability of national codes and standards for the design and fabrication of metallic and graphite components for service in HTGR and VHTR helium environments, (2) lack of appropriate databases for calculating fatigue, creep, and creep-fatigue interaction lifetimes of metallic components in high-temperature applications, (3) the effects of impurities (such as oxygen) in the helium coolant on component degradation, (4) aging behavior of alloys during elevated-temperature exposures, (5) sensitization of austenitic alloys, (6) the potential treatment of large-diameter pipes as vessels, (7) carburization, decarburization, and oxidation degradation of metals in HTGR and VHTR environments, (8) ISIs and online monitoring of reactor components, (9) performance and degradation of graphite under high levels of irradiation, (10) lack of knowledge and methods for predicting irradiated graphite properties from the as-received nonirradiated graphite properties, (11) lack of data on oxidation kinetics of reflector-grade graphite, fuel pebble matrix graphite, and graphite dust, and (12) applicability of graphite properties from small components to large-block graphite properties. The following paragraphs address each of these potential technical issues.

Another potential issue for PBRs is understanding and predicting 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, hangup, and bridging. Section III.2.2.2 of this document discusses this issue in greater detail.

III.2.4.2.1 Description of Issues, Metallic Components

The availability and acceptability of national codes and standards for the design and fabrication of metallic components for service in HTGRs represent a key issue. Background studies and activities for eventual development of codes and standards were conducted in the 1980s for application to the LMFBR. Of particular note is the work conducted by the Pressure Vessel Research Committee (PVRC) in its preparation of several technical reports that provided the basis for ASME development of high-temperature design codes. These reports give background and procedures for the design of components to resist fatigue, creep, and creep-fatigue failures. However, these reports did not address the effects of the helium environment, including the presence of impurities such as oxygen. In addition, improved correlations for creep and creep-fatigue have been developed based on research during the 1990s. The PVRC reports do not include these improvements, and the procedures must 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, these calculations need appropriate databases. Past experience and research indicate that environmental effects can reduce fatigue life and accelerate materials degradation. For example, small levels of impurities (e.g., <1 part per million of oxygen) in the high-purity water coolant of LWRs can significantly decrease the fatigue life and the resistance to stress-corrosion cracking of metallic components. The ASME Code did not address these effects. For example, the design data for fatigue were obtained from materials tests in air. Because helium is inert, there has been a tendency to obtain design data in pure helium, in helium containing some (but not all) impurities, or in air. The effects of all important impurities (such as oxygen) in helium should be taken into account with respect to reductions in fatigue and creep life, and such data and understanding must be developed.

To address degradation and aging of metals, the effects of high-temperature helium with impurities (including oxygen) at levels that will be present in HTGR and VHTR designs should 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 at elevated temperature. These transformation reactions are known as aging and can lead to embrittlement of the alloy. Aging and embrittlement occur, for example, in cast stainless steel components for LWR time and temperature conditions. For HTGR and VHTR operating temperatures, the reaction rates can be much higher (i.e., the aging and embrittlement would occur sooner). For different HTGR alloys and temperatures, potentially different aging reactions and mechanisms might occur, and some could be relatively rapid, rendering the material embrittled and susceptible to cracking. The aging reactions, as a function of time and temperature, in the different alloys used in safety significant components of HTGRs and VHTRs should be studied to establish the potential for material property degradation and embrittlement during the plant operating lifetime.

Sensitization is another solid-state reaction that can occur in stainless steels (and austenitic alloys). Sensitization results from the precipitation of chromium carbides at the stainless steel grain boundaries. This precipitation normally occurs during slow cooling of the metal through high temperatures, for example, when metal cools 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 (i.e., cracking along the grain boundaries) in oxidizing and impure environments. A less well-known method of producing sensitization is through low-temperature sensitization, which occurs over long periods of exposure to relatively low temperatures. Low-temperature sensitization in stainless steel has been studied for LWR temperature conditions. In these conditions, low-temperature sensitization would not occur in less than 40 years. However, the sensitization rate is exponential with temperature, and, at the higher operating temperatures of HTGRs, sensitization during the plant lifetime might happen. If such sensitization occurs, the stainless steel components would be susceptible to stress-corrosion cracking.

For some modular HTGR designs, the cylindrical pressure boundary conduit that connects the RPV to pressure vessels of the power conversion system (PCS) was itself treated as a pressure vessel because the conduit was designed, fabricated, and inspected according to the same rules as the RPV. As a result of this treatment, the designer did not consider a double-ended break of the connecting conduit as sufficiently probable to be included in the licensing basis. Treating the conduit as a vessel instead of a pipe would require additional investigation because a pipe has a much smaller diameter and therefore possesses a thinner wall than a pressure vessel designed to the same working pressure. If an unexpected degradation mechanism were to initiate in the cross-connect conduit, because of the thinner wall, it would propagate through the wall thickness in a relatively shorter time, and ISI might not detect it. 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 sufficient time for ISI to detect it.

Carburization, decarburization, and oxidation of HTGR metallic components are additional phenomena that can lead to degradation caused by the operating gaseous and particulate environment. Carburization occurs when 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 can be hard and brittle and might have greater strength than the substrate. Differences in strength and other physical properties between the surface layer and the substrate can lead to high stresses in the surface layer when the component is under load. In addition, carbides can 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 occurs when carbon is depleted from the steel. The carbon depletion results in a softer steel and in reduced fatigue and creep lives. Its occurrence depends on the composition of the gaseous environment. The presence of oxygen results in the formation of scale and general corrosion of metallic components. It can also render metallic components susceptible to stress-corrosion cracking.

To control carburization, decarburization, and oxidation, the levels of different impurities in the coolant must be carefully controlled. However, controlling impurity levels to avoid one degradation phenomenon might cause another degradation phenomenon to develop or intensify. For example, to avoid carburization, some HTGR designers might establish slightly oxidizing conditions by adding oxygen to the helium coolant. However, this oxygen might increase graphite oxidation, exacerbate general metal corrosion, and increase susceptibility to stress-corrosion cracking. Some research has studied these phenomena. However, additional research is necessary to better understand the conditions under which these phenomena can occur for safety-significant metallic components. In addition, much of the available research did not include oxygen in the helium environment. Because oxygen will be present in HTGRs at sufficiently high levels to affect the progression of the previously described phenomena and to reduce fatigue life, creep life, and resistance to stress-corrosion cracking, new experimental studies must include oxygen.

III.2.4.2.2 Description of Issues, Inservice Inspection and Monitoring

A number of potential issues are related to ISI of safety-significant HTGR components. HTGRs are designed to operate for much longer periods of time between ISI and scheduled short-duration shutdowns for maintenance or refueling. Accordingly, the time interval between ISIs will be longer, and the time available for these inspections might be limited. Therefore, the effectiveness of various ISI programs should be assessed as a function of both the frequency of inspections and the number and types of components inspected. In addition, many internal components are not easily accessible for ISI, and the safety impact of not inspecting these components should be assessed.

An alternative to conducting more ISIs during reactor shutdowns is continuously monitoring online the structural integrity of the entire reactor and reactor components during operation and leakage of the pressure boundary. Continuous monitoring techniques have been developed, validated, and codified for use in LWRs. If HTGR ISIs cannot be conducted on a sufficiently frequent basis and certain components cannot be inspected, then continuous online monitoring might be needed. The continuous monitoring techniques should be evaluated and validated for HTGR materials, environments, and degradation mechanisms.

III.2.4.2.3 Description of Issues, Graphite

To support the review of modular HTGR and VHTR designs, research is necessary 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 operation as well as accident conditions. Potential loss of strength and of resistance to fatigue and creep, shrinkage, swelling, cracking, and loss of material during operation could impact the performance and function of the graphite core structural elements, reflectors (side and bottom), and moderators. Graphite variables such as coke source, size, impurity, and structure; manufacturing processes; density; grain size; and crystallite size and uniformity determine both the as-received and the irradiated properties of the graphite component.

SECY-03-0059 identified the lack of information on irradiated properties of the newer nuclear graphites as an issue, particularly at high levels of irradiation and temperatures. The need for research to address these issues was documented. The irradiated material properties depend heavily on the particular makeup of the graphite and the manufacturing process. Irradiation affects, and in many cases degrades, physical, mechanical, and thermal properties of the graphite. Important properties that change with irradiation are density, thermal conductivity, strength, and dimensions. These changes have safety implications because they could degrade core structural integrity, geometry, and cooling. Some of these changes are not linear with irradiation dose. Graphite strength initially increases with irradiation dose, but then, at higher dose levels, it begins to decrease. With respect to dimensional changes, graphite initially begins to shrink with increasing dose, but then, beyond the so-called turnaround dose, graphite begins to swell with increasing dose. During reactor operation, core thermal gradients and irradiation-induced dimensional and strength changes can result in significant stresses, distortion, and bowing of graphite components. These can lead to loss of structural integrity, changes in core geometry, and the potential for hampering or preventing control rod insertion. At still higher doses, beyond the turnaround dose, the swelling makes the volume considerably greater than the original volume, and graphite structures will begin to disintegrate and experience total loss of integrity.

To evaluate the suitability of a particular graphite for a specific HTGR or VHTR application, property change data attributable to irradiation will be needed in addition to the as-received properties. Developing irradiation properties data for graphite is difficult, costly, and time consuming. Consequently, designers and potential applicants have proposed using irradiation data from older graphites and manufacturing the new graphite in a manner similar to that used for the older graphite. However, the as-received and irradiated graphite properties depend strongly on the raw (i.e., source) materials and manufacturing processes. Small variations in either of these can significantly affect graphite properties. Because the raw materials and processes will not be exactly the same and might change further in the future, there will be a need to determine whether a particular new graphite will behave the same as the older graphites under operating irradiation conditions. To accomplish this without irradiation testing, acceptable correlations must be developed that enable the irradiated graphite properties and changes in properties to be predicted from the as-received graphite materials characteristics, composition, processing, microstructures, and properties.

Graphite oxidation can occur from oxidizing impurities in (or impurities added to) the helium coolant, from in-leakage during normal operation, or during air or water ingress accidents. The oxidation of graphite is an exothermic reaction, so it is important to know the rate of heat generation, particularly during accidents. Oxidation will remove the surface layers of graphite components resulting in loss of structural integrity and will provide a source of heat in addition to the decay heat in the core during an air or water ingress accident. Furthermore, oxidation will change the thermal conductivity and will reduce the fracture toughness and strength of graphite components. The oxidation rates vary for different graphites and can be greatly affected by impurities in the original graphite. Therefore, oxidation rate data are necessary for the graphites proposed for new reactors.

The properties used for design, operating, and accident performance of graphite structures often are obtained from relatively thin graphite components or specimens. The relatively thin components are manufactured differently from many large structural blocks used in HTGRs, so the mechanical and other properties will differ. In addition, the properties of large-block graphite will vary through the thickness of the block and from block-to-block. The difference in properties between the thin and large blocks, through-thickness variations, and block-to-block variations should be established. The potential and extent of different irradiated properties through the thickness of large block graphite also should be evaluated. As is the case with other materials and structures, small specimen properties and structural behavior should be evaluated and scaled to predict the properties and performance of large graphite structural components.

Irradiation decreases graphite conductivity during reactor operation. Over time, the decrease in conductivity can increase the graphite core and fuel operating temperatures. The loss of conductivity can be recovered during a core heatup accident as a result of annealing of the graphite lattice damage. The loss and subsequent recovery of graphite conductivity must be acceptably quantified in models through appropriate graphite material irradiations and heatup testing.

There has been a lack of standards for nuclear-grade graphite, and HTGR designers have used measured properties of the particular graphite in design calculations. However, the expectation should be that nuclear graphites meet minimum requirements for important properties, such as strength, density, and thermal conductivity, as do materials used in other reactor systems and components. If a particular graphite has a relatively low strength and the designer uses that value in designing various components, a suitable component for the intended service might not result. The strength might be relatively low because of underlying reasons. For example, the graphite might contain excessive cracking and porosity, resulting in low strength. Although the component design might use the reduced strength (possibly resulting in a thicker component), the excessive cracking can grow during service and cause an earlier-than-predicted failure. Thus, the number, distribution, and size of pores and cracks should be controlled

in nuclear-grade graphite. Specific impurities in the graphite might be detrimental to irradiation properties of the component and should be limited in nuclear graphites. Impurities 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. Therefore, necessary standards were recently developed or are under development to establish the acceptable physical, thermal, and mechanical properties; microstructures; composition; and manufacturing variables for nuclear-grade graphite. The acceptability of these standards should be evaluated.
Accepted codes and code rules for the design of graphite (and composite) structures and components do not take into account the effects of temperature, irradiation, volume changes, stress, fatigue, creep, strength, toughness, and maintenance of acceptable margins over the operating lifetime. In addition, a need exists for ASME Code acceptance criteria or standards (size, distribution, number) for different types of flaws, particularly cracks and pores, in the as-manufactured graphite components. Furthermore, flaw evaluation procedures and disposition requirements are needed for flaws detected during HTGR operation. Moreover, ASME Code rules and requirements are needed for baseline and ISIs using qualified procedures and techniques capable of detecting and characterizing the relevant flaws identified in acceptance criteria.

III.2.4.2.4 Description of Issues, Carbon-Carbon Composites

Recently, HTGR designers have been considering the use of carbon-carbon (C-C) composites and SiC-SiC composites for safety-significant reactor components. Some of these applications include absorber rods, insulation and support structures, core restraints, hot pipe extensions, parts for reactor conditioning (cooling) systems and reactor control and shutdown systems, tie rods, stiffeners, and core barrels. Because of the C-C manufacturing and design advances in aerospace technology and applications, a variety of standard shapes are currently available from commercial manufacturers. These standard shapes include plates, tubes, and rods. Custom shapes can also be manufactured. Many of the commercially manufactured C-C composites typically have been for low-temperature applications and involve the use of carbon or graphite fibers in polymer matrices. Either two-dimensional hand layup or three-dimensional hand layup or automatic filament-wound technologies are used in the fabrication of such composites. In some instances, four-dimensional composites have also been fabricated.

