INEEL/EXT-05-02581

Next Generation Nuclear Plant Research and Development Program Plan

Idaho National Laboratory
Oak Ridge National Laboratory
Argonne National Laboratory

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Idaho National Engineering and Environmental Laboratory
Bechtel BWXT Idaho, LLC

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Idaho Falls, Idaho 83415

Prepared for the
U.S. Department of Energy
Assistant Secretary for Nuclear Energy
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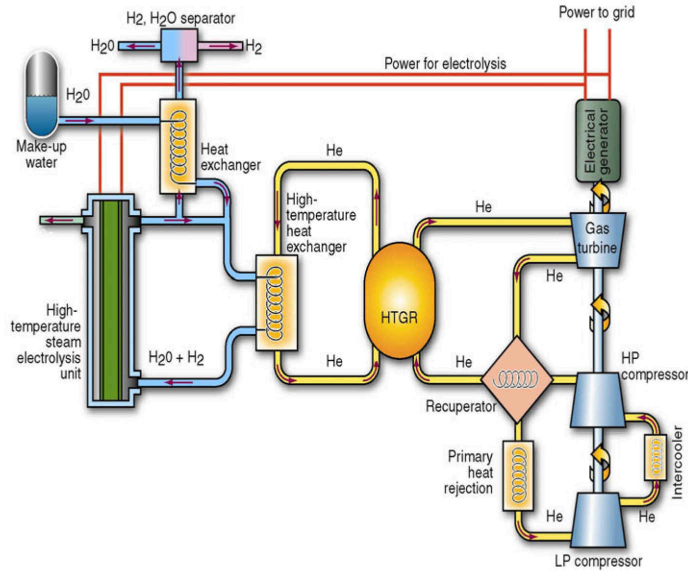
EXECUTIVE SUMMARY

The U.S Department of Energy (DOE) is conducting research and development (R&D) on the Very High Temperature Reactor (VHTR) design concept for the Next Generation Nuclear Plant (NGNP) Project. The reactor design will be a graphite moderated, thermal neutron spectrum reactor that will produce electricity and hydrogen in a highly efficient manner. The NGNP reactor core could be either a prismatic graphite block type core or a pebble bed core. Use of a liquid salt coolant is also being evaluated. The NGNP will use very high-burnup, low-enriched uranium, TRISO-coated fuel, and have a projected plant design service life of 60 years.



Sketch of how the NGNP might be configured to produce both electricity and hydrogen.

The VHTR concept is considered to be the nearest-term reactor design that has the capability to efficiently produce hydrogen. The plant size, reactor thermal power, and core configuration will ensure passive decay heat removal without fuel damage or radioactive material releases during accidents.

The objectives of the NGNP Project are to:

- Demonstrate a full-scale prototype VHTR that is commercially licensed by the U.S. Nuclear Regulatory Commission Demonstrate safe and economical nuclear-assisted production of hydrogen and electricity.

The DOE laboratories, led by the INL, will perform R&D that will be critical to the success of the NGNP, primarily in the areas of:

- High temperature gas reactor fuels behavior
- High temperature materials qualification
- Design methods development and validation
- Hydrogen production technologies
- Energy conversion.

The current R&D work is addressing fundamental issues that are relevant to a variety of possible NGNP designs. This document describes the NGNP R&D planned and currently underway in the first three topic areas listed above. The NGNP Advanced Gas Reactor (AGR) Fuel Development and Qualification Program is presented in Section 2, the NGNP Materials R&D Program Plan is presented in Section 3, and the NGNP Design Methods Development and Validation R&D Program is presented in Section 4. The DOE-funded hydrogen production [DOE 2004] and energy conversion technologies programs are described elsewhere.

Fuel Development and Qualification

Development and qualification of TRISO-coated low-enriched uranium fuel is a key R&D activity associated with the NGNP Program. The work is being conducted in accordance with the *Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program* [Bell et al. 2003]. The AGR Program includes work on improving the kernel fabrication, coating, and compacting technologies, irradiation and accident testing of fuel specimens, and fuel performance and fission product transport modeling. The primary goal of these activities is to successfully demonstrate that TRISO-coated fuel can be fabricated to withstand the high temperatures, burnup, and power density requirements of a prismatic block type NGNP with an acceptable failure fraction. It is assumed that TRISO fuel that is successful in a block reactor will also be successful in pebble-bed reactors since the particle packing fraction and the fuel temperatures are somewhat lower in pebble-bed reactors than in block reactors. In addition, commercialization of the fuel fabrication process, to achieve a cost-competitive fuel manufacturing capability that will reduce entry-level risks, is a secondary goal of the project.

An underlying theme for the NGNP/AGR fuel development and qualification work is the need to develop a more complete fundamental understanding of the relationship between the fuel fabrication process, key fuel properties, irradiation performance of the fuel, and release and transport of fission products in the NGNP primary coolant system. Fuel performance modeling and analysis of the fission product behavior in the primary circuit are important aspects of this work. Performance models are considered essential for several reasons, including guidance for the plant designer in establishing the core design and operating limits, and demonstrating to the licensing authority that the applicant has thorough understanding of the in-service behavior of the fuel system.

The AGR fuel development and qualification program consists of five elements: fuel manufacture, fuel and materials irradiations, post-irradiation examination and safety testing, fuel performance modeling, and fission product transport and source term modeling. Each task is discussed in some more detail below:

- ***Fuel Manufacture.*** The Fuel Manufacture task will produce coated-particle fuel that meets fuel performance specifications. This task also includes process development for kernels, coatings, and compacting; quality control (QC) methods development; scale-up analyses; and process documentation needed for technology transfer. Fuel and material samples will be fabricated for characterization, irradiation, and accident testing as necessary to meet the overall goals. Automated fuel fabrication technologies suitable for mass production of coated-particle fuel at an acceptable cost will also be developed. That work will be conducted during the later stages of the program in conjunction with a cosponsoring industrial partner.
- ***Fuels and Materials Irradiation***. The fuel and materials irradiation activities will provide data on fuel performance under irradiation as necessary to support fuel process development, to qualify fuel for normal operation conditions, and to support development and validation of fuel performance and fission product transport models and codes. It will also provide irradiated fuel and materials as necessary for post-irradiation examination and safety testing. A total of eight irradiation capsules have been defined to provide the necessary data and sample materials. The fuel irradiations will be conducted in the Advanced Test Reactor (ATR) located at the INL.
- ***Safety Testing And Post-Irradiation Examination.*** This task element will provide the equipment and processes to measure the performance of AGR fuel under accident conditions. This work will support the fuel manufacture effort by providing feedback on the accident-related performance of kernels, coatings, and compacts. Data from the post-irradiation examinations and accident testing will supplement the in-reactor measurements [primarily fission gas release-to-birth (R/B)] as necessary to demonstrate compliance with fuel performance requirements and support the development and validation of computer codes.

- ***Fuel Performance Modeling.*** The fuel performance modeling will address the structural, thermal, and chemical processes that can lead to coated-particle failures. The release of fission products from the fuel particle will also be modeled, including the effects of fission product chemical interactions with the coatings, which can lead to degradation of the coated-particle properties. Computer codes and models will be further developed and validated as necessary to support fuel fabrication process development. Results of these modeling activities will be essential to the plant designer in establishing the core design and operation limits, and demonstration to the licensing authority that the applicant has a thorough understanding of the in-service behavior of the fuel system.
- ***Fission Product Transport and Source Term Modeling***. This task will address the transport of fission products produced within the coated particles and the fuel element to provide a technical basis for source terms for AGRs under normal and accident conditions. The technical basis will be codified in design methods (computer models) validated by experimental data. This information will provide the primary source term data needed for licensing.

Materials Research and Development

The NGNP Materials R&D Program will focus on testing and qualification of the key materials commonly used in VHTRs. The materials R&D program will address the materials needs for the NGNP reactor, intermediate heat exchanger, and associated balance of plant. Materials for hydrogen production will be addressed by the DOE's Nuclear Hydrogen Initiative (NHI). Revision 1 of the *NGNP Materials R&D Program Plan* [Hayner et al. 2003] was issued in September 2004. The R&D discussed in this document is based on that plan.

The program is being initiated before the formal design effort to ensure that appropriate data will be available to support the NGNP design and construction process. The thermal, environmental, and service life conditions of the NGNP will make selection and qualification of some high-temperature materials a significant challenge; thus, new materials and approaches may be required. The following materials R&D areas are currently addressed in the R&D being performed or planned:

- ***Qualification and testing of nuclear graphite and carbon fiber/carbon matrix composites.*** Significant quantities of graphite have been used in nuclear reactors and the general effects of neutron irradiation on graphite are reasonably well understood. However, models relating structure at the micro and macro level to irradiation behavior are not well developed. Most of the past work was specific to a specific graphite known as H-451, which is no longer available. Therefore, the currently available nuclear grade graphites must be tested and qualified for use in the NGNP.
- ***Development of improved high-temperature design methodologies.*** The High-temperature Design Methodology (HTDM) project will develop the data and simplified models required by the ASME B&PV Code subcommittees to formulate time-dependent failure criteria that will ensure adequate high-temperature metallic component life. This project will also develop the experimentally based constitutive models that will be the foundation of the inelastic design analyses specifically required by ASME B&PV Section III, Division I, Subsection NH. Equations are needed to characterize the time-varying thermal and mechanical loadings of the design. Test data are needed to build the equations. The project will directly support the reactor designers on the implications of time-dependent failure modes and time and rate-dependent deformation behaviors. The project will also develop data for regulatory acceptance of the NGNP designs.
- ***Expansion of American Society of Mechanical Engineers (ASME) Codes and American Society for Testing and Materials (ASTM) Standards to support the NGNP design and***

construction. Much of this effort will provide required technological support and recommendations to the Subgroup on Elevated Temperature Design (NH) as they develop methods for use of Alloy 617 at very high temperatures. ASME design code development is also required for the graphite core support structures of the NGNP and later for the C_f/C composites structures of the core. A project team under Section III of ASME is currently undertaking these activities.

- ***Improving understanding and models for the environmental effects and thermal aging of the metallic alloys.*** The three primary factors that will most affect the properties of the metallic structural materials from which the NGNP components will be fabricated are the effects of irradiation, high-temperature, and interactions with the gaseous environment to which they are exposed. This work is focused on assessing the property changes of the metallic alloys as a function of exposure to the high-temperature and impure gas environments expected in the NGNP.
- ***Irradiation testing and qualification of the reactor pressure vessel materials.*** Some VHTR designs assume the use of higher alloy steel than currently used for LWR pressure vessels. The irradiation damage and property changes of these materials must be measured. Therefore, an irradiation facility that can accommodate a relatively large complement of mechanical test specimens will be installed in an appropriate material test reactor and used.
- ***Qualification and testing of the silicon carbide fiber/silicon carbide matrix composite materials needed for the NGNP.*** This program is directed at the development of C/C and SiC/SiC composites for use in selected very high temperature/very high neutron fluence applications such as control rod cladding and guide tubes (30 dpa projected lifetime dose) where metallic alloy are not feasible. It is believed that SiC/SiC composites have the potential to achieve a 60-year lifetime under these conditions. The usable life of the C/C compositiies will be less, but their costs are also significantly less. The program will eventually include a cost comparison between periodic replacement of C/C materials and use of SiC/SiC composites.
- ***Assessment of fabrication and transportation issues relating to the NGNP reactor pressure vessel.*** Materials issues associated with joining and inspecting heavy section forgings are covered in this task. This will initially be a scoping study to determine general transportation and fabrication issues associated with construction of the VHTR.
- ***Development of a materials handbook/database to support the Generation IV Materials Program.*** This is required to collect and document in a single source the information generated in this and previous VHTR materials R&D programs.
- ***NGNP reactor pressure vessel emissivity.*** The emissivity and other physical and mechanical properties of layers that form either by high-temperature environmental exposure or artificially engineered layers on the exterior surface of the NGNP reactor pressure vessel will be measured. These data are needed for off-normal and accident condition assessments.