Many of the general issues concerning the raw materials and fabrication methods used to generate the reactor component, which were addressed earlier for nuclear graphite, apply to the C-C composites. For example, for two-dimensional composites exhibiting layered structure, the directional-dependent significant properties should be studied as a function of temperature and fluence. Some of these properties are physical properties such as dimension and density; thermal properties such as specific heat, coefficient of thermal expansion, and thermal conductivity; and mechanical properties such as elastic modulus, shear and torsion modulus, flexural strength, shear strength, tensile strength, compressive strength, fatigue strength, and creep strength. Appropriate methods to handle the variability in properties (e.g., Weibull statistics) must be developed, tested, and validated to provide uncertainty and probability of failure input to lifetime prediction codes.

Research on the fracture mechanics of C-C composites is needed to establish the ranges for the critical stress intensity factor, critical strain energy release rate, and strain-to-failure under radiation environment. These properties have implications for predicting the useful life of reactor components and technical safety considerations for repair and replacement. Dimensional stability under radiation environments represents a significant property that must be generated and understood. The creation of neutron-induced voids and their possible linking can produce significant swelling in these composites.

Such damage could potentially lead to the delamination of the layered structure and to a significant or complete loss of the ability to carry the design loads. For three-dimensional fabric-woven composites, a properties database is needed for the three-orthogonal directions and their associated shear components. The nature of the fabrication process and the resulting microstructural features that govern the significant properties (such as thermal expansion, thermal conductivity, elastic modulus, and dimensional changes) dictate that properties must be determined in multidimensions with an understanding of the interacting effects.

Both short-term and long-term mechanical tests will be needed for both specimens containing known and well-characterized flaws and reference baseline samples to establish the crack growth behavior of components under reactor operating conditions. A quantitative understanding of time-dependent (dose-dependent) degradation mechanisms is necessary to develop criteria for the timely inspection, evaluation, and repair of components that degrade during reactor operation.

A significant issue under reactor operation and potential fault conditions is the oxidation behavior of the C-C composites. Given the nature of the fabrication of these composites, the interface between the reinforcing fiber and the matrix can be weak because of the existence of substantial porosity and lack of fiber-matrix bonding. In addition, any creation of voids attributable to irradiation and subsequent swelling and delamination will create newer and substantially increased active surfaces for easy oxidation and damage. Oxidation could preferentially and rapidly occur along these areas, rendering the structure weak and unable to sustain design loads.

The staff expects that the chemical and interfacial bond strength between the fiber and the matrix will be sensitive to impurities in the coolant environment. The tests should include several variations of expected coolant chemistry, including those that would be anticipated under upset conditions, to establish the oxidation and mechanical properties database.

III.2.4.3 Objectives and Associated Activities

NRC research is aimed at developing an independent capability for the agency to evaluate the integrity of important components in advanced reactors under operating and accident conditions. Researchers should conduct studies on metallic components 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 must be considered for updating the current national code design rules and procedures to provide improvements as necessary. The ASME Code procedures should also be updated to incorporate correlations developed from more recent research. Necessary research on graphite should (1) evaluate performance under high levels of irradiation and at high temperatures, (2) develop empirical correlations for predicting irradiated properties from as-received properties, processing, composition (including impurities), and microstructures, (3) develop data on oxidation kinetics, (4) evaluate variations in properties through the thickness of large blocks, (5) develop codes and standards for nuclear-grade graphite, and (6) develop an understanding of the mechanics of pebble flow. The following sections describe this research for metallic components, ISIs, graphite components, and C-C composites.

III.2.4.3.1 Metallic Components

Carburization, decarburization, and oxidation of HTGR high-temperature metals should be studied as a function of time and temperature in helium gas with impurities, including oxygen. Such studies should include different levels and ratios of impurities. Metallographic studies and mechanical testing should 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, the degree to which they occur under different conditions, the potential for their occurrence under the operating conditions of HTGRs, and their significance for the structural integrity of components.

Research should 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, researchers should investigate the effects of impure helium environments on the creep and creep-fatigue life of HTGR components. The objective of this research is to ensure that the available design rules and procedures address reductions in life attributable to the operating environment. If the codes and procedures do not consider these phenomena, then the database developed should be used to update the codes and procedures to provide design procedures and rules that prevent the failure of HTGR components during service. In addition, research should quantify the effects of carburization and decarburization on the reduction of fatigue and creep life to ensure that the design procedures and analyses account for these reductions.

Researchers should conduct studies 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 rates. The tests would use materials in the as-received condition and in the carburized and decarburized conditions. The objective of this research is either to confirm that these degradation mechanisms do not occur and that 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 should be conducted on high-temperature alloys used in HTGRs, employing samples in the as-received and the welded conditions. Samples should be exposed for different times to temperatures at and above the operating temperatures of the components. Exposure to higher temperatures will accelerate the aging and sensitization reactions. As long as the aging mechanisms at higher temperatures are the same as those at operating temperatures, researchers can develop correlations for quantifying the times required to reach different levels of aging and sensitization at the operating temperatures expected. Mechanical property testing on the aged samples would quantify the degree of embrittlement and other property changes as a function of aging time and temperature. Metallographic and microscopy studies would identify the aging and precipitation reactions, if they occur, to ensure that the reactions are the same at both 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 that, 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.

Numerous potential degradation and aging mechanisms in the HTGR operating environment have been discussed. Research on components removed from operating reactors could evaluate and validate these potential degradations. This could involve an international research program on components removed from the AVR, including microstructural studies and mechanical tests. Microstructural studies could determine whether solid-state changes and precipitation have occurred during operation to produce thermal aging, sensitization, carburization, and decarburization. In addition, metallographic studies could establish whether stress-corrosion cracking, crevice corrosion, general corrosion, and oxidation have occurred. Mechanical tests on materials removed from the AVR also would confirm whether any degradation in materials properties has occurred. Fatigue and creep tests would determine whether fatigue and/or creep damage have occurred, whether the design codes and methods correctly predict the damage, and whether the coolant environment had an effect in reducing fatigue and creep lives. The results would provide insights on whether and how the design codes/procedures would need to change to take into account the potential degradation mechanisms.

Internationally, considerable past and ongoing research by the E.C. and Japan focuses on high-temperature metals for HTGRs. In several identified areas, this research addresses NRC's research needs and objectives. E.C. work of interest includes its (1) review of RPV materials, focusing on previous HTRs to establish a materials property database on design properties, (2) compilation of existing data on materials for reactor internals and on selection of the most promising alloys for additional development and testing, and (3) compilation of existing data on turbine disk and blade materials and the selection of the most promising alloys for additional 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 strength, fracture toughness), including studies on both heavy-section base metal and weldments, (2) mechanical and creep tests of candidate alloys for reactor internals at temperatures up to 1100 °C, with a focus on the control rod cladding, and (3) tensile, fatigue, and creep tests from 850 °C up to 1300 °C for two different turbine blade materials, one forming an aluminum oxide protective layer and the other a chromium oxide layer.

The JAEA has conducted work of interest, including 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 RPV steel.

Other international efforts, such as work in the United Kingdom (where the issue has been raised), would be useful for determining the long-term degradation mode of glass fiber-encased insulation components. Researchers could conduct studies on the effects of vibrations and service conditions to determine the reliability of this insulation, which protects the metallic components and pressure boundaries in HTGR designs from unacceptably high temperatures.

As mentioned above, the E.C. and Japan have conducted, are conducting, or are planning considerable research of interest to the NRC on high-temperature materials for HTGRs. To leverage NRC resources and obtain timely data, the staff visited facilities and met with members of the international community in 2001 and 2002 to initiate a dialogue on cooperation. The international community, particularly DOE, Japan, and the E.C., has reviewed the materials research description in the 2003 infrastructure assessment. The NRC staff met with the technical staff and officials of the E.C. and JAEA to discuss potential cooperation. The E.C. agreed with the importance of, and need for, the research outlined in the NRC infrastructure assessment and welcomed the agency's participation in its high-temperature materials research program, known as HTR-M. Similarly, JAEA has agreed to cooperate with the NRC. Participation would be through the exchange of research results, but not funds, from the parties' research programs. Some key work possibly not fully addressed in other programs falls in the areas of (1) effects of the helium environment (with impurities) on degradation of materials and (2) aging and sensitization. The exchange of NRC research results in these areas could support cooperation with other international programs.

III.2.4.3.2 Inservice Inspection and Monitoring

In the nondestructive examination area, needed research should evaluate the impact of different ISI plans on structural integrity and risk. The key variables include the length of time between inspections, the reliability of inspection methods, and the number of components and locations tested. Probabilistic fracture mechanics analyses should consider different degradation mechanisms appropriate to the reactor design and operating environment, as well as the inspection variables, to evaluate the impact of potential failures on risk. The results of this work will support the evaluation of proposed ISI methods and the determination of the technical basis for improved, more frequent, or more extensive ISIs. The results will also offer guidance on the need for continuous online monitoring of structural integrity.

Because some components are inaccessible and because the interval between ISIs might be long, research should evaluate continuous monitoring of reactor components for crack initiation and crack growth and for leak detection. Researchers could use acoustic emission techniques for laboratory testing of specimens under simulated HTGR conditions (e.g., temperature, noise sources, coolant flow) to evaluate fatigue, creep, and stress-corrosion cracking. This would enable them to develop correlations for crack initiation and crack growth rates with the acoustic emission signals for HTGR materials and environments. The NRC conducted similar research in the 1980s and 1990s, when acoustic emission techniques were developed, validated, and codified for application to LWRs. The HTGR research, methods, and techniques took advantage of the knowledge gained in earlier work. Researchers need to evaluate similar acoustic emission techniques for detection, location, and quantification of coolant leakage from the pressure boundary and internal components under HTGR operating conditions. HTGR research should benefit from similar work for LWR applications. Once the laboratory research is completed and the correlations of acoustic emissions to crack initiation and growth are developed, an operating or test HTGR could be instrumented with acoustic emission sensors and then monitored during its operation to validate the methods and correlations developed in laboratory testing. The results from this work would provide an alternative to periodic ISIs and would demonstrate the advantages of continuous online monitoring of reactor structural integrity and leakage. The results will also generate technical databases for incorporating the techniques into codes and standards.

Potential areas of international cooperation and exchange would involve planned E.C. work on evaluating ISI methods and NRC work on risk-informed inspection program evaluation. 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.

III.2.4.3.3 Graphite

Researcher should be conducted to evaluate graphite for HTGR design applications. This would involve studies of the performance and degradation of graphite under high levels of irradiation and at high temperatures. The data would be used to determine the behavior of current graphites planned for HTGRs under operating conditions. In a previous research activity sponsored by the NRC at ORNL, a review examined available high-dose irradiation data for nuclear-grade graphite, including unpublished data from ORNL, collected under the DOE MHTGR program. In general, data are lacking for current graphites under the high-dose, high-temperature regime of the HTGR and VHTR operating environments. Additional research is necessary on current graphites planned for HTGRs to determine high-dose, high-temperature material behavior, properties, and degradation. 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 important properties of current graphites.

The staff is aware of a graphite irradiations program that the E.C. is conducting at the Joint Research Center in Petten, Netherlands, using a variety of newer graphites with variations in coke, binder, and processing. The staff also knows about the irradiation program plan at INL, conducted in collaboration with ORNL, to evaluate the effects of fluence and temperature on graphite properties. In addition, DOE is sponsoring a relevant research program conducted in Donetsk, Ukraine, at the L.M. Litvinenko Institute of Physical-Organic and Coal Chemistry, in collaboration with the National Science Center Kharkiv Institute of Physics and Technology. The staff will follow the progress of these experimental programs and assess the results for potential use in addressing licensing technical issues and will cooperate in the research if feasible.

Research also should be performed to determine the irradiated graphite properties from as-received graphite properties. The raw materials and manufacturing process determine as-received graphite properties. The research program should address effects processing, coke, pitch characteristics, graphitization temperature, composition (including impurities), and microstructure. A number of different graphites should be selected with carefully varied parameters. Studies would establish the as-received properties of the graphites. Selected properties to be measured include 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, studies would establish the chemistry of the graphites, including impurities. Because of the anisotropy of manufactured graphite, the materials properties should be determined for two orthogonal directions because graphite exhibits transverse isotropy. The graphites would then be irradiated at systematically varied irradiation doses and temperatures significant to HTGRs. Following irradiation, a reevaluation of the materials properties, compositions, and microstructures would determine the effect of irradiation and would establish correlations between the initial as-received properties and the postirradiation properties that could apply to any particular graphite used in HTGRs.

Investigators would need to undertake studies to understand oxidation and its effects on the physical, thermal, and mechanical characteristics of nuclear graphite. Data are lacking on oxidation kinetics of reflector-grade graphite, fuel pebble matrix "graphite" or carbon, and graphite dust. Experiments should determine weight loss and loss of mechanical integrity attributable to the oxidation of graphite samples under a variety of flow rates of air and coolant chemistry. The heat generated from the oxidation of graphite—and the potential detrimental effect on surrounding components because of this elevated temperature—needs to be studied.

Although continuous annealing effectively prevents any significant buildup of Wigner energy at the high operating temperatures of HTGR graphite, there could be a significant accumulation of higher energy graphite lattice distortions that anneal out only at the elevated graphite temperatures encountered in accidents (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 core heatup events. Research is needed to gather existing data on the high-temperature annealing effects and to establish definitively the role, if any, played by Wigner energy for the HTGRs.

In the case of pebble bed type reactor, a better understanding of the tribology involved between the "graphite" pebbles rubbing against each other and the "graphite" pebbles contacting graphite reflectors and moderators. Because friction coefficient is high between like-materials, it could be expected that high friction between particles may provide some adhesive bonding; however, the effects of atmosphere and temperature plays a significant role. Thus, at highly pure helium environment this may not be an issue.