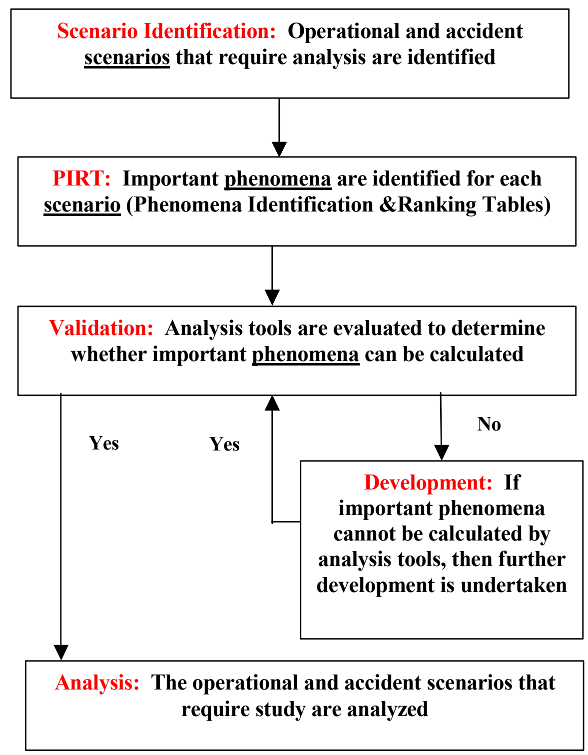
Not all of these program elements will be addressed in FY-05, due to limited funding; however, we envision that all of these areas will be addressed in the outyears. The Materials program plan will be updated periodically to reflect changes made to the NGNP program.

Design Methods Development and Validation

One of the great challenges of studying, designing, and licensing the NGNP is to confirm that the intended NGNP analysis tools can be used with confidence to make decisions and to assure all that the reactor systems are safe and meet their performance objectives. The R&D projects outlined in Section 4 of this document will ensure the tools used to perform the required calculations and analyses can be trusted.

Revision 0 of the *NGNP Design Methods Development and Validation R&D Program Plan* [Schultz et al. 2004] was issued in August 2004. The R&D discussed in this document is based on that plan and focuses on developing tools for assessing the neutronic and thermal-hydraulic behavior of the plant. The fuel behavior and fission product transport models are discussed in Section 2. Various stress analyses and mechanical design tools will also need to be developed and validated. Those tools will be addressed in a subsequent revision of this program plan.

The overall methods development process is outlined in the figure to the right. The requirements associated with scenario identification, defining the phenomena identification and ranking tables, completing the required development, and performing the necessary validation studies must all be completed before performing the required analyses confidently. The NGNP design has not yet been selected. Consequently, the R&D process is focused on scenarios and phenomena identified as important by the advanced gas-cooled reactor community, in the past.



Methods development process

The calculational and experimental needs, and consequently the required R&D, will be focused in eight distinct areas, based on the relative state of the software in each:

1. Material cross section compilation and evaluation
2. Preparation of homogenized cross-sections
3. Whole-core analysis (diffusion or transport), detailed heating calculation, and safety parameter determination
4. Thermal-hydraulic and thermal-mechanical evaluation of the system behavior
5. Models for balance of plant electrical generation system and the hydrogen production plant
6. Fuel behavior and fission product release
7. Fission product transport.

The R&D described in Section 4 of this document focuses on Areas 1 through 5. The fuel behavior and fission product transport is discussed in Section 2 with the other fuel related R&D. Based on the above areas, R&D projects have been defined.

CONTENTS

ACRONYMS

AGCNR	Advanced Gas-Cooled Nuclear Reactor
AGR	Advanced Gas-Cooled Reactor
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AVR	Arbeitsgemeinschaft Versuchsreaktor
B&PV	Boiler and pressure vessel
BET	Brunauer Emmett Teller (surface area measurement technique)
BWR	boiling water reactor
CEA	Atomic Energy Commission (France)
C_f/C	Carbon/carbon Composite
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CRBRP	Clinch River Breeder Reactor Project
CTE	Coefficient of thermal expansion
DCC	Depressurized conduction cooldown scenario
DLOF	Decompression Loss of Fluid Accident
DOE	U. S. Department of Energy
ENDF	Evaluated Nuclear Data Files
EOI	Expression of Interest
EUROFER	Specific European name of a steel alloy
FSAR	Final Safety Analysis Report
FSV	Fort St. Vrain
FTE	full-time equivalent

FY	fiscal year
GA	General Atomics
GDC	General Design Criteria
GFR	Gas-cooled Fast Reactor
GIF	Generation IV International Forum
GSI	Generic Safety Issue
GTHTR	Gas Turbine High Temperature Reactor
GT-MHR	Gas Turbine-Modular Helium Reactor
HENDEL	Helium Engineering Demonstration Loop
HFIR	High-Flux Isotope Reactor
HHT	High-temperature helium turbine system
HPC	High-Pressure Compressor
HTDM	high-temperature design methodology
HTGR	High Temperature Gas Reactor
HTR	high-temperature reactor
HTR-10	Chinese High Temperature Gas-Cooled Reactor
HTTR	High-Temperature Engineering Test Reactor
IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory (currently the Idaho National Engineering and Environmental Laboratory)
IPNS	Intense Pulsed Neutron Source
ITRG	Independent Technology Review Group
JAERI	Japan Atomic Energy Research Institute
KAERI	Korean Atomic Energy Research Institute
KFA	Kernforschungsanlage Julich (Institute for Chemical Technology, Germany)
LANSCE	Los Alamos Neutron Science Center
LMR	Liquid-metal reactor

LPC	Low-Pressure Compressor
LWR	light water reactor
MCNP	Monte Carlo physics code
MER	Materials for Energy Research
MHI	Mitsubishi Heavy Industries
MRC	INL Materials Review Committee
MWe	megawatts-electrical
MWt	megawatts-thermal
NACE	National Association for Corrosion Engineers
NDE	nondestructive examination
NE	DOE Office of Nuclear Energy
NGNP	Next Generation Nuclear Plant
NJOY	reactor assembly cross-section preparation computer code
NNC	National Nuclear Corporation (Great Britain)
NPH	Nuclear process heat
NRC	U.S. Nuclear Regulatory Commission
NTD	National Technical Director
ODS	Oxide dispersion strengthened
OECD/NEA	Organization of Economic Cooperation and Development - Nuclear Energy Agency
ORNL	Oak Ridge National Laboratory
PBMR	Pebble-bed modular reactor
PCC	Pressurized conduction cooldown scenario
PCU	Power conversion unit
PIRT	phenomena identification and ranking table
PMB	GIF VHTR Project Management Board
PMR	Prismatic modular reactor

PNNL	Pacific Northwest National Laboratory
PNP	Prototype Nuclear Process Heat
PSAR	Preliminary Safety Analysis Report
PSI	Paul Scherrer Institute (Research institute in Switzerland
PV	pressure vessel
PWR	Pressurized Water Reactor
QA	Quality Assurance
QAP	Quality Assurance Program
R&D	Research and Development
RCCM	European design code
RCCS	Reactor cavity cooling system
RCMR	European design code
SCS	Shutdown cooling system
SG-ETD	Subgroup on Elevated Temperatures Design
SGL	Name of a graphite company
SiC_f/SiC	silicon-carbide/silicon-carbide composite
SNECMA	A manufacture of C/C and SiC/SiC composites
THTR	Thorium Hochtemperatur Reaktor
TRISO	tri-isotopic ceramic-coated-particle fuel
UCAR	Name of a graphite company that is wholly owned by Graftek
UK	United Kingdom
USCSG	Ultra-Supercritical Steam Generator
VHTR	Very High Temperature Reactor

Next Generation Nuclear Plant Research and Development Program Plan

1. INTRODUCTION

In the approaching decades, the United States, the other industrialized countries, and the entire world will need energy and an upgraded energy infrastructure to meet the growing demands for electric power and transportation fuels. Anticipating this critical need, the Generation IV international Forum identified nuclear energy system concepts for producing electricity that excel at meeting the goals of superior economics, safety, sustainability, proliferation resistance, and physical security [GIF 2002]. One of these concepts—the Very-High-Temperature Gas-Cooled Reactor (VHTR)—is uniquely suited for producing hydrogen without consuming fossil fuels or emitting greenhouse gases because of its high outlet temperature. Working with its international partners in the Generation IV International Forum, the U.S. Department of Energy has selected this system for the Next Generation Nuclear Plant (NGNP) project, a project to demonstrate emissions-free nuclear-assisted electricity and hydrogen production by 2017.

The VHTR performance goals are:

- Plant overnight construction cost: <$1,000/kW,
- Electricity generation cost: <1.0¢/kW-hr, and
- Hydrogen cost: <$1.50/gallon – gasoline equivalent.

1.1 Hydrogen Production - A Major Administration Initiative

"Hydrogen holds the potential to provide a clean, reliable, and affordable energy supply that can enhance America's economy, environment, and security" [DOE 2002]. The U.S. hydrogen industry currently produces 9 million tons of hydrogen per year for use in chemicals production, petroleum refining, metals treating, and electrical applications. Nine million tons of hydrogen per year is enough to fuel 20 to 30 million fuel cell cars, or enough to power 5 to 8 million homes [DOE 2002]. The current use is experiencing rapid growth as more and more hydrogen is used to convert the lower-cost Western Hemisphere heavy crude oils to gasoline.

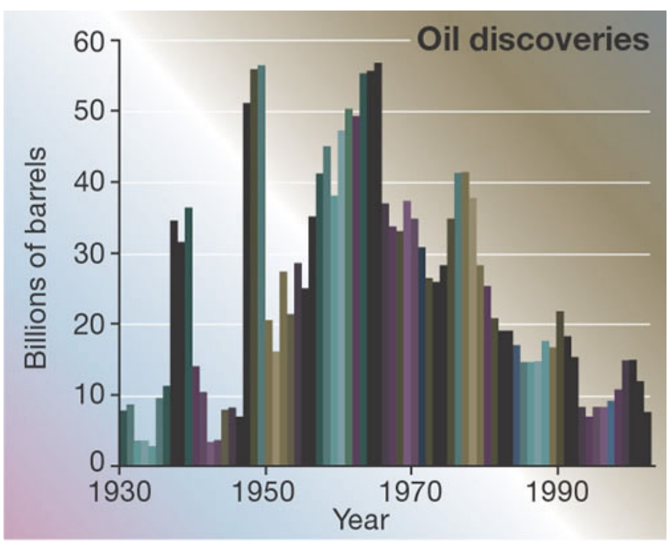


Figure 1-1. Worldwide oil discoveries plotted in billions of barrels of oil per year.

As shown in Figure 1-1, oil is not being discovered at a rate sufficient to meet demand [*Nature* 2004]. We are heading for a hydrogen transport economy; the only questions are (1) the form of hydrogen in the vehicle (gasoline, methanol, hydrogen, etc.) and (2) where it is used (refinery, tar sands plant, vehicle).

Although hydrogen is the most abundant element in the universe, it does not naturally exist in its elemental form in large quantities or high concentrations on earth. Steam reforming of methane accounts for more than 95% of the current hydrogen production in the United States.

Unfortunately, steam methane reforming diverts valuable natural gas from home heating uses and releases large quantities of carbon dioxide into the atmosphere. Hydrogen production currently uses 5% of the natural gas consumed in the United States. A much more environmentally friendly method of producing hydrogen would be to crack water at high temperatures using nuclear heat or solar energy.

1.2 Very-High-Temperature Gas-Cooled Reactor

The NGNP reference concepts are helium-cooled, graphite-moderated, thermal neutron spectrum reactors with an outlet temperature of up to 1000 °C [MacDonald et al. 2004]. As indicated in Figure 1-2, an outlet temperature near 1000 °C will allow the reactor to be used for a large number of process heat applications, including hydrogen production.