Nevertheless, the generation of “graphite” dust may be expected due to the tribological effect between contacting “graphite” spheres. It is also recognized that the “graphite” fuel balls are indeed not really graphite, having not been exposed to graphitization temperatures during manufacture. Therefore, the fuel balls have outer material which is incompletely graphitized and thus carbonaceous. Such materials have higher hardness than fully graphitized graphite. The fuel balls contacting graphite will cause erosion and removal of graphite reflectors and moderators and thus generating graphite dust.

A need exists to conduct research and establish a comprehensive understanding of dust generation under pebble dynamics because during operation the dust is transported within in the helium pressure boundary due to helium coolant forced flow, resulting in plateout and settling in different low coolant velocity locations, for example. Also the graphite dust can oxidize more readily due to its high surface area. Whether this is beneficial or harmful needs to be established. In addition to these effects, the irradiation itself causes damage to graphite. The dust generation and oxidation effects must be studied on irradiated materials to represent any air-ingress and moisture-ingress accident scenarios of operational graphite. The studies should also include a comprehensive data gathering on the size and shape distribution of the graphite dust, which will form input to dose calculations.

Research on large blocks of graphite should 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 that will be used for reflector material should be sectioned, tested, and evaluated to determine whether 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 the size of the final fabricated component. The sectioned large block specimens should 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 primarily on the work that the NRC initiated in 2003 under a contract to ORNL, a consensus American Society for Testing and Materials (ASTM) materials specification standard now exists for nuclear-grade graphite exposed to high temperatures and high levels of irradiation. ASTM is seeking to develop additional materials specification standards and specific standards for measurement of important physical, thermal, and mechanical properties for nuclear-grade graphite. ASME also has code activities under way to develop requirements for graphite components. To be fully effective, these code activities must address strength, fracture, fatigue, creep, irradiation damage, dimensional stability, oxidation, and any other appropriate design and fabrication considerations for HTGR service. The code activities also must consider (1) flaw acceptance criteria (numbers, sizes, distribution) for as-fabricated graphite components, (2) flaw evaluation procedures and disposition requirements for flaws detected during the operation of HTGRs, and (3) baseline and ISI requirements using qualified procedures. Ongoing meetings of international nuclear graphite specialists serve as a forum for research related to nuclear graphite.

Comprehensive models and methods are also necessary for predicting deformation and shape changes in graphite components as a function of radiation dose, temperature (as calculated from graphite neutronics, stresses), and creep under temperature and irradiation conditions.

Experimental data, analyses, and appropriate analytical models are needed for predicting pebble flow in a PBR core. Evaluations should consider how the predictive models were validated and how well they predict data from experiments and field experience. Pebble flow, temperature effects, friction, hangup, and bridging should be considered in the previously described evaluations. Conclusions should address the adequacy of currently available methods and analytical codes for predicting pebble flow through a pebble bed reactor (PBR) core as well as the need for follow-on improvements in predictive capabilities. The E.C. research has reviewed the state of the art on graphite properties to establish 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. The E.C. is continuing extensive characterization and irradiation testing of five different recent nuclear-grade graphites that could be used in future HTGRs. The properties of these graphites as a function of temperature and irradiation exposure will need to be studied. As mentioned previously, the E.C. plans to address a considerable quantity of work related to high-temperature metallic components. A key area of needed research is the correlation of as-received graphite properties and manufacturing parameters to irradiated graphite properties.

III.2.4.3.4 Carbon-Carbon Composites

The application of C-C composites as structural load-bearing components in a reactor requires that adequate properties remain throughout the operating lifetime of the reactor to ensure that the required safety margins are maintained.

The primary activities and areas of needed research in C-C composites to implement and assess appropriate application of these materials include the following:

- establishment and application of material specification standards
- development of methods to model variability in properties (e.g., Weibull statistics) for use in design codes
- development of procedures to scale properties data from small samples to product sizes
- development of nonirradiated and irradiated database on properties
- development of predictive models to estimate irradiated properties from nonirradiated properties
- establishment of design codes and standards
- development of mechanical properties related to crack growth behavior under reactor operating conditions
- development and application of probabilistic nonlinear and an isotropic fracture mechanics methods and crack growth data for input to life prediction models
- development of comprehensive models/methods for predicting deformation and shape changes in components as a function of radiation dose and temperature as calculated from neutronics, stresses, and creep under temperature and irradiation conditions
- development and application of inspection methods that provide data on C-C composite microstructure and flaws, including statistical relationships that characterize the fiber size, shape, and orientation distribution; porosity, pore structure (size and shape distribution), and pore orientation; and matrix grain orientation

- development of appropriate accept/reject criteria based on results from fracture mechanics for flaws and anomalies in as-fabricated components as well as development of accept/reject criteria for the size, orientation, and position and the number of cracks and crack-like defects that form during reactor operation

III.2.4.4 Application of Research Results

Results from the described research will provide the capability, expertise, and information necessary to support the staff review of an HTGR or VHTR application in the area of high-temperature materials. This will include evaluation of the designer and applicant licensing-basis component failure probability estimates used in the plant PRAs and staff safety evaluations. To reduce the uncertainties in the failure probability for components, information on failure probabilities should be derived from the research results on potential degradation mechanisms (fatigue, creep, creep-fatigue, oxidation, thermal aging, stress-corrosion cracking, crevice corrosion cracking, irradiation damage, and dimensional changes) in the operating environment of HTGRs as well as from quantitative information on the initiation times and growth rates.

The research will determine whether these materials are prone to degradation and will provide the technical basis or criteria for materials acceptability. This will include the effects of aging and degradation attributable to the high-temperature helium environment and high radiation dose. The research will furnish a technical basis for evaluating potential degradation mechanisms and the rate of degradation progression for materials used to connect the pressure boundary conduit between the RPV and the PCSs to assess whether the probability of conduit failure is sufficiently low that it should be excluded from the plant licensing basis.

The research on nondestructive examination and evaluations of ISI programs for HTGRs will provide the technical basis for reviewing the designer and applicant inspection plans and determining whether additional or modified requirements are necessary and whether any proposed methods for continuous online monitoring of integrity for reactor structures and components will be acceptable.

III.2.5 Structural Analysis

III.2.5.1 Background

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. Confirmatory research should address these structures because they are subject to nonlinear response during horizontal and vertical earthquakes.

In some of the advanced reactors, the seismic capacity of the fuel in the RPV governs the overall high confidence in low probability of failure (HCLPF) seismic capacity of the plant. Confirmatory research should determine the seismic margin capacity or the HCLPF capacity of HTGR fuel.

The proposed PBMR is a standardized design that is built with multiple modular units combined onto a single site. As the various modular units are constructed in sequence, the seismic response and capacity of the plant can vary at various stages of construction of the modular units. The variation in seismic response results in part from the overall dimensions (footprint size) of the modular unit foundation (i.e., a site with two modular units responds differently than a site with eight modular units). Confirmatory research is proposed for sites with multiple modular units to identify the seismic response at various stages of construction and to determine the effect of proximity of individual modules.

In the HTGRs, concrete structures can be subjected to sustained high temperature. Research should address issues related to the transient aspects of high-temperature performance of reinforced concrete (RC) structures during heating and cooling. Degradation of concrete attributable to tensile cracks and compressive crushing under high temperature results in reduced stiffness, decreased strength, and spalling of concrete. Explosive spalling occurs when large bodies of concrete experience high heating rates. In many cases, spalling produces a significant loss of cross-section integrity. Literature reviews also indicate that the difference between thermal expansions of concrete and steel can be as high as several hundred percent. Such a large difference in the thermal expansion of steel and concrete can generate very high stresses in the steel-concrete interface and can lead to loss of bond and hence composite action. The rate of heating and cooling also has a significant effect on the distribution of self-equilibrating thermal stresses and on the formation of large compressive stress gradients introducing spalling. Research is needed to model (1) the elastic-plastic-damage behavior of RC structures in compression and tension using general purpose finite element programs such as ABAQUS and LS-DYNA, (2) spalling of concrete under a given thermal environment, (3) loss of bond between concrete and steel at high temperatures, and (4) the effects of temperature cycles attributable to heating and cooling.

III.2.5.2 Purpose

The purpose of this research activity is to develop criteria for the structural and seismic evaluation of the new features of HTGR designs. Major seismic and structural design features that deviate from current practice should be reviewed to ensure that a level of safety, at least equivalent to that of currently operating LWRs, is provided and that uncertainties in the design and performance are taken into account. For unique features or areas that are not similar to existing operating nuclear reactors, the staff should conduct research to develop the technical basis for regulatory decisionmaking. Research also should improve the NRC's knowledge and understanding of new phenomena for which analytical methods and analyses are not currently available to the staff. The areas in which seismic and structural research is proposed include (1) seismic analysis, including nonlinear seismic analysis of reactor vessel and core support structures, seismic HCLPF capacity of nuclear fuel, and the effect of modular construction on the seismic response of the plant and (2) structural response, taking into account the effects of high temperature.

A key area of analytical research for HTGR 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 a need to assess high contact point stresses between the spherical fuel pebbles that are attributable to both deadweight and seismic events for the PBRs. Confirmatory research also should determine the seismic HCLPF capacity of nuclear fuel and the effect of modular PBR construction on the seismic capacity of the plant.

For concrete performance under high temperatures, research should focus on accumulating the existing database, expanding the database, and evaluating the impact of high temperatures on concrete properties. Combined elastic damage and plasticity models are necessary for realistic descriptions of the triaxial response behavior of concrete when reduced stiffness, decreased strength, and permanent strains develop simultaneously under high-temperature excursions. Both temperature and pore pressure should be considered as primary variables, in addition to displacements and porosity changes attributable to progressive microcracking, in a realistic finite element analysis. Researchers should evaluate the reduction in bond strength of deformed and prestressing steel bars with increasing temperature, the effect of shape and surface conditions of reinforcing bars, and the effect of concrete cover in the reduction of bond strength. To determine the effect of rate of heating, the time history of the temperature buildup as well as the spatial distribution should be considered.

III.2.5.3 Objectives and Associated Activities

The 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. The plan to implement this overall objective is based on the overall research objectives detailed below.

III.2.5.3.1 Seismic Structural Analysis of Structures and Components

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. Analytical research should 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 should be evaluated for applicability to nonlinear configurations such as the HTGR reactor components consisting of nonductile graphite core reflectors and supports.

Research should be conducted on the nonlinear static and dynamic structural analysis of advanced reactors with long fuel tubes and core support structures whose seismic margins might be controlled by the fuel design. For the PBMR reactor, fuel pebbles are piled into a considerably high configuration, resulting in nonlinear responses during horizontal and vertical components of earthquakes. Researchers should conduct linear and nonlinear elastic and plastic stress analyses that consider deadweight and seismic events, taking into account contact stresses between the spherical pebbles within the high piles of fuel pebbles.

Seismic margin studies attempt to determine the level of earthquake below which core damage is very unlikely. This level of earthquake is the HCLPF capacity of the plant. Because much credit is given to the integrity and quality control of the coated fuel pebbles to retain the radioactivity in the PBR design, it is important to conduct confirmatory research to determine whether the HCLPF of the fuel controls the seismic HCLPF capacity of the plant.

Confirmatory research should address the objective of evaluating the effect of modular construction on the seismic response of PBMR plants. In the multimodule PBMR power plant, the nuclear island consists of several modules constructed at various stages and placed on a common foundation mat. Both the seismic capacity and the seismic response of the plant depend on the overall foundation size of the plant and the interaction between the various modules. Confirmatory research is also proposed to determine the minimum separation between modules, taking into account seismic events.

III.2.5.3.2 Effect of High Temperature on Concrete

The operating temperatures of the primary reactor vessels for some of the advanced reactor designs being considered are significantly 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.

The objective of this research is to develop methodologies to consider the reduced stiffness and strength of concrete by applying a combination of damage and plasticity models. Temperature dependence of the stiffness and strength can be calibrated from available test data in the literature. To address the spalling of concrete, research should consider the effect of the change in porosity attributable to progressive microcracking on the buildup of pore pressure. Hence, fully coupled hygro-thermo-mechanical simulation might be the most appropriate approach to quantitatively analyzing spalling. Depending on whether a concrete wall is subject to uniform heating (i.e., the same temperature gradient applied to the entire RC wall) or nonuniform heating of the concrete surface, the coupled temperature and pore pressure could be solved as a one-dimensional or multidimensional field problem to identify input data for the subsequent mechanical degradation analysis.

The performance of RC structures is severely impaired by the loss of bond under high temperature when thermal mismatch and load-induced thermal strain effects lead to separation between concrete and reinforcing steel. In this case, zero-thickness interface models might have to be used to characterize the loss of bond at elevated temperatures. Research also should be performed to evaluate the effect of the rate of heating and cooling on the distribution of self-equilibrating thermal stresses and on the formation of large compressive stress gradients that could introduce spalling.

The objective of this research is also to participate in the revision of existing codes to address and evaluate the behavior of RC structures when subjected to sustained high temperatures. In the current American Concrete Institute Code, the temperature limits specified for concrete are 150 °F for long-term use, 200 °F for normal use, and 300 °F for abnormal conditions.

III.2.5.4 Application of Research Results

This research will consist of investigating state-of-the-art analytical techniques to develop regulatory guidelines and the technical basis for regulatory criteria that reflect the latest knowledge and understanding in the seismic and structural area. The end product of this work will be guidance published as a NUREG for each task as well as updates of regulatory guides and SRPs as necessary. The results of work on concrete performance at high temperatures also should stimulate staff interactions with the industry to help develop code revisions that address the effects of elevated temperatures on concrete and structural analysis.