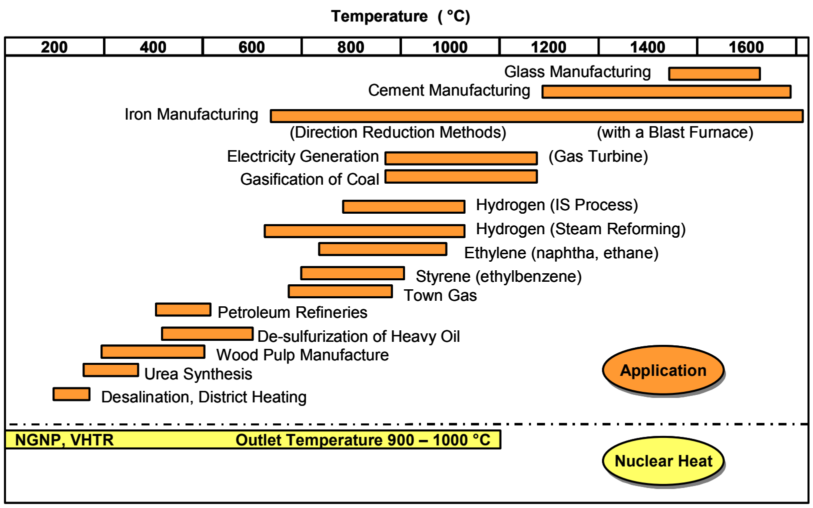


Figure 1-2. Temperature requirements for various process heat applications.

The NGNP reactor core could be either a prismatic graphite block type core or a pebble bed core. Use of a liquid salt coolant is also being evaluated. The NGNP will produce both electricity and hydrogen. The process heat for hydrogen production will be transferred to the hydrogen plant through an intermediate heat exchanger. The reactor thermal power and core configuration will be designed to ensure passive decay heat removal without fuel damage during hypothetical accidents. The fuel cycle will be a once-through very high burnup low-enriched uranium fuel cycle.

The basic technology for the NGNP has been established in former high-temperature gas-cooled reactor plants (e.g., DRAGON, Peach Bottom, Albeitsgemeinschaft Versuchsreaktor [AVR], Thorium Hochtemperatur Reaktor [THTR], and Fort St. Vrain). These reactor designs represent two design categories: the pebble bed reactor and the prismatic modular reactor. Commercial examples of potential NGNP candidates are the Gas Turbine-Modular Helium Reactor (GT-MHR) from General Atomics [GA 1996], the High Temperature Reactor concept from AREVA [Copsey et al. 2004], and the Pebble Bed Modular Reactor (PBMR) from the PBMR consortium [Nicholls 2000]. Furthermore, the Japanese High-Temperature Engineering Test Reactor (HTTR) and Chinese High-Temperature Reactor (HTR) are demonstrating the feasibility of the reactor components and materials needed for the NGNP. (The HTTR reached a maximum coolant outlet temperature of 950 °C in April 2004.) Therefore, the NGNP program focuses on building a plant to publicly demonstrate the safety and economics of the VHTR, rather than simply confirming the basic feasibility of the concept.

One or more technologies will use heat from the high-temperature helium coolant to produce hydrogen. The first technology of interest is the thermo-chemical splitting of water into hydrogen and oxygen. There are a large number of thermo-chemical processes that can produce hydrogen from water, the most promising of which are sulfur based and include the sulfur-iodine, hybrid sulfur-electrolysis, and sulfur-bromine processes. Schematics of those three processes are shown in Figure 1-3. Note that all three processes require the heat to be delivered at a temperature of at least 850 °C.

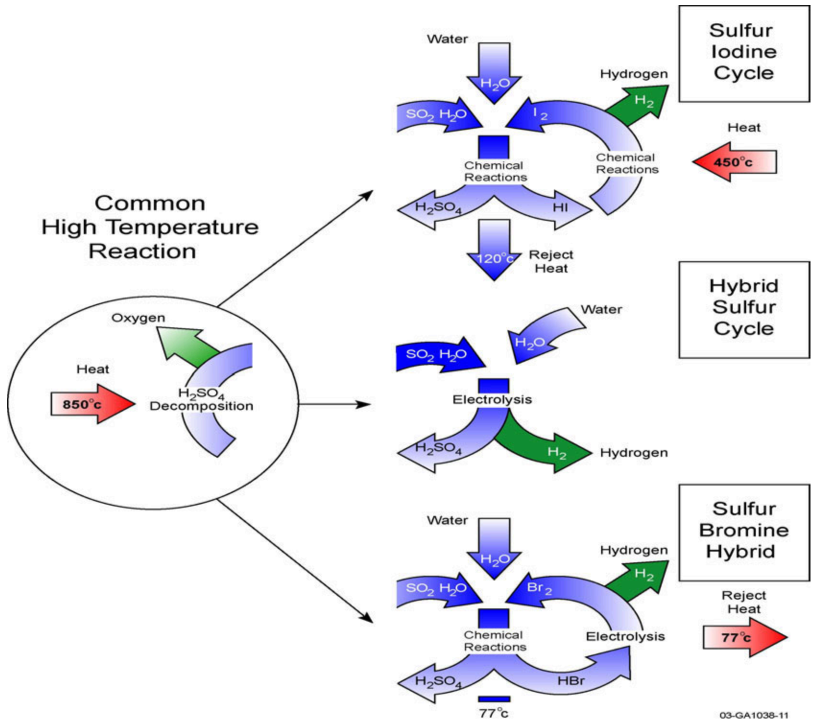


Figure 1-3. Schematic of the sulfur-iodine, hybrid sulfur-electrolysis, and sulfur-bromine processes

The second technology of interest is thermally assisted electrolysis of water. The high-efficiency Brayton cycle enabled by the NGNP may be used to generate the hydrogen from water by electrolysis. The efficiency of this process can be substantially improved by heating the water to high-temperature steam before applying electrolysis.

1.3 NGNP Implementation

The objectives of the NGNP Project are to:

- Demonstrate a full-scale prototype VHTR that is commercially licensed by the U.S. Nuclear Regulatory Commission Demonstrate safe and economical nuclear-assisted production of hydrogen and electricity.

The DOE published a draft strategy for developing and demonstrating the NGNP on May 27, 2004. Informed by public comment on the draft strategy, the Department is now formulating its final acquisition strategy. The strategy is designed to maximize industrial and international cost-shared participation. The Department contemplates the use of a financial assistance vehicle to enter into a cost-shared cooperative agreement with an industrial partner.

The preliminary rough order of magnitude cost estimate range for the NGNP is $1.8 to 2.4 billion. The scope of the work will be divided into five phases with provisions for go/no-go decisions at the end of each phase:

1. Project integration and formation
2. Research and preliminary design
3. Development and final design
4. Construction and viability testing
5. Project close-out.

The following tasks will be completed in Phase 1:

- Conduct a design competition for the NGNP reactor technology and select the NGNP technology
- Prepare a "Research and Development Plan" that supports the selected technology
- Prepare a "Fuel Plan" detailing the acquisition of licensed fuel for the NGNP
- Prepare a "Business Plan" that details the successive phases of the project and identifies the members of the international consortium that will cost share the project and lead its development.

The DOE will be substantially involved in the technology selection process and have ultimate approval. The Department will require quarterly reporting in accordance with 10CFR600, and the program will be run consistent with DOE Order 413.3

The Department will provide its laboratory system, led by the Idaho National Laboratory (INL), as the principal resource for conducting NGNP R&D. The DOE laboratories will perform R&D that will be critical to the success of the NGNP, primarily in the areas of:

1. High temperature gas reactor fuels behavior
2. High temperature materials qualification
3. Design methods development and validation
4. Hydrogen production technologies
5. Energy conversion

This document describes the NGNP R&D planned and currently underway in the first three of these five research areas. The organization of the work is shown in Figure 1-4. The NGNP Advanced Gas Reactor (AGR) Fuel Development and Qualification Program is presented in Section 2; the NGNP Materials R&D Program Plan is presented in Section 3; and the NGNP Design Methods Development and Validation R&D Program is presented in Section 4. The DOE-funded hydrogen production [DOE 2004] and energy conversion technologies programs are described elsewhere. The current R&D work is addressing fundamental issues relevant to a variety of possible NGNP designs. When the NGNP technology is selected, the DOE R&D will focus on that selection.

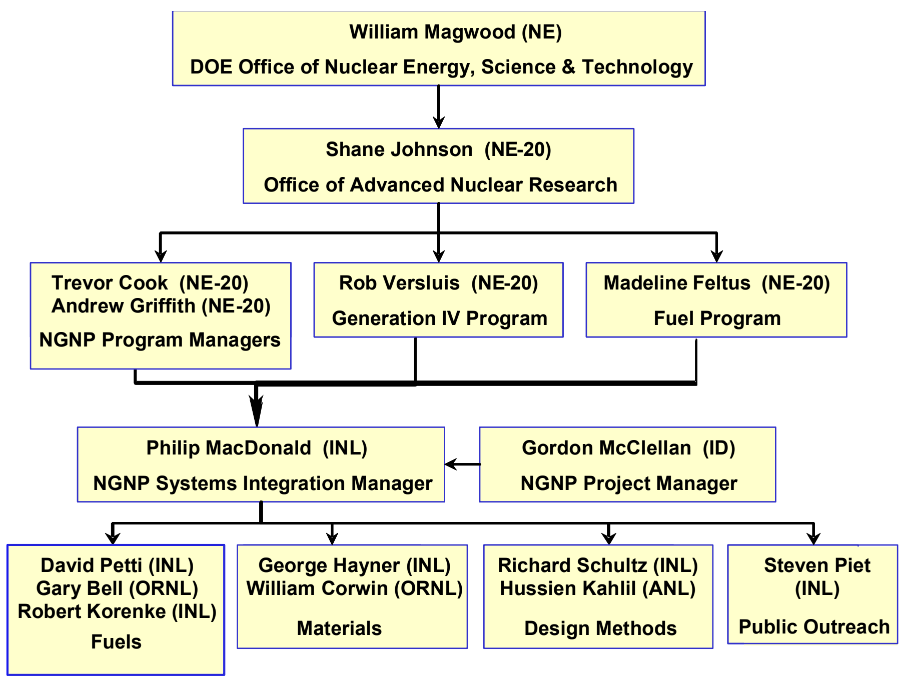


Figure 1-4. NGNP R&D Program organization.

1.4 International Collaborations

The Generation IV International Forum (GIF) was formed to define and implement an advanced generation of nuclear energy systems. Ten countries and Euratom have come together in the GIF to develop future-generation nuclear energy systems that can be licensed, constructed, and operated to offer competitively priced and reliable energy products while satisfactorily addressing nuclear safety, waste, proliferation, and public perception concerns. The participating countries are Argentina, Brazil, Canada, France, Japan, the Republic of Korea, the Republic of South Africa, Switzerland, the Unite Kingdom, and the United States. In 2002, the GIF identified six reactor systems that would best support these goals and established System Steering Committees for four of the six reactor systems.

These steering committees are charged to

- Guide the development of the system, including the definition of the system baseline and the required technologies.
- Assess the system designs versus the Generation IV goals and criteria for the purpose of evaluating the R&D progress and the merits of their system versus others.
- Establish and maintain an R&D plan for the long-term development of their system.
- Integrate and provide oversight for R&D projects that are in the R&D Plan. This includes defining the work scope required, reviewing proposed schedules and budgets, recommending performing Member institutions, and defining and maintaining important interfaces with other projects.
- Conduct the program in a manner that allows individual GIF Members to manage their intellectual property within their own established rules and guidelines. The Steering Committee does not have a role in managing intellectual property.
- Review and recommend participation levels of GIF Members and utilization of resources from outside the GIF.
- Request and review summary information about the R&D projects within the program for their system.
- Annually report on the progress of R&D for their system to the Policy Group.

The VHTR System Steering Committee shown in Figure 1-5 will advance the VHTR viability and performance by coordinating R&D among the member countries, thereby allowing maximum advancement for the dedicated resources. Timing of the R&D will also be coordinated in order to best leverage each country's contribution.

To conduct this detailed level of coordination of scope and schedule, Project Management Boards report to the VHTR Steering Committee, having been established to define collaborations in specific areas. Four project management boards are now active for the VHTR—

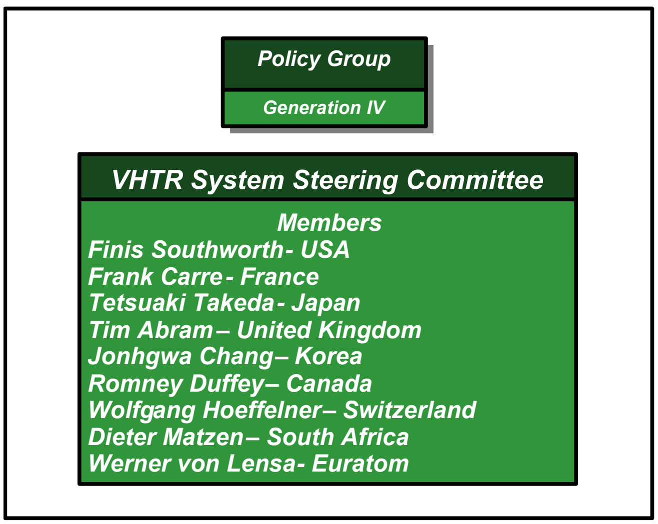


Figure 1-5. The VHTR System Steering Committee reports to the Generation IV International Forum Policy Group.

Materials and Components, Fuel and Fuel Cycle, Hydrogen Production, and Design and Safety Methods. The members of the four Project Management Boards are shown in Figure 1-6. Each project management board will develop multiple collaboration agreements within their area.

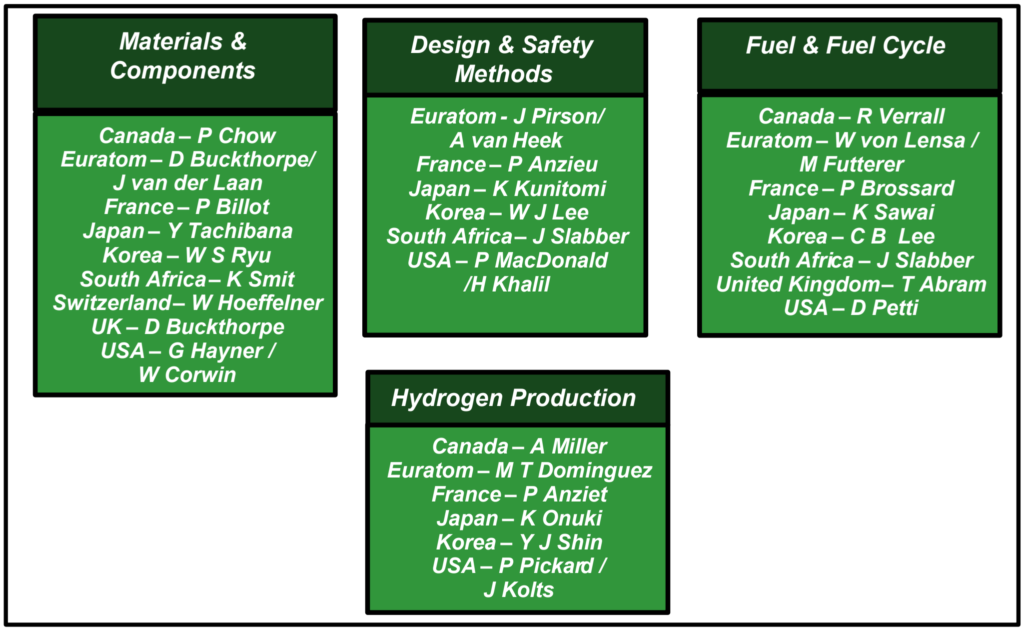


Figure 1-6. Active project management boards developing collaboration agreements for the VHTR.

The member countries of the Generation IV Forum are expected to sign a framework agreement (government to government) in the next few months that will establish the legal agreements allowing productive, yet protected, sharing of research and development. After the framework is in place, the system level agreement can be enacted for the nine member countries that have joined the VHTR Steering Committee. Lastly, collaboration agreements will be developed under each project board that will detail the shared research and budget and schedule commitments. This mechanism is expected to result in significant benefit to the NGNP, which is the first announced demonstration of the VHTR. Thus, it is expected that up to one half of the NGNP-related R&D may be contributed by the international participants on the VHTR Steering Committee.

1.5 Estimated Overall Project Costs and Schedules

Table 1-1 below shows the NGNP required budget over the next twelve years to support NGNP initial operation in 2017. The numbers for FY-03 and FY-04 were actual costs. The FY-05 numbers include carryover from FY-04 and also $1 million from AFCI to support the fuel program. The remaining budget figures represented below are initial estimates which may change when the preconceptual designs are completed such that the R&D can be focused on a specific core design, and engineering estimates can be made. No contingencies have been added to the estimates below.

Table 1-1. NGNP Annual Budget Profile

NGNP Budget Profile (2017)

Activity	FY-03	FY-04	FY-05	FY-06	FY-07	FY-08	FY-09	FY-10	FY-11	FY-12	FY-13	FY-14	FY-15	FY-16	FY-17	Total
Research and Development																
Fuel Development and Qualification	6,000	6,287	13,964	13,000	20,000	24,000	25,000	25,000	25,000	16,000	12,000	8,400	10,300	9,500	8,500	**222,951**
Materials Testing and Qualification	480	2,573	6,830	9,200	23,500	38,500	39,500	45,600	36,800	5,200	3,450	2,000				**213,633**
Design Methods and Validation	1,200	898	2,671	2,500	6,000	6,000	6,000	6,900	5,700	5,400	50					**43,319**
Design and Trade Studies			7,300	5,000	24,000	40,000	45,000	33,000								**154,300**
Public Outreach			200	300	500	500	500	500	500	500	500	500	500	500	500	**6,000**
NGNP Subtotal	**7,680**	**9,758**	**30,965**	**30,000**	**74,000**	**109,000**	**116,000**	**111,000**	**68,000**	**27,100**	**16,000**	**10,900**	**10,800**	**10,000**	**9,000**	**640,203**
Hydrogen Development	3,000	4,000	9,000	18,000	40,000	40,000	40,000	40,000	42,000	15,900	7,000	4,100	4,200	5,000	6,000	**278,200**
DOE SHARE SUBTOTAL	**10,680**	**13,758**	**39,965**	**48,000**	**114,000**	**149,000**	**156,000**	**151,000**	**110,000**	**43,000**	**23,000**	**15,000**	**15,000**	**15,000**	**15,000**	**918,403**
INDUSTRY SHARE					35,000	40,000	45,000	50,000	90,000	160,000	140,000	85,000	85,000	85,000	85,000	**900,000**
Total	**10,680**	**13,758**	**39,965**	**48,000**	**149,000**	**189,000**	**201,000**	**201,000**	**200,000**	**203,000**	**163,000**	**100,000**	**100,000**	**100,000**	**100,000**	**1,818,403**

Figure 1-7 provides a graphic profile of Research and Development funding needs against the assumed project time line. Note that the major portion of R&D work is completed before completion of Final Design.

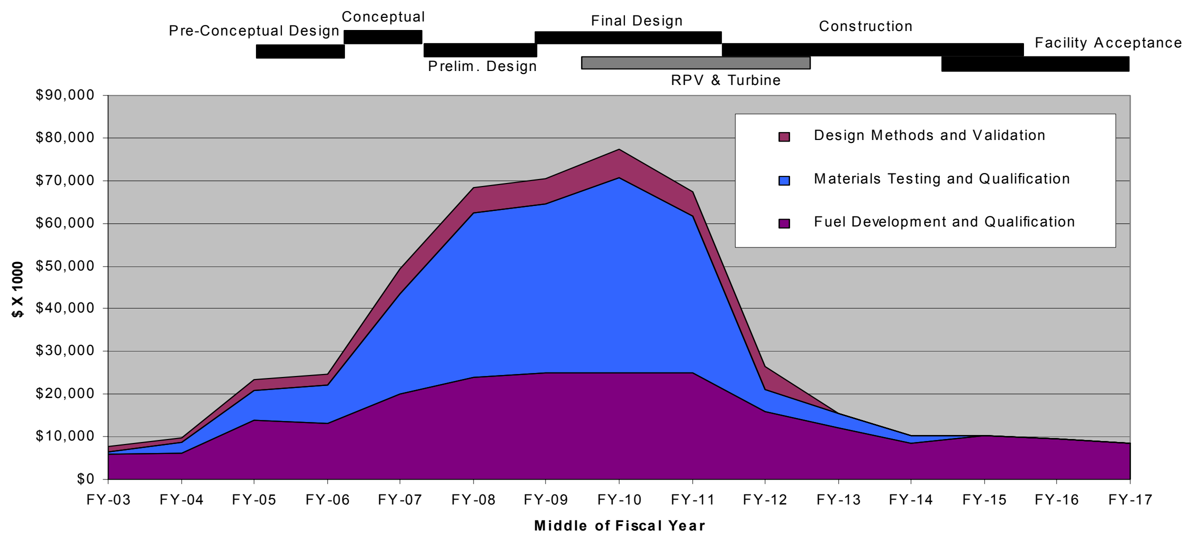


Figure 1-7. NGNP Research and Development Funding Profile.

The NGNP Summary Level Schedule is shown in Figure 1-8. The schedule shows the major activities under each of the major research and development areas at the top of the schedule, and the assumed design and construction activities are shown in the lower half. The schedule has been prepared from the latest R&D program plans for Fuel Development, Materials Selection and Qualification, and Design Methods and Safety. Energy Conversion is still in planning, and a placeholder has been inserted. The following assumptions were made in preparation of the schedule:

- The Design and Construction schedule will follow the principles of DOE O 413.3-1.
- Items shown under the Design and Construction (including the Environmental Impact Statement and Safety Analysis) will be performed by or controlled by the industrial partner as determined in the Acquisition Strategy.
- The industry partner will be the NRC licensee.
- The Reactor Pressure Vessel will be needed about 12 to 18 months into construction, and a 38- to 40-month procurement schedule is needed.