III.2.6 Hydrogen Production Plant Analysis

III.2.6.1 Background

A principal mission of the NGNP VHTR is the secondary production of hydrogen from process heat. The potential reactor safety implications of the production and storage of hydrogen near the VHTR must be understood and effectively analyzed for the NRC to license the NGNP. The safety analysis of the NGNP hydrogen production facility involves modeling the interaction of several physical phenomena and technical arenas, potentially including structural analysis, reactor systems analysis, accident analysis, and consequence analysis.

The basic configuration for the NGNP, which is still under development, involves coupling a VHTR with a

PCS and a hydrogen production plant. Within the NGNP concept, helium, the reactor coolant working fluid, transports heat from the VHTR core to both the PCS and an IHX. The IHX couples the VHTR to the hydrogen production plant. The VHTR can be either a modular PBR or a PMR. In addition, the NGNP PCS and the hydrogen production plant are in the development stage. Several technologies are being considered for the latter, including steam reforming of methane (the current standard), thermo-chemical water splitting, thermally assisted electrolysis, and hybrid cycles.

The NRC has been concerned about the safety hazards posed by the use of volatiles at commercial nuclear power stations. Past NRC regulations and guidance have addressed the risk of combustion of volatiles by generating practical methodologies and conservative recommendations. Such regulations and guidance include

- Regulatory Guide 1.70, Revision 2, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition,” issued September 1975

- Section 9.5.1, “Fire Protection Program,” of the SRP

- GI-106, Revision 2, “Piping and the Use of Highly Combustible Gases in Vital Areas”

- GI-167, Revision 1, “Hydrogen Storage Facility Separation”

- Regulatory Guide 1.91, “Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants,” issued February 1978

In particular, Regulatory Guide 1.91 outlines a methodology for evaluating the impact of transporting volatile chemicals in the vicinity of nuclear power stations. The staff also evaluated a problem similar to that of the NGNP in the context of the Grand Gulf Nuclear Station early site permit review. The analysis considered the detonation from the dispersion of volatile plumes from liquified chemicals on a passing river barge as well as potential detonations from a nearby gas transmission pipeline, N-hexane tanks, liquid hydrogen delivery trucks, and liquid hydrogen storage tanks.

Chemical explosions are also an important consideration in the design of structures and processes in the petroleum and mining industries. These considerations have resulted in the development of commercially available design-analysis tools. The U.S. military has also funded research on the blast effects of conventional weapons on structures.

To support the NRC’s independent safety review of the NGNP, analytical tools should be developed to evaluate the potential reactor safety impacts posed by the NGNP hydrogen production facility The potential events and their consequences should be fully understood.

III.2.6.2 Purpose

The purpose of the infrastructure assessment is to evaluate the technical and safety issues and the safety R&D needs associated with events initiated in the NGNP hydrogen production facility. Because the R&D needs overlap those of other HTGR R&D needs (described in other technical sections of this infrastructure assessment), some of the activities may be specific extensions of the other R&D needs.

This infrastructure assessment includes the identification of potential accident scenarios; a suitable figure of merit such as the damage states associated with civil structures, the helium pressure boundary (HPB), or the core components or FP release; important scenarios; and the associated physical phenomena for the safety analysis. This assessment also includes a literature review to obtain an understanding of potential models and to support the selection of the most appropriate models for the safety analysis. The expectation is that these independent models should be validated for the range of parameters involved for the NGNP before they can be specifically applied to the NGNP safety evaluation. By the end of this research activity, personnel should acquire a broad understanding of the relationships among the physical phenomena within the possible accident sequences to support an independent safety review of the NGNP.

III.2.6.3 Objectives and Associated Activities

The objectives of this infrastructure needs assessment are to support the creation of an associated R&D plan for the development of independent expertise, tools, and capabilities to support the staff review of the safety implications of the NGNP hydrogen production facility. One objective is for the models, tools, and methods to be conservative, but not overly conservative. To this end, a range of topics are discussed that might overlap with other research areas. In addition, because the phenomena are interrelated, R&D possibly should be completed as new information becomes available.

III.2.6.3.1 NGNP PIRT

Because of the complexity of the NGNP balance of plant designs, an NGNP PIRT is needed to identify the hydrogen production plant initiating events, associated safety concerns, and important phenomena that should be modeled. This PIRT includes classifying the associated hazards (e.g., internal, external, IHX connectivity); their initiating events attributable to equipment malfunctions, human error, and natural disasters such as earthquakes; relevant physical phenomena; and a figure of merit for each accident scenario or class of accidents.

At the PIRT stage, it is imperative to separate the initiating events into groups of generic issues that are common to several designs and into groups of design-specific concerns. Initiating events could be attributable to random equipment failure such as a fatigue pipe break, natural disasters such as an earthquake or tornado, or even intentional sabotage. In addition, a range of engineering analysis might be needed to model a volatile explosion and its effect on the VHTR, including flow and combustion analysis of the volatile; dynamic structural deformation of the reactor building as well as degradation or failure of the helium pressure boundary, PCS, and piping; graphite oxidation core damage; and FP release.

A preliminary review has identified the following three potential hazard classes associated with the hydrogen production facility:

- External hazards. These are attributable to an external unconfined detonation, deflagration, or fire hazard to the reactor building civil structure posed by volatiles associated with the production of hydrogen. Volatiles can include hydrogen and liquefied natural gas.
- Internal hazards. These are attributable to helium pressure boundary failures within the reactor building civil structure caused by a confined detonation or fire inside the reactor building. This postulated detonation or fire would occur because of seepage of flammable gas into the reactor building. The pressure pulse might fail the helium internal pressure boundary.

- Connectivity hazards. Certain NGNP hydrogen production design concepts involve a heat exchanger between the primary helium loop and a secondary steam/hydrogen mixture. An event in the hydrogen production plant might result in an IHX coil failure, which might allow hydrogenous mixture to enter the reactor helium pressure boundary. This might result in the ingress of water and hydrogen into the primary helium loop. Water ingress could introduce additional moderator into an undermoderated reactor core, thereby increasing core reactivity. Water would also increase the transport of FPs from the kernels of any failed fuel particles within the core.

III.2.6.3.2 Literature Review

A comprehensive literature review of the available models and experimental data will be necessary to isolate knowledge gaps and select an appropriate model for each accident scenario.

For example, many analytical and computational models embed the use of trinitrotoluene (TNT) equivalence (relating the mass of a volatile to an equivalent TNT mass). An assessment will be necessary to determine whether an equivalent TNT mass approach is appropriate for the NGNP safety review.

III.2.6.3.3 VHTR Core Graphite Oxidation from a Detonation or Fire in the Hydrogen Production Plant

An internal explosion in the hydrogen production plant piping system might result in an overpressurization and failure of the IHX boundary and the subsequent failure of the helium pressure boundary. If such an event occurs, it would result in an HPB blowdown. In the case of a PBR, this might ignite the dispersed graphite dust blown outside the reactor vessel. Graphite burning can lead to degradation of the matrix and possibly the release of FPs outside of the confinement boundary.

III.2.6.3.4 NGNP PBR Criticality Events

Pebble compaction attributable to seismic events are considered design-basis events and beyond-design-basis events for PBR designs. For the NGNP PBR design, a seismic event might also produce a volatile explosion outside the reactor building and result in the failure of the IHX boundary between the helium and pressurized water/hydrogen. Such an event would involve pebble compaction with the introduction of additional moderator into the core, thus increasing the criticality.

III.2.6.3.5 External Events

External events initiated in the hydrogen production plant will require assessment of the reactor building civil structure integrity. An external event could be initiated from the area of the hydrogen production plant.

If an external event is initiated in the NGNP hydrogen production plant, the NRC staff expects that the potential reactor safety consequences to the VHTR are a function of the distance between the hydrogen plant and the VHTR reactor building as well as the extent to which the VHTR reactor building civil structure is elevated above grade. On the basis of conceptual design information, the expectation is that the PMR is below grade-level, while the PBR design involves a reactor system profile that is partially above grade-level. From a reactor building structural analysis perspective, two classes of external plant events should be considered, specifically the blast overpressure from an unconfined volatile detonation and the heat generated by a fire. The engineering analysis of the effects of fire and explosive blast initiated in the hydrogen plant should be divided into (1) an estimate of the incident blast overpressure and impulse or, in the case of fire, the thermal energy and disturbance in atmospheric current and (2) a calculation of the structural response of the confinement building to that loading.

A preliminary review of available analytical tools indicates that existing tools and engineering methods can be used to evaluate the separation requirements between the reactor building civil structure and the hydrogen production plant. In addition, existing combustion physics literature, current computational tools, experimental data, and analytical methods that have been developed for industry and the military also can be used.

Emphasis should be placed on selecting the most appropriate combustion model for the NGNP blast conditions. This model should be capable of estimating the blast wave parameters that might deform the reactor building structure. Given certain assumptions (such as a TNT equivalence, hemispherical explosion, and one-dimensional blast wave), a computer code might be unnecessary, and simple hand calculations might yield the necessary results with an acceptable uncertainty. However, such an approach would require verification and validation of the external conditions and underlying assumptions for the physics of the blast wave. Commercially available analysis codes such as PHOENICS and AutroReaGas could also be used to model the gas dispersion and explosion. U.S. Army-sponsored tools for detonations include BlastX and CONWEP; certain NRC staff members have expertise in CONWEP. ALOHA is a DOE-sponsored code that could be used to analyze detonations caused by volatile dispersions from the hydrogen plant.

After the modeling framework is chosen, the next step involves assessing the evaluation model against the experimental data that are representative of the physical conditions of an NGNP blast. To this end, the calculation uncertainties, range of parameters, and simplifying assumptions must be thoroughly understood.

Using the chosen modeling framework, analysts then estimate the response of the reactor building structure to the impulse loadings attributable to the explosion. Structural properties to consider are the dynamic deformation of the reactor building structure; the shear, tensile, and compressive stresses inside the structure from the loading; and the structural capability of the reactor building to withstand the impulse loadings. The tools for assessing structural integrity might be a combination of analytical, experimental, and computational techniques. In this regard, the NRC possesses significant experience with finite element structural codes such as ABAQUS and LS-DYNA, which would be used in a deterministic analysis of reactor building integrity.

Other parts of the HTGR research assessment consider the effects of the other two categories, natural disasters and missiles, on the confinement. The NRC is actively researching the response of the reactor confinement to an airplane impact or missile attack and the structural research necessary for the NGNP in this area will be conducted in parallel with structural research for HTGRs and LWRs. In addition, the expectation is that the reactor building will be designed and constructed to meet NRC seismic loading and tornado loading requirements.

III.2.6.3.6 Internal Events

Damage to VHTR systems can occur because of a fire or explosion from volatile fluid leakage inside the reactor building. In a confined space, a fire or explosion might degrade the structural integrity of reactor systems such as the pressure vessel, cross-vessel, PCS, IHX, and other connecting helium pressure boundary piping. An analysis of a helium pressure boundary failure would also address potential air and/or water ingress into the core. If a confined explosion is sufficiently strong, an assessment also is required of the capability of the reactor building structure to reestablish low leakage capability (e.g., in response to damage to the reactor building ventilation dampers). The HTGR and VHTR reactor building ventilation dampers are relied on to limit both air ingress and FP release to the environment following an HPB break blowdown.

The fluid, combustion, and structural analysis for the confined case would pose a more complex problem than an analysis of an external unconfined explosion. When considering a confined combustion event, important factors include the transition from deflagration (thermal propagation) to detonation (shock propagation) and the reflected impulse and overpressure inside the confinement. The structural heterogeneity inside the confinement building might necessitate a three-dimensional treatment. As in the stress profile in the confinement building, the stresses within the structural components of the internals of the reactor building, the pressure vessel, cross vessel, PCS, piping, and IHX should also be computed.

Consequently, consideration should be given to selecting models that most closely model the combustion phenomena. Simplifying assumptions for the confined detonation and combustion might not be appropriate. One study (Madsen et al., 1994) indicates a need to accurately model the reflected overpressure from the blast wave.

III.2.6.3.7 Connectivity Hazards

In some NGNP concepts, the hydrogen plant is coupled to the VHTR through the IHX. Certain designs for the high-temperature electrolysis process involve an interface between the hydrogen/water mixture and the helium loop within the heat transfer surfaces of the IHX. If an external explosion does not significantly damage the outer structure, but induces a water hammer in the IHX, the inner coils within the IHX might fail. If a break in the IHX occurs, water and hydrogen might enter the reactor core, leading to a criticality excursion and FP release via the IHX.

The tools for analyzing the connectivity should be similar to those for the external and internal events, specifically a CFD code such as PHOENICS to predict the water/hydrogen pressure to the IHX, a finite element code such as LS-DYNA to predict the shear and normal stresses inside the coils of the IHX, and a criticality code such as SCALE to calculate the reactivity change with this event.

III.2.6.3.8 NGNP Damage State Framework

Specifically for the NGNP, the breadth of physical phenomena covered in an accident progression can result in several potential damage state figures of merit. For example, for an unconfined vapor explosion at the hydrogen plant, a suitable figure of merit could be the structural integrity (in terms of permissible deformations) of the reactor building (including ventilation dampers) integrity for the calculated impulse pressure-loading conditions.

In addition, the other proposed NGNP research topics must be conducted in parallel with the development of insights into the frequencies of each event. This will be needed to assess the event categorization (e.g., design-basis accident and beyond-design-basis accident), event analysis method (conservative or best estimate), and allowable consequences (e.g., structural limits). If the PIRT process yields a potentially high-consequence scenario, then the interests of physical understanding necessitate consideration of both a deterministic best estimate and conservative analyses.

III.2.6.3.9 NGNP Hydrogen Production Plant Security

The Homeland Security Appropriations Act of 2007 temporarily gave the U.S. Department of Homeland Security (DHS) the authority to regulate the security of certain chemical facilities that are classified as high risk. The risk classification label is the result of applying a vulnerability assessment methodology to the facility. The assumption is that the NGNP chemical process facility will be classified as high risk and thus will be subject to regulation by DHS at a minimum. Therefore, the NGNP production facility must be assessed in terms of the potential for introducing vulnerabilities to the VHTR. The analytical tools that would be developed should be used to identify a need for any physical security measures beyond those required by DHS.