ID	Task Name	Start	Finish
1	**Research and Development**	**Oct 1 '04**	**Dec 29 '17**
2	**Fuel Development**	**Oct 1 '04**	**Dec 29 '17**
3	**Fuel Supply Fabrication Development**	**Oct 1 '04**	**Oct 1 '12**
4	Fuel Fabrication Small Coater for AGR 1	Oct 1 '04	Sep 1 '06
5	Coater Scaleup and Fuel Fabriation for AGR 2,3,4	Jun 6 '06	Sep 3 '08
6	Fuel Fabrication AGR5-8	Jan 2 '12	Oct 1 '12
7	**Fuels and Materials Irradiations**	**Jan 12 '07**	**Oct 14 '16**
8	AGR 1 Irradiation - Shakedown Small Coater Fuel	Jan 12 '07	Jul 14 '09
9	AGR 2 Irradiation - Large Coater fuel Performance	Jan 30 '09	Aug 2 '11
10	AGR 3 Irradiation - Fission Product Transport Data	Aug 3 '09	Feb 1 '12
11	AGR 4 Irradiation - Fission Product Transport Data	Jul 30 '10	Jan 29 '13
12	AGR 5 & 6 Irradiation - Fuel Qualification	Oct 1 '13	Mar 31 '16
13	AGR 7 & 8 Irradiation - Fuel Performance & Fission Prod	Apr 16 '14	Oct 14 '16
14	Post Irradiation Examination and Safety Testing	Jun 24 '09	Mar 24 '17
15	Fuel Performance Modeling - Code Development	Oct 1 '04	Jul 3 '15
16	V&V of Fuel Performance Code	Jul 1 '15	Dec 29 '17
17	Fission Product Transport and Source Term	Oct 1 '04	Jul 3 '14
18	Fission Product Transport & Source Term Code Va	Jul 1 '09	Mar 31 '17
19	Preliminary NRC Topical on fuel Performance	Dec 31 '14	Dec 31 '14
20	Final NRC Topical on Fuel Performance	Jul 1 '16	Jul 1 '16
21	NGNP source Term Validated	Mar 31 '17	Mar 31 '17
22	Fuel Performance Model Validated	Dec 29 '17	Dec 29 '17
23	First Core Fuel Fabrication	Oct 2 '14	Oct 5 '16
24	Complete the NGNP Fuel Qualification Program	Dec 29 '17	Dec 29 '17
25	**Materials Selection & Qualification**	**Oct 1 '04**	**Feb 6 '15**
26	**Graphite**	**Oct 1 '04**	**Dec 17 '10**
27	Select graphite	Oct 1 '04	Feb 4 '05
28	Procure graphite	Feb 7 '05	May 11 '05
29	Design ATR test	Oct 1 '04	Oct 14 '05
30	Irradiation Creep Test in ATR	Oct 17 '05	Nov 21 '08
31	Model Development	Feb 7 '05	Mar 2 '07
32	Graphite Design Methodology	Mar 5 '07	Mar 27 '09
33	Post Irradiation Evaluation (HFIR)	Oct 1 '04	Oct 27 '05
34	High Temp Irradiation Tests	Nov 13 '06	Dec 17 '10
35	GIF Irradiation Review	Oct 4 '05	Dec 31 '07
36	**Hi Temp Design Methods**	**Oct 1 '04**	**Jan 13 '15**
37	Data Assembly	Oct 1 '04	Oct 14 '05
38	Testing of Alloy 617	Oct 1 '04	Jan 11 '13
39	Testing of Alloy 617 Welds	Oct 1 '04	Dec 12 '12
40	Develop Methods	Oct 4 '05	Nov 10 '08
41	Develop Simplified Methods	Oct 4 '05	Nov 10 '08
42	Testing other Alloys	Oct 4 '06	Jan 13 '15
43	Testing other Alloys Welds	Oct 4 '06	Dec 12 '14
44	**Code Case Support**	**Oct 1 '04**	**Dec 30 '11**
45	ASME High Temp Metals	Oct 1 '04	Dec 30 '11
46	ASME Graphite	Oct 1 '04	Dec 30 '11
47	Air Oxidation	Oct 4 '05	Oct 29 '07
48	BET Surface Method	Oct 4 '05	Oct 29 '07
49	ASTM SiC/SiC	Oct 1 '04	Dec 30 '11
50	ASTM Fracture Toughness	Oct 1 '04	Oct 27 '06
51	**Environmental Testing and Thermal Aging**	**Oct 1 '04**	**Dec 30 '11**
52	Evaluate existing data	Oct 1 '04	Jul 13 '05
53	Refurbish creep-fatigue equipment	Oct 1 '04	Jul 13 '05
54	High Velocity Helium Testing	Oct 1 '04	Nov 9 '07
55	Low Velocity Helium Testing	Oct 1 '04	Nov 9 '07
56	High Temp Long Term Aging Tests	Oct 1 '04	Dec 30 '11
57	Long Term Creep Testing in Helium	Oct 1 '04	Dec 30 '11
58	**Reactor Pressure Vessel Material Testing**	**Oct 4 '05**	**Dec 31 '12**
59	Select Facility	Oct 4 '05	Oct 10 '06
60	Fabricate Testing Hardware	Oct 4 '05	Oct 29 '07
61	Irradiation of RPV materials	Oct 30 '07	Dec 31 '12
62	**Composites Development**	**Oct 1 '04**	**Dec 17 '10**
63	Helium Environment on SiC/SiC	Oct 1 '04	Oct 14 '05
64	Tube Geometry Tests	Oct 1 '04	Oct 14 '05
65	SiC/SiC Control Rod Tests	Oct 17 '05	Dec 17 '10
66	C/C Control Rod Testing	Oct 1 '04	Nov 9 '07
67	C/C Other Structures Testing	Nov 12 '07	Dec 17 '10
68	Database Handbook	Oct 4 '05	Dec 17 '10
69	RPV Transport and Fabrication	Oct 1 '04	Dec 17 '10
70	RPV Emissivity	Oct 4 '05	Oct 29 '07
71	Metallic Core Internals	Oct 4 '05	Oct 29 '07
72	Hot Duct & Insulation	Oct 4 '07	Dec 5 '12
73	IHXA & Piping	Oct 6 '08	Dec 19 '14
74	IHX Pressure Vessel	Oct 6 '08	Dec 19 '14
75	Administration	Oct 1 '04	Feb 6 '15
76	**Design Methods and Validation**	**Oct 1 '04**	**Sep 28 '12**
91	**Energy conversion (Turbine Development)**	Oct 3 '05	May 7 '10

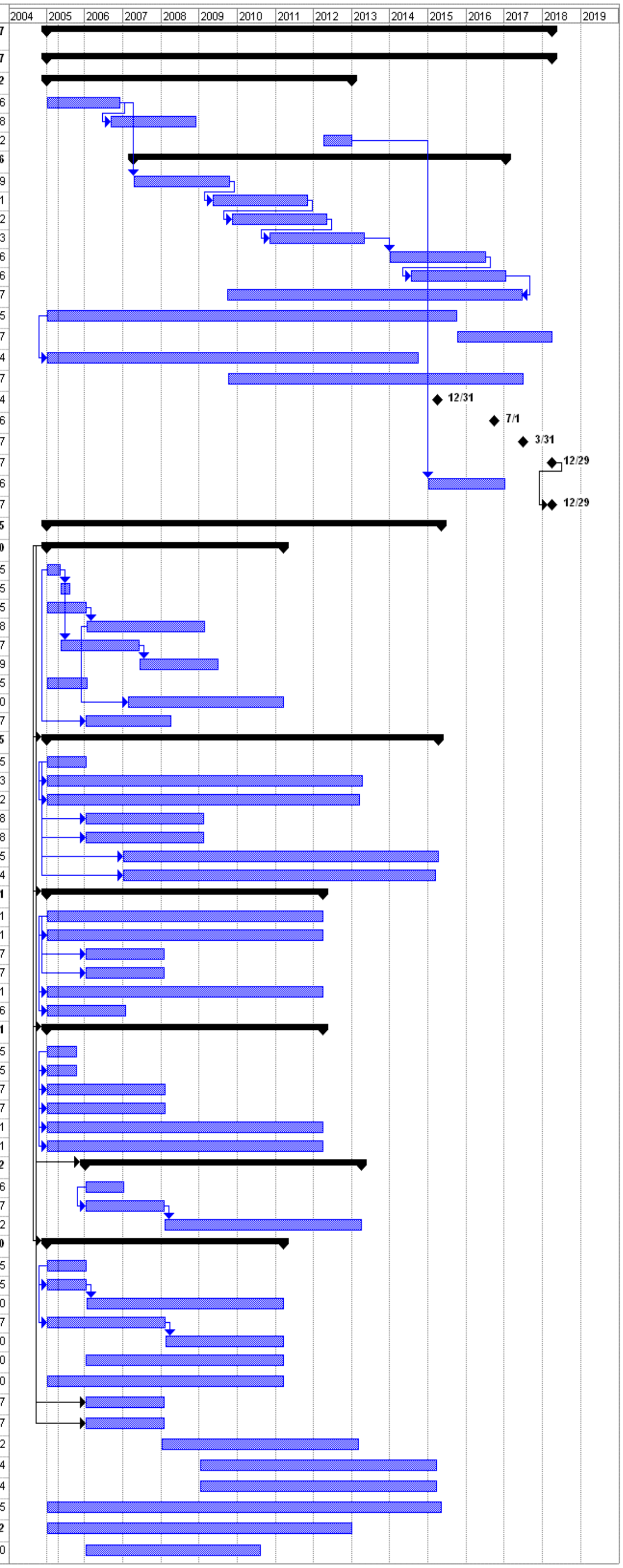


Figure 1-8. NGNP Summary Schedule.

Note that when the NGNP Acquisition Strategy is developed and approved, expect changes in order for the R&D planning to correlate with the industry partner's plans. At that time some issues will need to be resolved. Some of the major issues with the current planning are as follows:

- Fuel qualification irradiations ATR-5 and ATR-6 will be completed in 2015. Therefore the first core load of fuel must be fabricated in parallel with the final qualification irradiations and post-irradiation examinations.
- A commercial fuel manufacturing facility needs to be considered in the overall planning.
- The assumptions for the reactor pressure vessel procurement require that Materials R&D and vessel design be completed by the end of FY-09 in order to support the 2017 schedule above. The current R&D schedule will need to be compressed and funding will have to be accelerated if the vessel is not made of SA508.
- Design Methods and Safety R&D schedule and funding will also need to be accelerated to support preliminary design, the preliminary safety analysis, and the NRC construction permitting process.

The major milestones, as outlined in the Office of Nuclear Energy, Science and Technology Program Plan GPRA Unit 14: "Develop New Nuclear Generation Technologies," are as follows:

FY 2006: Complete the Pre-conceptual Design (CD-0)

FY 2007: Complete the Conceptual Design (CD-1)

FY 2009: Complete the Preliminary Design (CD-2)

FY 2012: Start Construction of the NGNP (CD-3)

FY 2013: Complete the NGNP Fuel Qualification Program

FY 2016: Obtain the NRC Operating License

FY 2017: Begin NGNP Operations (CD-4).

2. NGNP FUEL DEVELOPMENT AND QUALIFICATION

The fuel for the NGNP builds on the potential of the TRISO-coated particle fuel design demonstrated in high temperature gas-cooled reactors in the UK, U.S., Germany, and elsewhere. The TRISO-coated particle is a spherical-layered composite about 1 mm in diameter. It consists of a kernel of uranium dioxide (UO_2) or uranium oxycarbide (UCO) surrounded by a porous graphite buffer layer that absorbs radiation damage and allows space for fission gases produced during irradiation. Surrounding the buffer layer are a layer of dense pyrolytic carbon called the inner pyrolytic carbon (IPyC), a silicon carbide (SiC) layer, and a dense outer pyrolytic carbon layer (the OPyC). The pyrolytic carbon layers shrink under irradiation and create compressive forces that act to protect the SiC layer, which is the primary pressure boundary for the microsphere. The inner pyrolytic carbon layer also protects the kernel from corrosive gases present during deposition of the SiC layer. The SiC layer provides primary containment of fission products generated during irradiation and under accident conditions. Each microsphere acts as a mini pressure vessel, a feature intended to impart robustness to the gas reactor fuel and plant safety system.

The baseline fuel kernel for the NGNP is low-enriched (about 15% U-235 in the prismatic block reactor version of the NGNP and about 8% in the pebble bed version) UCO instead of UO_2, owing to performance issues associated with the UO_2 fuel at high power, temperature, and burnup. At the high power densities expected in the NGNP (>6 W/cm^3), the associated large thermal gradients can drive kernel migration in UO_2-coated particles. Migration of the kernel through the buffer and inner pyrocarbon layers and subsequent contact with the SiC layer generally results in extensive damage to the SiC layer. Furthermore, at the high burnups proposed for NGNP (15 to 20% FIMA), the CO pressure in a UO_2 fuel particle can be substantial, resulting in particle failure, especially under accident conditions. The high NGNP fuel temperatures (maximum time averaged temperature ~1250 °C) increase the effect of both of these mechanisms. UCO was selected because the mixture of carbide and oxide components precludes free oxygen from being released due to fission. As a result, no carbon monoxide is generated during irradiation, and little kernel migration (amoeba effect) is expected. Yet, like UO_2, the oxycarbide fuel still ties up the lanthanide fission products as immobile oxides in the kernel, which gives the fuel added stability under accident conditions.

For the pebble bed version of a NGNP, the coated particles are over-coated with a graphitic powder and binders. These over-coated particles are then mixed with additional graphitic powder and binders and then molded into a 50-mm-diameter sphere. An additional 5-mm fuel free zone layer is added to the sphere before isostatic pressing, machining, carbonization, and heat-treating.

For the prismatic version of the NGNP, a similar process is envisioned, where the over-coated particles are mixed with graphitic powder and binders to form a cylindrical compact about 50 mm long and 12.5 mm in diameter. After final heat treatment, these compacts are inserted into specified holes in the graphite blocks. Figure 2-1 shows a sketch of a

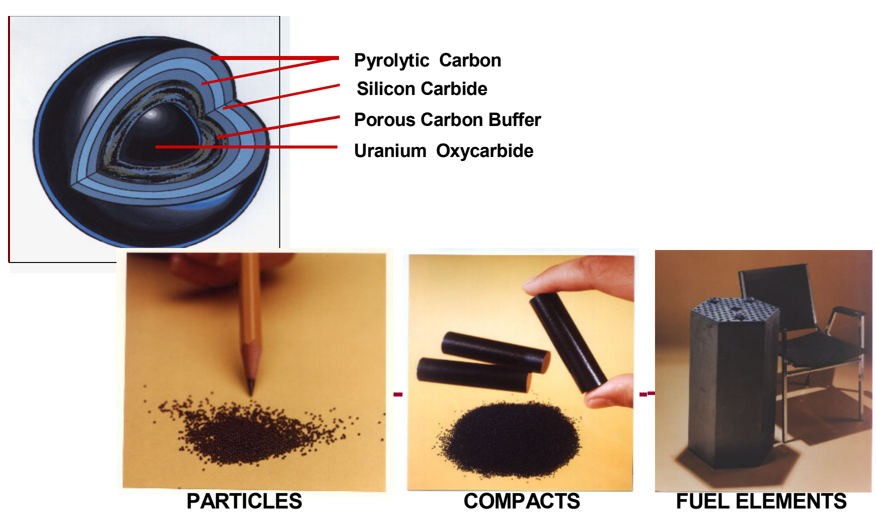


Figure 2-1. Cutaway of a TRISO-coated fuel particle and pictures of prismatic fueled high-temperature gas reactor fuel particles, compacts, and fuel elements.

TRISO-coated fuel particle and photographs of fuel particles, compacts, and fuel elements (prismatic blocks of graphite with fuel compacts and coolant channels) used in the high-temperature gas reactor at Fort St. Vrain. The Advanced Gas Reactor Fuel Development and Qualification (AGR) Program is currently focusing on the prismatic fuel form.

2.1 Advanced Gas Reactor (AGR) Program Justification and Need

A recent review [Petti 2003] concludes that there has historically been a difference in the quality of U.S. and German high temperature gas reactor fuel. This fact is illustrated in Figure 2-2, where the krypton release rate to birth rate (R/B) measurements from most of the U.S. and German TRISO coated fuel irradiation experiments are plotted versus fast fluence. The U.S. data from individual experiments are shown as lines whereas the yellow band in Figure 2-2 shows the range of the German data. This difference has been traced to technical differences in the fabrication processes used in Germany and the United States, as well as differences in the irradiation and testing programs in the two countries. Review of the fabrication processes used in Germany and the United States to make coated particle fuel indicates that the scale of fuel fabrication and development efforts in the last 25 years have been quite different. German fabrication was at an industrial/production scale supporting the German AVR and THTR reactors and providing an established infrastructure for additional production of high quality fuel in support of HTR-Modul development. Only about 100 defects were measured in the German high quality fuel among 3.3 million particles produced in support of HTR-Modul development. The post-Fort St. Vrain U.S. program has been a mixture of laboratory- and larger-scale fabrication. The initial defect levels varied greatly and were much greater than those produced in Germany. Also, the U.S. program was scattered and disjointed, and multiple variables were "attacked" in each irradiation experiment, leading to a situation where it was not always possible to isolate the cause for poor results.

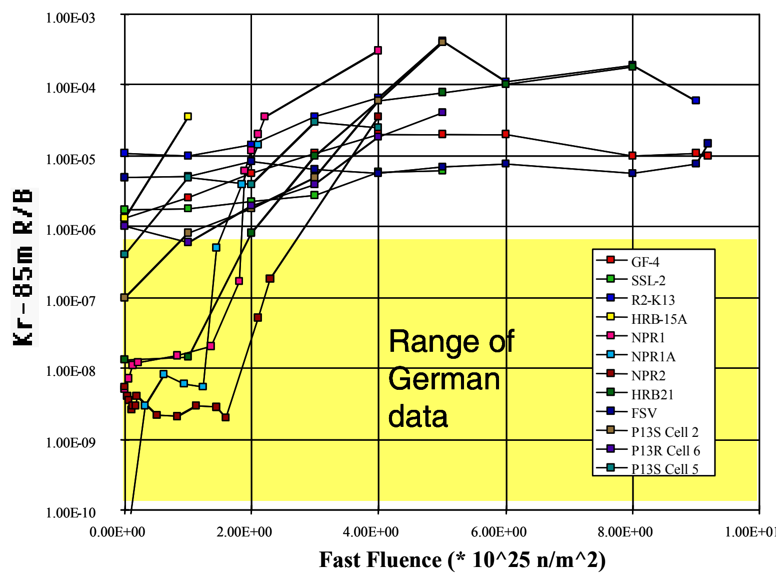


Figure 2-2. Krypton release to birth ratios versus fast fluence from a variety of U.S. and German fuel irradiation experiments showing the better performance of the German fuel.

Comparison of the U.S. and German fabrication processes has revealed many differences. Three specific technical differences in the TRISO fuel coating layers produced by the respective fabrication processes have important impacts in terms of performance under irradiation and accident conditions: pyrocarbon anisotropy and density, IPyC/SiC interface structure, and SiC microstructure.

2.1.1 Pyrocarbon Coating Rate

The density and anisotropy of the pyrocarbon layers of the TRISO fuel particle is determined by the conditions in the coater [Martin 2000]. The German pyrocarbon was deposited at a higher coating gas concentration, which in turn results in a higher coating rate (~4-6 μm/minute) than generally used in the U.S. The German pyrocarbon was very isotropic and thus survived irradiation quite well. However, the German fabrication conditions appear to lead to somewhat greater surface porosity than in U.S.

pyrocarbon, possibly leading to increased permeability to chlorine gas during the SiC coating process, and reaction with the UCO kernel. U.S. pyrocarbon has been coated under a variety of conditions. In many cases, it was coated at very low coating gas concentrations, which results in a lower coating rate (2-4 μm/minute), and leads not only to a very dense and impermeable IPyC layer, which is important to preventing attack of the kernel by chlorine during deposition of the SiC layer, but also to excessive anisotropy, which can cause cracking of the pyrocarbon under irradiation.

A plot of the irradiation-induced strain as a function of coating rate is shown in Figure 2-3. This plot indicates that strains induced in irradiated pyrocarbon are much greater for pyrocarbon coated at very low coating rates. Post-irradiation examination of many of the U.S. capsules indicate shrinkage cracks in the inner pyrocarbon layer, which has been shown [Baldwin 1993; Miller 2001; Maki 2002] to lead to stress concentrations in the SiC layer and subsequent failure of the SiC layer. Furthermore, anisotropy measurements on pyrocarbon have not adequately correlated processing parameters to pyrocarbon isotropy, and have not yet proven to be a reliable predictor of in-reactor pyrocarbon failure. More reliable methods of anisotropy characterization are needed to ensure a link between acceptable coating processing parameters and satisfactory pyrocarbon in-reactor behavior.

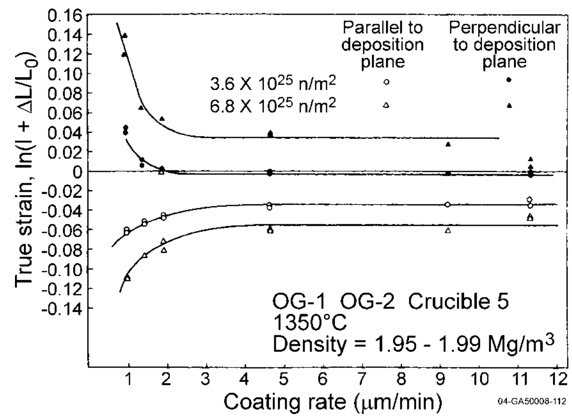


Figure 2-3. Irradiation-induced strains in PyC as a function of PyC coating rate.

2.1.2 Nature of the IPyC/SiC Interface

Differences in the microstructure and surface porosity between the German and U.S. IPyC also led to differences in the nature of the bond that existed between the layers. Photomicrographs of the IPyC/SiC interface in German and U.S. fuel are shown in Figure 2-4. The figure shows that the interface in German fuel is more tightly bonded because the SiC was deposited into the IPyC, which has apparently greater surface porosity. The U.S. fuel's denser, less porous IPyC surface resulted in a smoother, lower strength bond. The TRISO coating on the German fuel never exhibited debonding under irradiation, whereas irradiation results indicated that the TRISO coating on the U.S. fuel debonded frequently. The debonding is believed to be related to the strength of the IPyC/SiC interface and can lead to stress intensification in the SiC layer, which may cause failure.

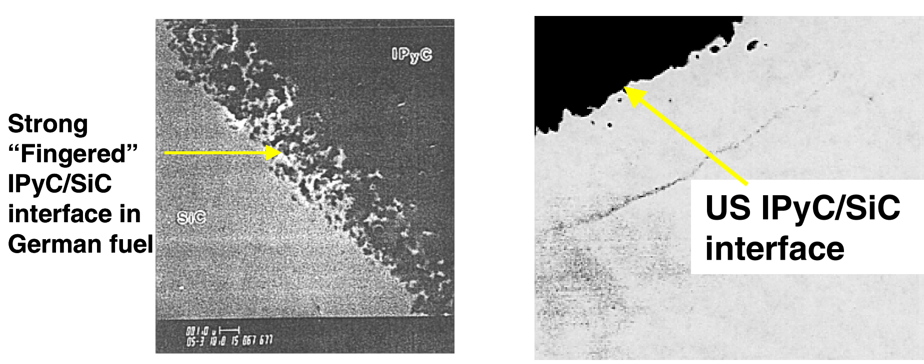
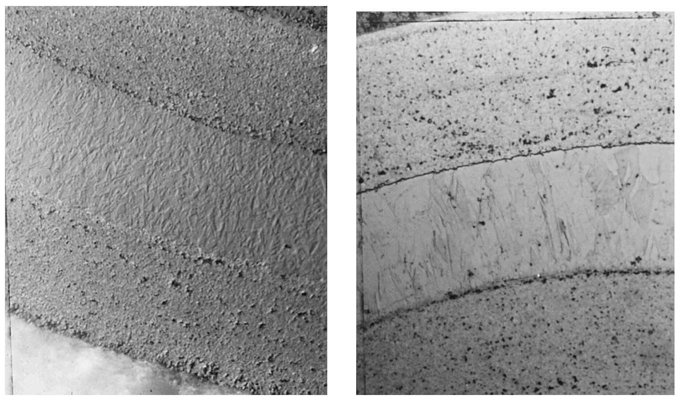


Figure 2-4. Comparison of SiC/IPyC interface in German (left) and U.S. fuel.

2.1.3 SiC Microstructure

The microstructures of German and U.S. SiC were different, as illustrated in Figure 2-5. The German process resulted in small equiaxed grains, whereas the U.S. process produced larger columnar (sometimes thru-wall) grained SiC. This difference in microstructure is believed to be primarily a function of the temperature used during the SiC coating phase in the coaters, with the U.S. coater producing SiC at a higher temperature in some or all regions of the coater compared to the German process. These differences are important from a performance perspective because the smaller-grained German SiC, with its higher tortuosity, should in principle retain metallic fission products better than the large thru-wall columnar U.S. SiC with more direct grain boundary pathways through the layer.



German United States

Figure 2-5. Comparison of the microstructure of German and U.S.-produced SiC.