III.2.6.4 Application of Research Results

The hydrogen production plant research area will be multifaceted. The first stage will be the identification of possible accident scenarios associated with operation of the NGNP; the next stage will include isolating the important physical phenomena that should be part of the engineering analysis. A detailed literature review will help to focus the selection of the most appropriate tools and methodology for this multidisciplinary analysis. The most important facet of the methodology will be the verification and validation of the selected tools for the identified range of accident conditions as well as the educated justification for the assumptions. The identification of suitable figures of merit for each different NGNP accident scenario will proceed in parallel with the deterministic analysis. The result will be a set of tools, data, and knowledge to support physics-informed and risk-informed regulatory decisions when the time arrives to license the NGNP.

III.2.7 Consequence Analysis

III.2.7.1 Background

Offsite consequence analysis is the final aspect of a PRA, the so-called Level 3. The mix of radionuclides and the chemical forms in the releases from severe accidents occurring in advanced reactors might differ from those in releases during accidents in LWRs. Therefore, comparisons of current and advanced technologies will require the comparison of full Level 3 analyses. Past evaluations of LWR technology issues have often stopped at the stage of large early release frequency.

III.2.7.2 Purpose

The normal input to the NRC Level 3 evaluation code, MELCOR Accident Consequence Code System 2 (MACCS2), is based on LWR technology. A review appears warranted to ensure a consideration of any important differences in user inputs to the code stemming from advanced reactor technologies. The outcome of this effort will be an NRC choice of site- and technology-specific input parameters for the Level 3 analysis.

III.2.7.3 Objectives and Associated Activities

MACCS currently considers 87 parent and daughter radionuclides. The impact on offsite consequences—in terms of early and latent fatalities, doses to specific organs, and economic consequences of these radionuclides—depends on their chemical forms. Dose conversion factors and other factors such as uptake in foodstuffs account for these chemical forms. If new biologically important radionuclides are produced, they must be added to the library. If new chemical forms are important, revised dose and uptake factors must 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.

III.2.7.4 Application of Research Results

Research results would be incorporated into the NRC 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) might be needed. The supporting calculations should be commensurate with the calculations used in choosing the current 10-mile EPZ for operating LWR plants. 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,” issued March 2002, refers to these calculations. This document also discusses choosing the size of the EPZ. 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,” discusses these calculations more fully.

III.3 Materials Safety and Waste Safety

III.3.1 Nuclear Analysis for Materials Safety and Waste Safety

III.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, 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, 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. Other sections of this document discuss nuclear analysis research for reactor safety and safeguards.

Although nuclear analysis is by not the only technical discipline of importance to the regulation of nuclear materials safety and waste safety, it is a key 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 sections are therefore cross-referenced, in footnotes, to other sections of this report that address related technical areas and that discuss multidisciplinary research activities from the perspective of systems and processes (e.g., fuel enrichment, fabrication, transport, storage, and disposal).

III.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 materials and waste safety, the expectation is that nuclear analysis issues will arise concerning (1) the out-of-reactor criticality safety analyses needed at the front end of the respective fuel cycles for HTGR designs and (2) the various safety analysis work that will be necessary for at-reactor storage and away-from-reactor storage, transport, and disposal of the spent fuels to be discharged from HTGRs.

III.3.1.2.1 Nuclear Criticality Safety at the Front End of the Fuel Cycle[5]

Enrichment plants, fuel fabrication facilities, and transportation packages for LEU commercial LWR fuel materials and fuel assemblies are not currently licensed to handle uranium enrichments significantly greater than 5 weight percent (wt%) ^{235}U. The expectation is that criticality validation issues will arise for HTGR materials safety because of the shortage of evaluated critical benchmark experiments involving neutron moderation by graphite and graphite with water, fuel materials with 5 to 19.9 wt% ^{235}U enrichment, and particle fuel geometries. In addition, technical studies might be needed to support the independent staff assessment of acceptable criticality modeling practices for HTGR particle fuel forms. For example, LEU fuel 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.

In addition, in the context of nuclear criticality safety, nuclear-grade graphite, which has a macroscopic thermal neutron absorption cross-section about 50 times smaller than that of light water, has been used as the moderating material in several heterogeneous reactor designs fueled by unenriched natural uranium (e.g., the first Chicago pile, the first Hanford production reactors, the Windscale production reactor, the British Magnox power reactors). The ability of graphite to form critical masses with natural uranium, and more generally with a wider range of fissionable material compositions than can be made critical with hydrogenous moderators (e.g., light water), is further characterized by the fact that the graphite neutron slowing-down power (i.e., the macroscopic scattering cross-section times the average neutron lethargy gain per scatter) is about 20 times weaker, and its neutron migration length about 5 to 10 times longer, than those for light water. The relatively weak slowing-down power and long migration lengths in graphite result in the need for relatively large volumes of material being needed to form such critical masses.

The development and deployment of HTGRs will give rise to a growing commerce in nuclear-grade graphitic materials and components for HTGR fuel elements, reflectors, and associated structures. The front-end fuel cycle activities (e.g., enrichment, conversion, fabrication, transport, storage) for HTGRs will thus include the planned or potential comingling of such graphitic materials with various fissionable materials. Given the wider ranges of fissionable material compositions potentially made critical by graphitic compared to hydrogenous (e.g., water) moderating materials as well as the higher enrichments

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5 See also separate sections on uranium enrichment and fuel fabrication (Section III.3.2) and transportation and storage (Section III.3.3).

(5 to 19.9 wt%) of proposed HTGR fuels, literature research and analytical studies might be necessary to ensure adequate identification and understanding of the material compositions, quantities, and configurations that potentially should be considered for ensuring criticality safety throughout the front-end and back-end HTGR fuel cycle activities. Subsequent sections of this report discuss in greater detail these and related studies potentially necessary to support criticality safety evaluations at the back end of the fuel cycle (e.g., spent fuel storage, transport, and disposal).

III.3.1.2.2 Safety Analyses for Spent Fuel Management[6]

The maximum discharge fuel burnup levels proposed for modern HTGR designs typically range from 80 to 150 GWd/t, significantly higher than the 45 to 60 GWd/t fuel burnup levels reached in current LWRs with their 5-percent maximum fuel enrichments. Nuclear analysis issues for storing, shipping, and disposing of the high-burnup spent fuels and underburned fuels discharged from HTGRs will involve the assessment of modeling assumptions and approximations, need for specific validation data, and validation of uncertainty treatments in the prediction of (1) high-burnup long-term decay heat sources for cooling, (2) associated 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. For at-reactor handling and storage as well as for away-from-reactor storage, transport, and disposal, significant economic and technical incentives might encourage requesting burnup credit in the criticality safety analyses for fuels discharged from HTGRs, which would result inenable burnup credit computational modeling and validation to becoming significant technical issues. In this context, the residual ^{235}U content (or residual enrichment) of discharged HTGR fuels can typically exceed 2 wt%, which is much higher than the residual enrichment in discharged LWR fuels (<1 wt%).

III.3.1.3 Objectives and Associated Activities

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

The sections below list 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.

III.3.1.3.1 Nuclear Data Libraries

- <u>Preparation, testing, and use of modern cross-section libraries</u>. See Section II.2.2.2 for additional information.

III.3.1.3.2 Nuclear Criticality Safety at the Front End of the Fuel Cycle

6 See also separate sections on transportation and storage (III.3.3) and waste disposal (III.3.4).

- Criticality safety evaluation, validation, and modeling guidance for HTGR fuel materials. (1) Perform literature research and analytical studies as needed to help identify any graphite-moderated fissionable material compositions that potentially should be considered in the criticality safety evaluations for proposed HTGR front-end fuel cycle activities (including material transportation) that would not be included in the fissionable material compositions thought to have criticality potential in the context of existing front-end fuel cycle activities. Provide the associated technical bases for developing any additional guidance or requirements that might be needed for preventing criticality events that would result from the comingling of such fissionable materials with graphite. (2) 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, fuel elements, and associated graphitic materials produced for the respective PBR and PMR HTGR design types. Develop options and recommendations for the evaluation and treatment of remaining validation uncertainties. Develop modeling recommendations for pebble and block-compact fuel forms to help ensure appropriate treatment of the resonance escape and self-shielding effects that make the particle fuel forms more reactive than simplified smeared models would predict. 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. Important elements of the modeling guidance to be developed for criticality safety analyses involving HTGR fuel materials and processes, including those for spent HTGR fuel as discussed below, will build on the related modeling studies for HTGR reactor physics described previously in Section III.2.2.2 of this report.

III.3.1.3.3 Safety Analyses for Spent Fuel Management

- Criticality safety evaluation, validation, and modeling guidance involving discharged HTGR fuel. (1) Perform literature research and analytical studies as needed to help identify any graphite-moderated fissionable material compositions that potentially should be considered in the criticality safety evaluations for proposed HTGR spent fuel activities (e.g., storage, transport, disposal) that would not be included in the fissionable material compositions thought to have criticality potential in the context of existing spent fuel activities. Provide the associated technical bases for developing any additional guidance or requirements that might be needed for preventing criticality events that would result from the comingling of such fissionable materials with graphite. (2) Identify and review existing and planned spent fuel isotopic assay databases as well as potentially relevant critical (and subcritical) benchmark experiments and then use sensitivity methods to assess their applicability for code validation in applying burnup credit to criticality safety evaluations involving spent fuel from the respective HTGR types. Develop options and recommendations for the evaluation and treatment of remaining validation uncertainties. Develop modeling recommendations for applying burnup credit to the respective HTGR 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 HTGR spent fuel. Building on closely related work on burnup credit (see previous item), reactor physics, and short-term decay heat sources for reactor safety (see Section II.2.2.2 of this report), 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 HTGR 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.

III.3.1.4 Application of Research Results

Results from the previously described research activities would be applied to enable and support the independent staff assessment of nuclear analysis issues associated with nuclear materials safety, waste safety, and material safeguards in HTGR reactor fuel cycles. As outlined in the preceding sections, the nuclear analysis research activities would result in the development of staff technical insights in these areas and the application of those insights to establishing independent review and analysis capabilities as well as any necessary new technical guidance or requirements. Development activities include the assessment of validation issues and modeling approximations to inform the staff evaluation and treatment of potential biases and uncertainties in the respective nuclear analysis areas. Especially important in this context are the development, testing, and use of state-of-the-art master cross-section libraries, as discussed in Section III.2 of this report.

III.3.2 Uranium Enrichment and Fuel Fabrication

III.3.2.1 Background

The fuel elements for proposed HTGRs will have higher uranium enrichments (i.e., 5 to 19.9 wt% ^{235}U) and physical characteristics substantially different from those of existing LWR types. Therefore, new fuel enrichment and manufacturing facilities are likely to be required. Current operating experience will provide valuable insights to the owners and managers of those facilities manufacturing fuel for HTGR designs as they consider the accumulated knowledge from 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, “Minimization of Contamination.” This activity is consistent with the Commission’s desire for risk-informed regulation.

III.3.2.2 Purpose

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

III.3.2.3 Objectives and Associated Activities

Reports to the NRC from the existing fuel manufacturing facilities need to be surveyed and evaluated for insights into potential improvements. The integrated safety analysis summaries that the fuel facilities will submit need to be reviewed. In addition, the fabrication processes and materials for pebble and compact-in-block HTGR fuels may present a larger fire hazard than those in existing fuel fabrication facilities. Section II.3.1 of this report identifies and discusses specific technical issues and research activities for criticality safety in facilities for enriching and fabricating the respective HTGR fuel materials and elements.

III.3.2.4 Application of Research Results

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

III.3.3 Transportation and Storage

III.3.3.1 Background

The fuel elements for PBR HTGRs will be enriched up to 9 wt% ^{235}U; those for PMR HTGRs may be enriched up to 19.9 wt% ^{235}U. Further, the graphite fuel pebbles are relatively small in size (6 cm in diameter, weighing 200 grams including 9 grams of uranium), very large in number, and not individually marked with identifiers. Prismatic HTGR fuel blocks are similar in width (~ 35 cm) to LWR fuel assemblies but shorter (~ 60 cm versus ~ 4 m).

The cylindrical fuel compacts within the prismatic graphite fuel element blocks are fabricated separately from the blocks. Because of the required effective retention of FPs in the contained TRISO-coated fuel particles (~ 0.9 millimeter in diameter), the subsequent separation of spent fuel compacts from the graphite blocks appears to be feasible by relatively simple means and with relatively little radioactive contamination, but has not yet been demonstrated. Industry research has started to investigate the different processes that would be needed for separating individual TRISO fuel microparticles, intact, from the matrix graphite of fuel pebbles. Without such separation of HTGR fuel material from fuel element graphite, the volume of spent fuel elements discharged per megawatt-year from HTGRs would be approximately an order of magnitude larger than that from LWRs, necessitating a similarly larger number of eventual spent fuel shipments and larger accommodations for spent fuel storage and eventual disposal.

Regulatory requirements and technical guidance documents already exist for (1) the packages and casks used in transporting fresh fuel and spent fuel under 10 CFR Part 71, “Packaging and Transportation of Radioactive Material,” (2) the at-reactor storage of fresh and irradiated fuel under 10 CFR Part 50, and (3) the storage of spent fuel in casks under 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.” However, HTGR 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 19.9 wt% versus 5 wt%). Furthermore, such technical issues as the assessment of high burnup (80–150 GWd/t) of HTGR spent fuel materials in storage and transport casks and the application of burnup credit in the criticality safety evaluations for HTGR spent fuels 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.

III.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 HTGR fuels will be evaluated.

III.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, is needed to determine continued applicability to advanced reactor fuels. Physical differences between existing LWR fuels and proposed HTGR fuels need to be considered. If the existing data and analyses do not apply to proposed fuels, then 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 evaluation of radionuclide inventories available for release, would be addressed through the nuclear analysis efforts described in Section III.3.1 of this report.

III.3.3.4 Application of Research Results

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

III.3.4 Waste Disposal

III.3.4.1 High-Level Waste

III.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 to meet the design objectives and support the development of regulatory requirements and compliance criteria. In the United States, sufficient performance assessment and analyses have not been performed with respect to HTGRs using TRISO-coated particle fuel (outer PyC, SiC, inner porous carbide layers surrounding the kernel uranium dioxide or oxycarbide fuel) and generating graphite and other types of HLW. For the disposal of radioactive waste generated by HTGRs, limitations exist 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 HTGR spent fuel and radionuclide source term releases from HTGR spent fuel under repository disposal conditions are not available for advanced reactor fuel having high enrichments (>5–9 percent and possibly up to 20 percent ^{235}U) and peak burnup levels of 90–150 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 that could significantly underestimate or overestimate individual exposure resulting from a complex disposal facility. In this case, opportunity and obligation exist to improve the NRC's performance assessment capabilities. The HTGR research program should address uncertainties associated with the disposal of HTGR spent fuel and reactor materials to improve the efficiency, effectiveness, and realism of agency analyses and decisions involving the performance of HLW repositories. Necessary research includes long-lived radionuclides source term releases, the behavior of TRISO spent fuel under the chemical environment of a HLW repository, higher fuel burnup and enrichment parameters, the effect of increased storage volumes of advanced reactor spent fuel and materials (e.g., graphite), a reevaluation of criticality codes, and the effect on transportation of an increased amount of advanced reactor spent fuel.

III.3.4.1.2 Purpose

The purpose of the advanced reactor waste disposal research plan 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 HTGR fuel from typical LWR fuel, determine radionuclide inventories for HTGR reactor fuel, understand HTGR fuel behavior under repository disposal conditions, and determine model, parameter and data uncertainties to estimate radionuclide source term releases. The research for HTGR fuel behavior and radionuclide source terms would focus on TRISO-coated fuel particles and other advanced reactor fuel types. Emphasis will be placed on obtaining experimental data and information under varying enrichment, burnup, and chemical disposal conditions. The research program is not expected to include research on the transport of radionuclides in the environment, biosphere pathways, and volcanism release scenarios. However, such research could be performed in those situations in which radionuclides present in HTGR fuel were found to be either different from or above the dose impact threshold of those radionuclides currently employed in performance assessments using typical LWR spent fuel.

Further research is needed to develop confidence in HTGR 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 HTGR spent fuel are not available, or if available, the data are generally of either poor quality or have been obtained under differing conditions than 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 HTGR spent fuel in a waste disposal repository and to update and modify source term computer codes used in HLW repository performance assessments. (A paper published in 2004 on the EURATOM HTGR Technical Network cooperative research described plans to test the long-term performance of the SiC layer in irradiated TRISO fuel particles in various types of geologic repository environments.)

Research in advanced reactor issues is also needed to understand the effects of the increased volume of waste generated by HTGR spent fuel. Unless fuel particles or fuel compacts can be separated from the graphite of the fuel elements at the reactor site or at a predisposal processing facility, the disposed spent fuel element volume for PBR HTGRs is expected to be 10 times higher, and for PMR HTGRs, 2–5 times higher, per megawatt day (MWD) than that generated by LWRs. The information will be used to reevaluate the transportation assumptions in risk studies on fuel transportation and to evaluate the source term implications of different storage configurations necessitated by larger volumes. Recent papers have been published on research and development of techniques for separating fuel compacts from prismatic HTGR fuel elements and separating coated particles from pebble matrix graphite.

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 HTGRs may require enrichment of up to 20 percent; however, current enrichment capacity at Paducah is only 5 percent and is limited by criticality safety issues. Current criticality codes are less well validated for enrichments above 5 percent and for combined moderation by carbon and water.

A discussion of the impacts of disposing HTGR spent fuel must address the criticality issues associated with the TRISO-coated particle fuel. The presence of silicon or silica might have an effect on the neutronics behavior of the criticality analysis. However, silicon or silica is much weaker than graphite in slowing down neutrons, and it also absorbs very few neutrons. In fact, silicon has such a small scattering cross-section that the neutronic effect of SiC in an HTGR fuel element is very minor and, in fact, almost negligible. Section II.3 of the research plan discusses nuclear analysis of criticality issues for the disposal of HTGR spent fuel.

III.3.4.1.3 Objectives and Associated Activities

Other activities discussed elsewhere in this plan will provide certain necessary information for this project. For instance, the nuclear analysis research will yield information about the inventories of long-lived radionuclides in HTGR fuels, the behavior and chemical form of HTGR spent fuels under varying enrichment and burnup conditions, and the validated criticality tools for higher enrichments and carbon+water moderation.

The HTGR-specific waste disposal research program has the following nine objectives:

(1) Characterize long-lived radionuclide inventories and chemical forms in HTGR spent fuel under varying enrichment and burnup conditions.

(2) Scope and determine radionuclide source term releases under varying burnup and repository chemical and physical conditions.

(3) Improve existing radionuclide source term models and computer codes for assessing the performance of a HLW repository containing HTGR spent fuels.

(4) Determine the releases of radionuclides from a repository containing HTGR spent fuel to the environment as a function of time up to 10,000 years and 1 million years.

(5) 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 to establish the calculational bias and uncertainty.

(6) Provide data in probabilistic distribution and associated statistical parameters format for use in risk-informed and performance-based assessments.

(7) Quantify chemical effects that may impact the parameters that control radionuclide releases, mobility, solubility, and sorption.

(8) Assess impacts of increased volume of HTGR waste.

(9) Evaluate HTGR-specific transportation needs.

Research is needed on the following tasks:

- Characterize long-lived FPs and transuranic nuclides in HTGR TRISO-coated particle spent fuel under varying enrichment and high burnup conditions.

- Determine the chemical form of radionuclides in high burnup HTGR TRISO-coated particle spent fuel.

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

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

- Determine release rates for FPs and transuranics from HTGR spent fuel under varying physical and chemical conditions.

- Determine solubilities of important radionuclides released from HTGR spent fuel in the chemical environment expected at a proposed repository.

- Obtain data on fuel rod or element corrosion/dissolution under repository chemical conditions. (This will be performed only when the fuel cladding or fuel element differs from the cladding materials used in current LWRs and disposed in a HLW repository (i.e., graphite fuel elements)).

- Evaluate repository near-field chemistry effects on HTGR spent fuel material behavior under varying burnup and fuel enrichments conditions.

- Determine radionuclide content and distribution in irradiated components (e.g., graphite) contemplated for HTGRs.

- Determine the presence of radiocolloids formed in the near-field of the repository as a result of cladding, structural material, and other materials present in the near-field of the repository interacting with radionuclides released from HTGR spent fuel.

- Study the behavior of irradiated HTGR fuel under geochemical disposal conditions and determine the long-term behavior of HTGR fuel in a geological repository. Experimentally study the behavior of irradiated HTGR fuel in the geochemical environments of the near-field repository conditions using long-term analytical experiments in which different radionuclide-chemical interactions of the different fuel barriers with the geochemical environment are tested separately. From the test results, develop mathematical models of these interactions and integrate them into computer codes. From these results, estimate the lifetime of an irradiated TRISO fuel particle over 1 million years.

- 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 HTGR releases.

- Assess increased waste volume storage and transportation needs.

- Reevaluate the assumptions of the NRC's Transportation Risk Study (NUREG-0170, "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes," issued in December 1977.) and NUREG/CR-6672, "Reexamination of Spent Fuel Shipment Risk

Estimates."

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

- Assess and validate criticality codes for enrichments greater than 5 percent ^{235}U.

- Study the reprocessing of HTGR TRISO-coated particle fuel. Develop a method for separating the fuel kernels from the silicon and carbon coating layers to enable reprocessing of HTGR fuel. (The "usual" reprocessing methods are feasible for HTGR UO_2 fuel only if the fuel kernel is fully separated from the carbonaceous and silicon coating layers). Investigate the feasibility of graphite recycling. Further information on the reprocessing of HTGR spent fuel will be found in the NRC's forthcoming spent fuel reprocessing research plan.

III.3.4.1.4 Application of Research Results

The NRC's HLW performance assessment computer codes will incorporate many of the results from this research. The research results are expected to provide an independent basis for evaluating and auditing licensees' HTGR-specific data, information analyses, models, and computer codes. The results are also expected to provide a base of physical data, information, and scientific expertise that the NRC staff can use to quantify uncertainties in the technical basis for licensing. In addition, the data and information are expected to reduce uncertainty and improve realism in the technical basis of licensing. Finally, the research results will support the development of regulatory criteria and resolve key technical issues associated with the licensing and approvals of a HLW repository containing HTGR spent fuel.

III.3.4.2 Low-Level Radioactive Waste and Site Decommissioning

III.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 HTGRs, as well as from decommissioning sites containing materials used in HTGRs. 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 to ensure compliance with the regulatory requirements of 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," 10 CFR Part 20, "Standards for Protection against Radiation," and the policies in NUREG-1757, "Consolidated Decommissioning Guidance," Volumes 1–3 and Supplement 1.

Determining radionuclide releases from HTGR 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 HTGR LLW disposed in LLW disposal sites and from decommissioning materials at decommissioned HTGR sites, the radionuclide inventories (both for the LLW and decommissioned materials), radionuclide releases, and solubilities must be considered. For LLW, additional data and information on waste types and forms and information on waste containers are required. Recent papers have been published on research into the radionuclides present in graphite waste streams from HTGRs and on the development of processes for decontaminating HTGR graphite wastes to minimize the volume of material disposed as radioactive waste

For decommissioning HTGR sites, data and information about planned decommissioning activities will be helpful in establishing specific decommissioning plan requirements. For example, certain radionuclides present in HTGR decommissioning materials may present a unique decommissioning challenge. For HTGRs, ^{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 pose a hazard from routine reactor operations. Novel decontamination methods may be required and higher decommissioning costs may be encountered when dealing with the decommissioning of HTGR. This highlights the need to consider decommissioning issues during the design phase in accordance with 10 CFR Part 20.

Uranium enrichment produces depleted uranium as a byproduct and, because of their higher enrichment needs, HTGRs may cause additional environmental impacts. For a typical LWR, 6–8 tons of depleted uranium are produced for each ton of fresh fuel. An HTGR could generate twice as much depleted uranium and create concerns about the impacts from disposal of depleted uranium mill tailings.

To allow for decay of radioactive material, PBMR Pty. proposes "safe store" of spent fuel on site for 40 years. However, since the September 11, 2001, incident, keeping the spent fuel on site appears to be a less attractive option.

III.3.4.2.2 Purpose

The HTGR LLW and decommissioning site research plan has the following 10 goals:

(1) Determine whether the radioactive waste generated from the operation of HTGRs and the long-lived radionuclides present in the waste differ from the waste and distribution of radionuclides in current NRC-licensed LWRs.

(2) Determine whether the 10 CFR Part 61 waste classification system makes HTGR waste streams ineligible for disposal in LLW disposal facilities.

(3) Estimate how much LLW is generated by HTGRs and determine the important waste streams and types of HTGR LLW.

(4) Determine whether long-lived radionuclides are present in HTGR LLW that are not present in LWR LLW.

(5) Assess the importance of activated metals in HTGR LLW.

(6) Evaluate how much transuranic waste LLW HTGRs generate.

(7) Determine whether packaging and shipping requirements must change.

(8) Assess disposal impact of higher enrichment on HTGR fuel depleted uranium.

(9) Assess decommissioning and plant operation hazards.

(10) Evaluate safe storage issues for waste generated by HTGRs.

III.3.4.2.3 Objectives and Associated Activities

The primary objective of the HTGR LLW disposal and site decommissioning research program is to (1) provide experimental data and information to be used to determine realistic radionuclide inventories, (2) calculate realistic radionuclide source term releases from LLW disposal facilities and decommissioned sites, (3) determine decommissioning, operation, and maintenance hazards, (4) evaluate the impacts of higher enrichment on depleted uranium, and (5) assess safe storage options.

Research is needed on the following tasks:

- Characterize HTGR LLW waste and decommissioning materials for radionuclide and chemical content.
- For LLW, determine radionuclide concentrations by HTGR waste stream, waste type, waste form, and waste classification.
- Identify differences between HTGR LLW streams and radionuclides and the LLW waste generated by current LWRs.
- Determine radionuclide releases, and the chemical and physical factors affecting releases, by performing laboratory and field leaching studies on HTGR waste, including activated metals and decommissioning waste materials.
- For important long-lived radionuclides that may be present in HTGR waste and not in typical LWR waste, determine radionuclide sorption coefficients using soil and groundwater typically found at LLW disposal facilities and decommissioning sites.
- Confirm probabilistic distributions and associated statistical parameters for radionuclide releases for use in risk-informed and performance-based computer codes.
- Develop an understanding of the issues involved in dismantling and disposing of graphite used for moderator, reflector, and other structural purposes.
- Determine whether ^{14}C, other radionuclides, or carbon dust presents a unique decommissioning challenge and poses a hazard during decontamination.
- Establish whether ^{110m}Ag or other radionuclides pose a hazard during routine HTGR operations.
- Assess the importance of using novel decontamination methods for HTGR decommissioning purposes.
- Develop assessment tools and evaluate the impact of high enrichment (up to 20 percent ^{235}U) on the quantity of depleted uranium produced from fuel fabrication processes.
- Determine suitable safe store parameters for spent fuel and waste materials from HTGRs.

III.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 HTGR operations and the decommissioning of HTGR sites, (2) provide an independent basis for evaluating applicant or licensees' data, information, analyses, and computer codes, (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 store assessments, and (4) reduce uncertainty and improve realism in the technical basis of licensing and regulation of LLW facilities and decommissioning sites.