2.1.4 Irradiation Testing

Review of the U.S. and German irradiation programs over the last 25 years indicates that the irradiation programs were implemented differently, with vastly different results. The focus of the German program was on UO_2-TRISO fuel for the pebble bed reactors AVR and THTR and all future pebble bed reactor designs, such as the HTR Modul design. The U.S. program examined many different variants (different coatings, different kernels) with apparently few lessons learned from one irradiation to the next or feedback to the fabrication process. Furthermore, the U.S. did not always conduct post-irradiation examinations; photomicrographs were limited from those examinations that were conducted; and characterization of the layer failures was sometimes sporadic. Even more striking is that the on-line gas release measurements indicated that the German fuel exhibits about a factor of 1000 less fission gas release under irradiation than the U.S. fuel under a broad range of irradiation conditions (temperature, burnup, fluence; see Figure 2-6). Furthermore, the post-irradiation examination of the U.S. fuel confirmed the more extensive gas release data.

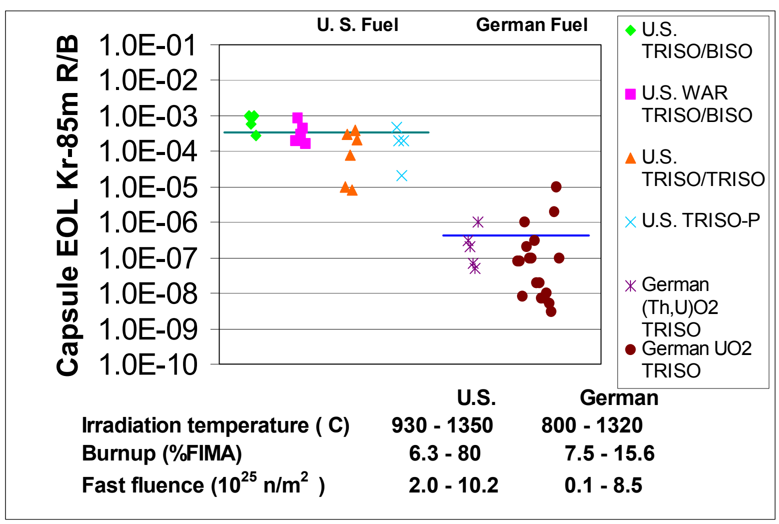


Figure 2-6. Comparison of end-of-life Kr-85m R/B from historic German and U.S. irradiations.

In summary, the German fuel was excellent. Of about 340,000 particles tested, there were no in-pile failures and only a few “damaged” particles from experimental anomalies. The fission gas release that did occur was attributed only to as-manufactured defects and heavy metal contamination. The U.S. fuel did not perform very well. There were relatively high numbers of failures of individual layers of the TRISO coated U.S. fuel and, in many cases, a significant fraction (~1 to 10%) of the total particles completely failed, (see Figure 2-7; note that individual layer failure fractions are plotted, not TRISO-coated particle failure fractions). A variety of failure mechanisms were noted relating to effects of accelerated irradiation and attributes of the fabrication process.

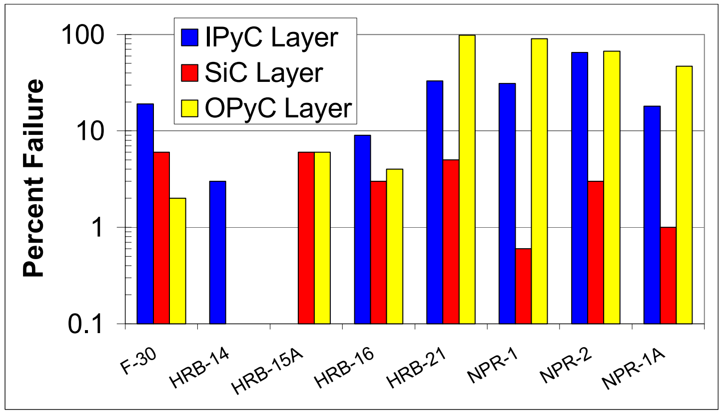


Figure 2-7. Individual layer failures observed during post-irradiation examination of U.S.-coated particle fuel over the past 25 years.

This comparison strongly supports the need for process improvement studies for fuel manufactured using traditional U.S. methods and potential scoping irradiations to demonstrate the effectiveness of any changes in the process.

2.2 Advanced Gas Reactor Program Structure

Development and qualification of TRISO-coated low-enriched uranium fuel is a key R&D activity associated with the NGNP Program. The work is being conducted in accordance with the *Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program* [Bell et al. 2003]. The AGR Program includes work on improving the kernel fabrication, coating, and compacting technologies, irradiation and accident testing of fuel specimens, and fuel performance and fission product transport modeling. The primary goal of these activities is to successfully demonstrate that TRISO-coated fuel can be fabricated to withstand the high temperatures, burnup, and power density requirements of a prismatic block type NGNP with an acceptable failure fraction. It is assumed that TRISO fuel that is successful in a block reactor will also be successful in a pebble-bed reactor, since the particle packing fraction and the fuel temperatures are somewhat lower in pebble-bed reactors than in block reactors. In addition, commercialization of the fuel fabrication process, to achieve a cost-competitive fuel manufacturing capability that will reduce entry-level risks, is a secondary goal of the project.

The project is co-managed by INL and Oak Ridge National Laboratory (ORNL) against a resource-loaded critical path schedule with three levels of key milestones. This schedule clearly defines the activities and deliverables required and determined feasible through early schedule and cost analysis.

Implementation of the quality assurance requirements delineated in the Technical Program Plan will be in accordance with DOE quality assurance requirements specified in 10 CFR 830, “Nuclear Safety Management,” Subpart A, “Quality Assurance Requirements,” and in DOE Order 414.1B, “Quality Assurance.” In addition, all activities that have direct input to irradiation test specimen fabrication and irradiation campaigns will be conducted in accordance with the national consensus standard ASME NQA-1-2000, “Quality Assurance Requirements for Nuclear Facility Applications,” published by the American Society of Mechanical Engineers (ASME).

2.2.1 Fuel Manufacture

This program element addresses the work necessary to produce coated-particle fuel that meets fuel performance specifications and includes process development for kernels, coatings, and compacting; material characterization and quality control methods development; scale-up analyses; and process documentation needed for technology transfer. The effort will produce fuel and material samples for characterization, irradiation, and accident testing as necessary to meet the overall goals. There will also eventually be work to develop automated fuel fabrication technology suitable for mass production of coated-particle fuel at an acceptable cost; that work will be conducted during the later stages of the program in conjunction with cosponsoring industrial partners. Fuel manufacture development is guided by a detailed fuel product specification established based on historical US and international experience.

Near-term activities focus on production of UCO kernels, coating of particles in a continuous process using a small (2-inch) laboratory-scale coater, production of fuel compacts, and characterization of the resulting materials. The goal of the ***kernel studies*** is to better define the operating window that will produce kernels meeting all specifications. For example, studies in early 2005 demonstrated carbon dispersion parameters that would result in adequate sintered kernel density. Following fabrication of the AGR-1 kernels, additional kernel development studies are needed to further define the operating envelop for both broth and sintering parameters relative to the fuel specification and other properties such as kernel strength and friability, and surface reactivity.

The goal of the initial ***coating studies*** is to produce coatings like those produced by the German program in the late 1980s. All three layers were coated in a continuous manner in the German process, whereas in the U.S. process, the fuel particles were unloaded after each coating layer to perform quality measurements. Additional coating variants are planned that will confirm understanding of the historical coating fabrication database and enhance the prospects for one or more successful outcomes, and the baseline and selected variants will then be irradiated in the first irradiation test, AGR-1. Recommended coating rates and temperatures for the coating variant candidates planned for the AGR-1 fuel fabrication campaign are listed in Table 2-1 (these conditions may be adjusted based on understandings gained from early fuel production and characterization).

Table 2-1. Candidate coating variants for AGR-1.

Variant	IPyC Conditions	SiC Conditions	Comment
1	1300 ºC; 4.5 μm/min	1510 ºC; 0.2-0.25; μm/min	German baseline
2	1300 ºC; 4.5 μm/min	1580 ºC; 0.2-0.25 μm/min	Higher SiC deposition temperature
3	1300 ºC; 3.0 μm/min	1510 ºC; 0.2-0.25 μm/min	Low IPyC coating rate (anisotropic)
4	1300 ºC; 3.0 μm/min	1580 ºC; 0.2-0.25 μm/min	Low IPyC coating rate (anisotropic)
5	1300 ºC; 6 μm/min	1510 ºC; 0.2-0.25 μm/min	High IPyC coating rate
6	1300 ºC; 6 μm/min	1580 ºC; 0.2-0.25 μm/min	Higher SiC deposition temperature
7	1300 ºC; 4.5 μm/min	1510 ºC; 0.2-0.25 μm/min	Interrupted variant of Case 1
8	1300 ºC; 4.5 μm/min	~ 1300 ºC with Argon	

Coating conditions are planned that span the range from producing highly anisotropic/high density PyC to highly isotropic/low density PyC. Two different SiC coating temperatures are planned to determine an acceptable window for producing the desired fine-grained SiC. An interrupted run is also planned to more quantitatively characterize fuel produced in both interrupted and uninterrupted modes. In addition, a variant in which argon gas is used during SiC coating is planned, since the UK Dragon project and current microelectronics production has demonstrated that good SiC can be produced at much lower temperatures when this gas is used.

The second phase of coating development involves scaleup of the continuous coating process to production size (e.g., 6-inch) coaters. The goal is to produce high quality coatings for performance demonstration and, ultimately, qualification.

The laboratory scale coating development work includes the development of a comprehensive coating process model to support small coater process development and the transition from laboratory scale coaters to production scale coaters. A major challenge is to account for the effects of the turbulent gas-solids interactions in the fluidized bed reactor on the rate controlling processes and the final product quality of the chemical vapor deposition. The modeling team will make use of the latest computational fluid dynamics computer codes and correlations available for simulating the hydrodynamics, heat and mass transfer, and chemical reaction kinetics on the particle surfaces. In addition, experimental validation will be needed at each stage of development to ensure that the model predictions are consistent with the actual physics and chemistry. The latter is critical to the implementation of successful scale-up from the laboratory to production prototype.

Because of the complexity of the various interacting physical phenomena, coating process model development will follow a series of stages in which the major elements of the CVD process are addressed one at a time. The basic order of the model component development is expected to be as follows:

1. Hydrodynamics
2. Heat and mass transfer
3. Reaction/deposition kinetics
4. Particle evolution.

Coated particles will then be over-coated and molded into cylindrical compacts using a matrix of graphite flour and carbonized resin. The thermosetting resin based matrix and warm pressing compacting process selected for the program is similar to processes used in Germany and Japan, and a substantial departure from the thermoplastic matrix injection process used previously in the US. Development work is required to adapt the process to the U.S. fuel compact specifications. Although the matrix is similar to the German matrix, the ratio of matrix to particles is quite different, approximately 72:28 versus 90:10 for the Germans. Also, the German pebbles were isostatically pressed into spheres while the AGR compacts will be compression molded (via the warm pressing step) into cylindrical compacts. Being that the fuel particles are non-compressible, this reduction in amount of matrix and change in molding technique requires a consistent particle overcoat thickness and careful pressing of the particles into compacts. A primary objective of the compacting development is to limit particle damage within the very low levels required by the fuel product specifications, with allowance for a low level of defects in the coated particles used to form the compacts.

Parameters needed to establish a uniform overcoat have been optimized using surrogates, and compacts have been warm pressed and carbonized (see Figure 2-8). Future plans include optimizing the final heat treatment and compacting uranium bearing coated particles.

Figure 2-8. Compacts produced using ORNL thermosetting resin process.

In parallel with the fuel fabrication, additional effort is being expended in the area of fuel characterization, with the goal of providing feedback to fabrication process development, demonstrating compliance with product specifications, and establishing more advanced and more robust techniques to measure key attributes of the fuel that can be integrated into a continuous production-scale coating process. Initial activities focus on reestablishing conventional characterization procedures and developing improved anisotropy and optical image measurement and analysis techniques. Advanced tomography techniques to measure layer thickness and densities are also planned and ORNL is acquiring a high-resolution (1-2 μm) x-ray inspection system to support this effort. Figure 2-9 shows results from the computer automated optical characterization equipment developed at ORNL.