III.3.5 Personnel Exposure Control during Operation[7]

III.3.5.1 Background

Because most of the facilities associated with advanced reactor concepts would be new facilities, the opportunity to design them with particular attention given to minimizing 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 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 downstream equipment. Furthermore, shielding designs for advanced reactors with graphite reflectors may develop streaming paths, posing a future exposure issue or vessel damage issue.

III.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 caused by changes in graphite geometry.

III.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,[8] and assess associated radiation streaming issues[9] in view of potential concerns over vessel fluence[10] as well as radiation protection. In addition, the work would evaluate different advanced reactor designs to identify any other issues that may pose radiological hazards that differ from those in conventional LWRs.

III.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.

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7 Applies to reactor safety as well as materials and waste safety.

8 See related activities described in the section on nuclear-grade graphite under Reactor Safety.

9 See related activities described in Section IV.3.1, Nuclear Analysis for Material Safety and Waste Safety.

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

III.4 Safeguards

III.4.1 Background

This research area addresses material and reactor safeguards, including the analyses and information needed for (1) assessing proliferation potential, material diversion vulnerabilities, sabotage vulnerabilities in facilities and material processes, and the potential radiologic and nonradiologic consequences of sabotage scenarios and (2) considering associated needs for facility and material security provisions and MC&A measures throughout the fuel cycles of the respective HTGR designs.

The fuel elements for PBR HTGRs will be enriched up to 9 wt% ^{235}U; those for PMR HTGRs may 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 (3–5 wt%) fuel for conventional LWRs. Further, the graphite fuel pebbles are relatively small in size (6 in cm diameter, weighing 200 grams including 9 grams of uranium), very large in number, and not individually marked with identifiers, thus making MC&A potentially more difficult. Prismatic HTGR fuel blocks are similar in width (~ 35 cm) to LWR fuel assemblies but are shorter (~ 60 cm versus ~ 4 m).

The cylindrical fuel compacts within the prismatic graphite fuel element blocks are fabricated separately from the blocks. Because of the effective retention of FPs in the contained TRISO-coated fuel microparticles (0.9 mm in diameter), the subsequent separation of spent fuel compacts from the graphite blocks appears to be feasible by relatively simple means and with relatively little radioactive contamination. Industry research has started to investigate the new processes that would be needed for separating individual TRISO fuel microparticles, intact, from the matrix graphite of fuel pebbles.

Without such separation of HTGR fuel material from fuel element graphite, the volume of spent fuel elements discharged per megawatt-year from HTGRs would be approximately an order of magnitude larger than that from LWRs, necessitating a similarly larger number of eventual spent fuel shipments and larger accommodations for spent fuel storage and disposal. As is also the case with the Canadian deuterium uranium reactor (CANDU) fuel technology, the relatively small size of the fuel elements for HTGRs enables the postulated use of small vehicles (e.g., cars or vans) for diverting intact fuel, whereas long trucks are needed for transporting intact LWR fuel assemblies.

III.4.2 Purpose

The purpose of HTGR research activities for the safeguards arena is to support the establishment of a technical basis for the staff's assessment of HTGRs 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 sabotage vulnerabilities throughout the HTGR fuel cycles, evaluated in relation to those in corresponding LWR fuel cycles, and the relative ease and potential consequences of diverting and misusing the respective HTGR 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.

III.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 HTGR 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 will be described for the respective HTGR technologies.

Specific activities include the following:

Material diversion safeguards. Nuclear analysis tools and methods need to be used in the arena of material diversion safeguards to assess weapons proliferation potential and radiological threats, material security technology, and the MC&A measures needed throughout the fuel cycles of the respective HTGR designs.

For example, PBR HTGRs use pebble fuel elements in a multiple-pass, continuous online fueling scheme; this configuration 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 that, in this context, the higher burnup levels (e.g., 90 GWd/t) of spent fuel from a PBRHTGR 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-m-long LWR fuel rods or assemblies, questions will arise about the potential for early discharge and diversion of standard fuel pebbles (i.e., with 4–9 percent 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.

Scoping studies on proliferation resistance of HTGR 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 that involve the introduction, early discharge, and diversion of standard fuel elements, as well as special plutonium-production fuel elements. Calculations to predict associated radionuclide inventories, including the quantities and isotopic compositions of plutonium produced per fuel element will need to be performed. Using credible assumptions regarding specific MC&A and material security measures, a comparison of the proliferation resistance of the PBR and PMR HTGR 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 radiologic weapons that use chemical explosives or other means for dispersing radioactive materials (i.e., dirty bombs). Recent analyses of these issues have been published by PBR HTGR development staff at INL and PBMR Pty.

Assessment of technical requirements for MC&A and material security in the PBR and PMR HTGR 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, 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 proposed PBRand PMR HTGR 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 HTGR fuel cycles.

III.4.4 Application of Research Results

Research in this area would provide safeguards reviewers with the expanded technical bases needed when considering potentially new HTGR-specific approaches and criteria for evaluating (1) sabotage and attack vulnerabilities and associated needs for preventive and mitigative measures in reactor plants and fuel cycle facilities and in the transport, storage, and disposal of spent fuel and waste, (2) domestic material safeguards and MC&A in the proposed HTGR fuel cycles, and (3) international material safeguards and MC&A for the nonproliferation of fission weapons.

III.5 Fast Liquid Metal Reactors

III.5.1 Background

LMRs use a liquid metal (usually sodium, lead, or a mixture of lead/bismuth) as the primary coolant. Heat from the liquid metal is transferred to water to produce steam in a liquid metal to water heat exchanger. LMR designs are basically of two types—(1) loop-type, where the primary coolant system (piping, pumps, heat exchangers) are located in a compact loop layout, outside of the reactor vessel and (2) pool-type, where the entire primary coolant system is located inside a reactor vessel. For both types, the primary coolant has a relatively large thermal inertia. LMRs are designed to have a large margin to coolant boiling. This is considered an important safety characteristic of LMRs. Sodium-cooled LMRs, for example, generally operate at near atmospheric pressure in the 480–540 °C (900–1000 °F) coolant outlet temperature range, which is significantly below the sodium boiling point of 900 °C (1650 °F) atmospheric pressure. Key safety characteristics of LMRs include the high thermal conductivity and high boiling point of the coolant (which promotes heat removal through conduction and natural circulation without the complications of a two-phase coolant). Additionally, reactor operation at near atmospheric pressure reduces primary pressure boundary stresses and lowers the potential for coolant leaks.

Single-phase liquid sodium heat transfer is highly efficient even at atmospheric pressure. By comparison, commercial pressurized-water reactor coolant must be pressurized to about 150 atmospheres to prevent departure from nucleate boiling, which could result in a decreased fuel-to-coolant heat transfer coefficient, increased fuel temperature, and potential fuel cladding failure. Sodium coolant provides significant core cooling margins at near atmospheric pressure because the boiling point of sodium is 300–400 °C

(575–750 °F) higher than the reactor core coolant operating temperature. Also, LMRs are designed with a large pool of sodium which significantly reduces the potential for coolant boil-off and core uncovery during an accident. By submerging the core in a large mass of liquid sodium, the core is also provided a very large heat sink which adds to its inherent safety characteristics. If the core starts to heatup, the pool can absorb a very large amount of heat before reaching the sodium boiling point.

LMRs also do not have a conventional emergency core cooling system. Instead, to prevent a loss of coolant that could result in the uncovery of the fuel in the core, LMRs are designed with a second vessel(s) which is referred to as a "guard vessel." The guard vessel surrounds the reactor pressure vessel (and for loop plants the guard vessel also surrounds the primary system pumps and heat exchangers). The guard vessel is designed to contain coolant that may leak from the reactor vessel system. In loop-type LMRs, the primary piping is also elevated relative to the core to ensure that the piping does not provide a low point in the system that could cause the coolant to drain from the core. LMRs generally rely on natural convection through a heat exchanger to remove decay heat.

Sodium reacts exothermically with air and water. Accordingly, LMRs are designed to reduce the probability and consequences of such reactions. Sodium-cooled LMRs generally have a secondary/intermediate sodium flow loop which is connected to the primary sodium loop via a heat exchanger. The secondary/intermediate loop separates the primary loop, containing radioactive sodium from the steam or water working fluid in the tertiary loop of the balance of plant system. The secondary/intermediate loop also prevents the sodium in the primary loop from reacting with water in the event of a steam generator tube leak. Additionally, the intermediate loop is operated at a higher pressure than the primary coolant system to prevent primary radioactive sodium from entering the intermediate system. When liquid lead or lead/bismuth is the coolant, an intermediate loop may not be used since these coolants do not react chemically with water. However, molten lead coming in contact with water could result in a steam explosion. Sodium is generally preferred as a core coolant because of its superior heat transfer properties and relatively low density.

III.5.2 Purpose

The purpose of this section is to survey the infrastructure needs for LMRs in the area of reactor systems analysis, which includes T/H analysis, nuclear analysis, and severe accident and source term analysis. Accident analysis considered will include events that fall within the licensing basis including DBEs and BDBEs (i.e., severe accidents).

There are several major T/H issues for LMRs. Many of these issues are related to the advanced LMR (ALMR) design based on the PRISM concept that uses sodium as the liquid metal coolant. However, these issues would also apply to other LMR designs using the same systems and components. Since several reactors using sodium are already in existence, a significant experience base exists for sodium-cooled reactors.

<u>Demonstration of passive safety design</u>. The physical phenomena and design features that are relied upon to achieve passively safe responses to design-basis transients and ATWS should be adequately characterized (e.g., axial thermal expansion of the fuel pins and radial expansion of the core grid plate

structure). A change in local geometry significantly affects local core reactivity. Research and development to evaluate these physical phenomena and design features and validate their models through experimentation would involve in-pile experiments using a transient test facility. Assurance of passive safety response, including the modeling and validation of models through experimentation, is an important technology issue.[7]

Electromagnetic (EM) pumps. The ALMR is designed with EM pumps for forced convection through the core. These pumps, which are unique to NPPs, are cooled by the surrounding sodium. They are constructed of a series of coils wrapped in insulation that generate an oscillating magnetic field. Since sodium is an electrical conductor, its movement in the magnetic field creates a motive force on the sodium. Since there are no moving parts, an EM pump alone has no coastdown to maintain a safe power-to-flow ratio when electrical current to the coils stops. Flow coastdown is provided by running a synchronous machine in parallel with the device. If power is lost, the synchronous machine (i.e., essentially a flywheel and generator) provides a voltage and current to the EM pump to cause a sufficiently slow coolant coastdown. The coastdown must maintain a power-to-flow ratio in the fuel such that large safety margins are sustained for the peak cladding and fuel pellet temperature. The response of the EM pumps during a loss of flow event is crucial to the fuel and thermal-fluid safety margins from these events.

There are two major areas with EM pumps for which a database is needed to accurately predict DBEs. The following areas need a computer model and a database for the phenomena.[5]

(1) For LMR designs that use EM pumps as the primary coolant pumps, their operation is important to safe operation. A sudden loss of pumping caused by a coil failure could lead to excessively high fuel temperatures and/or sodium boiling which can lead to large local reactivity insertions. A prototypical test would be needed to demonstrate that the coils that make up the EM pump have the projected life and reliability in terms of irradiation damage to the coils and the performance of the material insulating the coils.

(2) The coastdown curves used in the ALMR Preliminary Safety Information Document for the EM pump for all unprotected loss of flow (ULOF) events are calculated values. Data are needed to validate these coastdown curves. This would also be the case for any other LMR design that uses EM pumps.

Flow in the upper internal structure (UIS). Control rod drive line thermal expansion is a significant negative reactivity feedback that plays a major role in the safety analysis of several of the LMR deterministically selected DBEs (UTOP, ULOHS, ULOF). The coolant flows in and around the UIS that flow past the control rod drive line are an issue. Data are needed to quantify the flow rate and heat transfer to the control rod drive line during normal operation and off-normal conditions. The resulting thermal expansion of the drive line that inserts the control bundle into the core needs to be characterized as a function of position relative to the time in the fuel cycle.

The ultimate shutdown system (USS). The last revision of the PRISM design incorporated an alternate scram system known as the USS. The system is essentially a box of many small spherical boron carbide balls suspended above a hollow, central fuel-type channel within the core. The central channel has a small bypass flow through it. When activated, the bottom of the box opens to allow the boron carbide balls to fall into the central channel. The USS is supposed to function regardless of the central channel geometry and insert sufficient negative reactivity to terminate the core fission power. The concept will need a proof of principle demonstration test and a database to provide estimates for its activation time and rate of reactivity insertion.

<u>Sodium and water representation (two fluid)</u>. The steam generator tubes of a sodium-cooled LMR provide a boundary between the secondary sodium in the intermediate heat transport system (IHTS) and the higher pressure tertiary steam system. If one or more steam generator tubes fail, water will enter the sodium-filled IHTS. When the two fluids come in contact, an exothermic reaction will occur which could result in the failure of the IHTS pipe and potentially the failure of the IHX tube. Since the IHX would open the primary containment (i.e., the reactor vessel), an analytical code will be needed to predict the sodium/water interactions under these conditions, IHX pressurization, and the structural margins to IHX damage/failure. Similar consideration should also be given to lead-water reactions (i.e., steam explosions) in the case of a lead-cooled reactor.

<u>Leak detection</u>. One approach to addressing the sodium-water reaction safety issue in the steam generator would be to base its design on the successful approach used at EBR-II. The EBR-II steam generator employed a double tube wall heat exchanger. EBR-II had no tube leaks in about 30 years of operation. In this design, the sodium flows on the outside of the outer tubes, while the high-pressure steam would be on the inside of the inner tube. A small gap would be left between the two tubes, which could be filled with a porous wire mesh and helium. A leak detection system would monitor both the gap and the shell-side sodium. Moisture in the helium would indicate an inner-tube failure, while helium in the sodium would indicate an outer-tube failure. A sodium-water reaction could occur only if both tubes failed. A sodium dump system would actuate in such an event to alleviate this potentially damaging event. The Toshiba 4S design proposed for the village of Galena in Alaska has used this approach. For more information on sodium reactions with air and water and leakage detection, see the section on materials under Sodium.

<u>Two-phase sodium</u>. If sodium boiling can occur during a transient, a model is needed to track the boiling location and extent. Boiling will impact the heat transfer within the assemblies and the local reactivity insertion caused by the void generation. To evaluate events with sodium boiling, the code will need to represent a two-phase sodium boiling model with the appropriate constitutive package for bubble size, interfacial shear, interfacial heat and mass transfer, and two-phase friction multipliers.

<u>Multidimensional upper plenum</u>. The need for a three- or two-dimensional thermal-fluid model for the upper plenum during an accident calculation is not clear at this time. If the UIS which supports the control rod drive lines and in-vessel refueling machine were to have a complex flowpath to direct and wash the control rod drive lines with sodium from hot driver channels, then a two-dimensional T/H model would be needed in the upper plenum.

<u>Reactor vessel auxiliary cooling system (RVACS)</u>. Under normal operating conditions the sodium level in the air jacket between the reactor vessel and the vessel liner is relatively low. As such, during normal operation, only a small fraction of the reactor generated heat is transferred to the air jacket surrounding

the reactor vessel. However, once a postulated loss of flow event begins and the pumps are tripped, the sodium level in the air jacket increases until it matches that in the upper plenum. If the normal paths to reject heat through the IHX are lost, the decay heat can be rejected to the outside air through the RVACS. In addition to the increased conduction through the liquid metal, as the primary sodium heats up, its density decreases, causing the sodium to swell and flow over the vessel liner. This results in increased heat rejection to the RVACS by means of forced convection rather than just conduction through the sodium. On the air side, the increased heat rejection to the air increases its mass flow rate which allows for further increase in heat rejection.

This is both a model component that needs to be developed for computer codes analyzing an LMR event and a phenomenon that needs a database to establish its performance.

Upper plenum sodium level tracking. A model to track the sodium level in the upper plenum is required for simulation of LMRs that use the RVACS for passive cooling of the reactor vessel. During events when the RVACS is required, the sodium level swell in the upper plenum determines when the liner spill over begins. The entry of sodium into the RVACS greatly increases the heat transfer, and thus the heat rejection, in the RVACS. This phenomenon must be properly modeled to accurately calculate the temperatures in the vessel and core.

Auxiliary cooling system (ACS). The ACS is based on natural circulation air cooling of the steam generator. An air jacket surrounds the steam generators with a set of dampers at the inlet. During an event in which the water loop is lost, but the primary and intermediate loop are available and continue to transfer heat to the steam generator, the ACS can reject this heat can be rejected by air cooling by the ACS. While this may not be a safety-grade system and thus not assumed to function during many events, a model should be developed to analyze the system performance while the ACS is operating. This model will also be needed if a plant PRA were to become an important basis for LMR licensing.

Metal mass temperature model (thermal mass). The temperature and thermal expansion of several components during transient heatup such as reactor vessel, control rod drive, above-core load pads, and the lower core grid plate, are significant phenomena that any safety analysis calculation model must include.

Natural circulation model. The thermal-fluid code must calculate the flow rates associated with natural circulation in LMRs. The flows are driven by small density differences and are in the laminar regime. The Super System Code has been assessed in this area.[10,11]

Forced circulation model. The thermal-fluid code must have the appropriate models for heat transfer and friction factors in the higher Reynolds number regions associated with forced flow.

Balance of plant model. The balance of plant in an LMR is the tertiary loop in the system that contains the steam generator, feedwater pumps, piping to the turbine, and control system. This part of the system is not safety grade and is usually assumed to be unavailable during an accident. However, a computer code will need these models to better predict how the system will respond as a whole during normal operation.

In the Super System Code, the MINET code has models for these components and interfaces with the Super System Code during any calculation in which these models are activated.

Fuel assembly heat transfer. A computer code would need models for the flow and heat transfer within a fuel bundle. Extensive work has been performed in this area for EBR-II, FFTF, and CRBR. The Super System Code has models to represent these heat transfer phenomena.

Intermediate heat transport system. The IHTS, which is between the primary (radioactive sodium) system and the water loop where steam is produced, contains nonradioactive sodium. The effect that the IHTS has on the core inlet temperature makes its modeling important for many transients.

In addition to the research needs identified above, the NRC has also identified the following objectives and planned activities:

Related NRC research. Brookhaven National Laboratory developed the Super System Code series, comprising SSC-L for loop-type LMRs and SSC-P for pool-type LMRs, for the NRC in the late 1970s. This code series has many of the models required to evaluate LMRs. The code needs to be revisited to update its models and add new models, such as a two fluid model in the IHTS, two-phase sodium model, multidimensional model for the upper plenum if needed, and models for the EM pump, RVACS, and ACS. The Natural Convection Shutdown Heat Removal Facility at Argonne National Laboratory has demonstrated the concept of a passive decay heat removal system for LMRs and has validated code models. Further experiments should be conducted to enhance confidence in the performance of such passive cooling systems. For the multidimensional upper plenum, water simulation tests with 1/5-scale model using laser technology have been used for flow visualization. This and additional work done in this area need review.

Related international research.

Identified NRC research activities. The NRC needs an independent capability for LMR T/H analyses that has been thoroughly assessed and peer reviewed. Whether this effort will be focused on adding the necessary capability for LMR analysis to the Super System Code is yet to be determined.

III.5.3 Objectives and Associated Activities

For fast LMR designs and technology, the NRC conducted a technical infrastructure survey. at a higher level than the indepth HTGR assessment documented in this Enclosure. This assessment identifies key fast LMR technology safety and technical issues and needed areas for infrastructure research and development. The survey is intended to provide a framework for the scope and direction of a followup indepth LMR technology infrastructure assessment. When conducted, the followup fast LMR infrastructure assessment would need to be sufficiently detailed to identify specific safety research and development tasks that the NRC would need to conduct to support the review fast LMR applications.

III.5.4 Application of Research Results

Research will be applied to develop and demonstrate the ability to predict the behavior of the new LMR 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 such insights to the establishment and qualification of 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. PHENOMENA IDENTIFICATION AND RANKING TABLES

IV.1 Background

As part of the overall objective to prepare the 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 PIRT process will be used to focus resources on those tests and analysis that will contribute significantly to demonstrating, 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), identify and rank safety-significant phenomena and associated research needs through the sequential consideration of the following:

➜ 1. Plant Designs
➜ 2. Safety Analysis Figures of Merit
➜ 3. Representative Scenarios
➜ 4. Important Phenomena
➜ 5. Important Data and Models
➜ 6. Available Data and Models
➜ 7. Gaps in Available Data and Models

For a given design (e.g., a reactor system, fuel transport cask, storage facility), this kind of approach becomes risk-informed by employing PRA and/or other risk evaluation techniques (e.g., Hazops) to help guide and confirm the selection of representative scenarios or event sequences. Deterministic engineering judgment also needs to be applied in the selection of scenarios to bound uncertainties in the event probabilities and to provide a basis for the scenario used for the siting source term calculation.

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.

More recently, formal PIRT processes have been conducted in which a panel of outside experts was tasked with considering a limited set of scenarios or associated safety-related phenomena in a given system. Examples include the PIRT conducted for (1) ACR-700 nuclear analysis, T/H, and severe accidents, (2) HTGR TRISO-coated particle fuel performance, (3) AP-600 test and analysis needs, (4) performance of high-burnup LWR fuels in reactor accidents, and (5) using burnup credit to predict the subcritical margins for spent pressurized-water reactor fuel in shipping cask accidents.

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

IV.2 Initial Umbrella PIRT

For PBR and PMR HTGR designs, a team of NRC staff and contractors, whose collective areas of expertise could cover the full range of anticipated processes and phenomena for that reactor design, may develop a draft PIRT document. If developed, the resulting document could 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. Such an umbrella PIRT team would consist of NRC staff and/or contractors with expertise covering a broad range of technical areas (e.g., PRA, thermal and fluid flow, nuclear analysis, severe accidents, fuel fabrication and performance, FP transport, materials, SSCs, containment/confinement, human factors, I&C, maintenance and inspection).

For PBR and PMR HTGRs, such an umbrella PIRT activity could build upon results from (1) the October 2001 NRC Workshop on HTGR Safety and Research Issues, (2) the June 4, 2001, the Advisory Committee on Reactor Safeguards Subcommittee on Advanced Reactors meeting, (3) relevant NRC preapplication review and research efforts conducted most recently for PBMR and during the 1985–1995 timeframe for the DOE MHTGR design, (4) the "first-cut" NGNP reactor systems analysis PIRT prepared by INEL and KAERI, (5) recent NRC HTGR knowledge management efforts, including GRSAC scoping analyses and staff monitoring of HTGR-related IAEA and NEA activities, (6) NRC technology-neutral framework development efforts, and (7) the NRC PIRT on HTGR TRISO fuel performance and its supporting air-ingress accident scoping calculations at INL with the MELCOR code. Insights from the HTGR TRISO coated particle fuels PIRT an RES contractor's PIRT-like report on MHTGR safety evaluation modeling and data needs, could also be used.

Selected off-normal and accident event sequences should be chosen to represent the major safety-related processes and phenomena encountered in all anticipated licensing-basis events (design-basis accidents, beyond-design-basis accidents). The selected event sequences would be based upon the licensing-basis events that have been proposed by HTGR preapplicants, supplemented 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 the NRC and outside efforts. Accident sequences beyond the licensing basis may 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 should be addressed as needed for establishing accident initial conditions. These would include temperatures, pressures, flows, power densities, irradiated fuel conditions, and properties and dimensions of irradiated materials as well as FPs which are circulating and plated out on internal surfaces within the helium pressure boundary.

Results from any such 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 SSCs. 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 being assigned to efforts that address the largest gaps in the most safety-significant data and analysis tools.

IV.3 Continuing Umbrella PIRT Activities

Results from any strawman umbrella PIRT activities for HTGR systems may be peer reviewed, as needed, before publication of a formal 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.

IV.4 PIRT Activities

Following, and in some cases preceding or concurrent with, any umbrella PIRT, the NRC staff and contractors may conduct topical PIRT activities that focus on particular subgroupings of phenomena with their associated event sequences and affected SSCs.

First among the NRC's formal topical PIRT efforts for HTGRs was the PIRT activity completed in 2003 that 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.

As suggested by results from any umbrella PIRT exercises and other research efforts, additional topical PIRT efforts may be conducted to focus attention on such HTGR areas as (1) maximum normal fuel operating temperatures in relation to TRISO fuel performance, (2) reactivity and power transients, (3) helium hot jets and thermal striping affecting potential material fatigue and failures, (4) graphite oxidation, (5) passive conduction-cooldown decay heat removal, (6) high-temperature materials, (7) FP transport for containment/confinement performance issues, or (8) human factors and I&C. To help conserve limited resources and meet schedules, such topical PIRT exercises may initially be limited to small teams of NRC staff and contractors. As warranted and possible within resource and schedule constraints, some of these less formal PIRT exercises may be followed by a second phase which would include more formal PIRT panels or peer review processes.

In support of the analytical tools development and refinement safety research needed in several key areas of this infrastructure assessment, it is anticipated that PIRTs will need to be conducted to provide additional focus and enhance the quality and completeness of the infrastructure needs assessments documented in Section III.2 of this report for the VHTR designs that DOE is considering for the NGNP reactor. These PIRTs would be needed in the areas of (1) thermo-fluidics and accident analysis (including appropriate consideration for neutronics and criticality issues), (2) high-temperature materials, including graphite, (3) the reactor safety issues associated with the process heat and hydrogen production facility, and (4) FP transport and consequence analysis. The PIRT results would inform the decisions on the tool development and other safety research and development that the NRC would need to conduct in these technical areas to support the agency's review of an NGNP application.

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 of this report, the NRC will implement a systematic and logical PIRT process 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. The agency would have to give due consideration 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 preapplication 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.

V. COOPERATIVE RESEARCH

Unlike proven LWR technology for which extensive LWR-related operational worldwide experience exists, operational experience related to the GCRs is limited and that which is available may not be directly applicable. For instance, while the graphite-related AGR experience in the United Kingdom 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 versus 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 be considered for developing insights. However, such data may be applicable only to a limited extent and will have to be used with caution.

V.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 and expensive venture. Consequently, the NRC expects 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.

V.2 Relevant International Efforts

Extensive operational experience exists for GCRs in Germany and the United Kingdom, 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 the HTR-10 in China, HTTR in Japan, HFR in the Netherlands, and ATR at INL 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 vital to developing a thorough understanding of and establishing the necessary confidence in the HTGR design, safety, and technology issues.

V.3 Workshops and Meetings

Since SECY-03-0059 was issued, the staff has continued, on limited a basis, to participate in both national and international conferences, meetings, and workshops related to HTGR design, development, and research. Participation was principally aimed at maintaining cognizance of national and international research and development as part of the NRC's HTGR knowledge management activities. By its participation at these workshops and meetings, the staff has been able to remain current on worldwide HTGR technology design, development, and research. In this regard, the staff participated in the HTR-2004 and HTR-2006 International Topical Meetings on High-Temperature Reactor Technology, the IAEA Coordinated Research Project 5, Neutronic and Thermal-Hydraulic Benchmarks for High-Temperature Gas Cooled Reactors, IAEA Coordinated Research Project 6, Advances in HTGR Fuel Technology, International Nuclear Graphite Specialist Meetings, ASME Boiler and Pressure Vessel Code Meeting, and a Committee on the Safety of Nuclear Installations (NEA) meeting on HTGR licensing requirements with international regulatory organizations on AGRs.