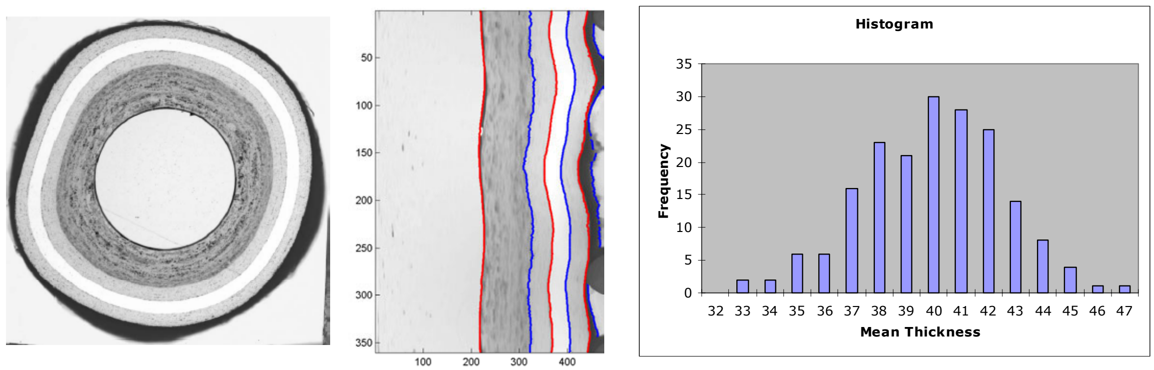


Figure 2-9. Example information from the ORNL computer automated optical characterization system. An IPyC histogram is shown on the right.

Computer controlled sample positioning and digital imaging plus ORNL-developed image analysis software is used to quickly and easily analyze 1000’s of particles for size and shape with a 2-μm resolution. The system is also capable of quickly and easily analyzing 100’s of particle cross-sections with 1 μm resolution and providing copious data from which particle dimensions, layer thickness and particle shape can be obtained. ORNL has also developed an advanced optical method to measure pyrocarbon anisotropy. The degree and direction of pyrocarbon crystallite orientation is measured by a scanning ellipsometry technique called the 2-MGEM (2-modulator generalized ellipsometry microscope). Figure 2-10 shows typical results from that equipment. Recent data indicates a 2 μm spot size has been achieved providing new information on the variation in pyrocarbon properties within a layer for both archived US and German fuel as well as material produced by the Program.

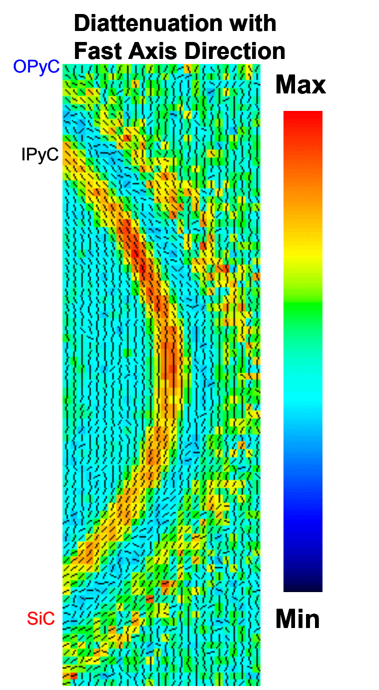


Figure 2-10. Typical results from the ORNL equipment for measuring pyrocarbon anisotropy.

2.2.2 Fuels and Materials Irradiation

The fuel and materials irradiation activities will produce data on fuel performance under irradiation as necessary to support fuel process development, to qualify fuel for normal operating conditions, and to support development and validation of fuel performance and fission product transport models and codes. The irradiations will also produce irradiated fuel and materials as necessary for post-irradiation examination and ex-core high-temperature furnace safety testing.

A total of eight irradiation capsules will be used to obtain the necessary data and sample materials. Details on each irradiation are listed in Table 2-2. The purpose of AGR-1 is to shakedown the new multi-cell capsule design, fabrication, and operation to reduce the chances of capsule failures in subsequent irradiation tests. If successfully taken to a substantial fraction of design burnup and fast fluence, the test will yield key irradiation performance data from a number of early fuel variants produced under different processing conditions from laboratory-scale coating equipment, as discussed in Section 2.2.1 above. AGR-2 will be a performance demonstration irradiation with fuel fabricated from a production-scale coater. Feedback to the fabrication process is expected following both AGR-1 and AGR-2. AGR-3 is devoted to obtaining data on fission gases and fission metals under normal irradiation conditions. AGR-4 will study fission product behavior in fuel compact matrix and graphite materials.

Table 2-2. Planned AGR irradiation capsules.

Capsule	Task
AGR-1	Shakedown and early fuel
AGR-2	Performance test fuel
AGR-3	Fission product transport - 1
AGR-4	Fission product transport - 2
AGR-5	Fuel qualification - 1
AGR-6	Fuel qualification - 2
AGR-7	Fuel performance model validation
AGR-8	Fission product transport –3

Given the statistical nature of coated particle fuel, a large number of fuel specimens are needed to fully qualify[a] the fuel and demonstrate compliance with the fuel failure specification. AGR-5 and AGR-6 are identical irradiations that will be used to qualify the fuel for the NGNP. AGR-7 and AGR-8 are irradiations designed to provide data with which to verify and validate fuel performance and fission product transport models.

A schematic of the test train to be used for AGR-1 is shown in Figure 2-11. Each capsule will be a highly instrumented multi-cell capsule capable of irradiating six different fuel forms with different thermal conditions, if required. Flux wires will be used to measure the thermal and fast neutron fluences. Thermocouples in graphite bodies surrounding the fuel will be used to monitor temperatures during the irradiation. The graphite bodies may contain boron carbide to control power generation during the irradiation and prevent large power swings historically experienced when irradiating fuel to high burnup. A low flow of inert sweep gas is used during irradiation to provide the correct thermal conductance to allow the fuel to be irradiated at the proper temperature. Usually, most of the sweep/thermal control gas is helium. Small amounts of neon are used to change the overall conductance to compensate for depletion of

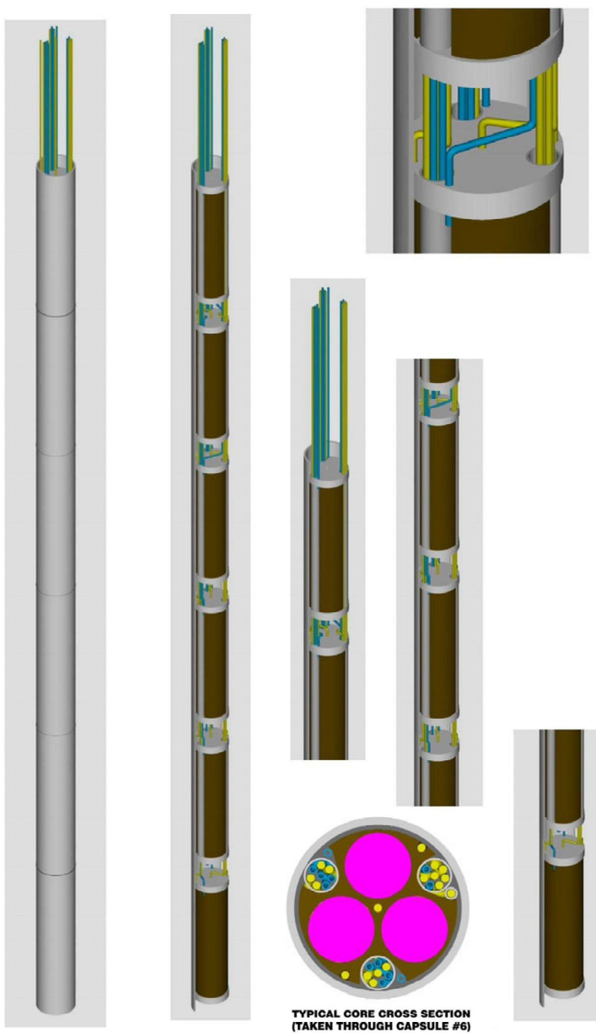


Figure 2-11. Schematic of AGR-1 multicell capsule.

[a] We are not doing 10CFR50 qualification. Page 5 of the AGR technical plan states: “Fuel qualification is herein defined as the demonstration of the robust performance and efficacy of the reference coated-particle fuel form through presentation of experimental data and analysis results. This fuel qualification effort is meant to support the NRC-RES in its preapplication review efforts for the VHTR concept and to support the NRC in its eventual issuance of the Gen-IV VHTR license.”

uranium due to burnup and still keep the fuel at the required temperature.

Planned AGR-1 irradiation conditions are a peak burnup of 18 to 20% FIMA, a volume average time average temperature of 1150 °C, a time average peak temperature of 1250 °C, and a fast neutron fluence of 5×10^{25} n/m^2 (E>0.18 MeV). The capsules will be irradiated in one of the large B positions at the Advanced Test Reactor at the Idaho National Engineering and Environmental Laboratory. The large B position has a neutron spectrum very similar to that expected in a gas reactor. Preliminary calculations suggest that each capsule will be irradiated for 2.5 years to meet the requirements stated above, which will simulate a three to four year irradiation in the NGNP.

An important objective of the irradiation is to measure the fission gas release from the fuel and correlate it to the operating parameters in the irradiation. The sweep gas from each cell containing fuel specimens will be "sniffed" for fission gas. The sweep gas also transports any fission gases released from the fuel to a location outside of the reactor, where an ion chamber with enough sensitivity to indicate a single fuel particle failure (evident by a spike in its signal) will measure gross radiation in the line. The isotopic content of the gas in the line will be monitored on line using the state-of-the-art fission product monitoring system shown in Figure 2-12. This system consists of a gamma spectrometer for continuous measurement of the concentration of the various fission gas isotopes in the sweep gas. With this instrumentation, particle failures can be monitored and correlated to conditions in the cell. The isotope concentration data will be used to calculate the R/B ratio for various fission products, a key measure of fission product retention and fuel performance.

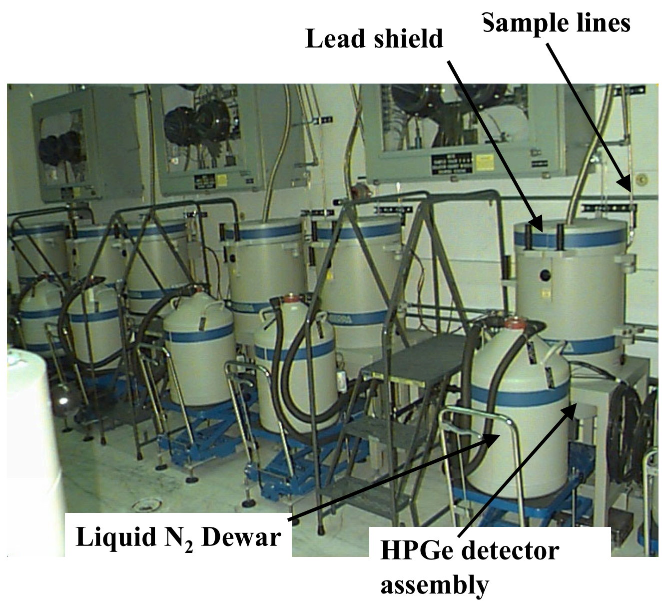


Figure 2-12. INL fission product monitoring system.

2.2.3 Post-irradiation Examination and Safety Testing

Data from the post-irradiation examination and safety testing will supplement the in-reactor measurements (primarily fission gas release-to-birth ratio measurements) as necessary to demonstrate compliance with the fuel performance requirements and support development and validation of the computational fuel performance models. This work will also support the fuel manufacture with feedback on the performance of kernels, coatings, and compacts.

2.2.3.1 Post-irradiation Examination

Post-irradiation examination is a collection of nondestructive and destructive techniques that can be used to characterize the state of the fuel either after irradiation or after safety testing. The different types of analyses or measurements that will likely be performed, the purpose of the measurements, and their value to the overall fuel qualification plan are discussed in the next few paragraphs.

Following removal of the irradiation test train from the reactor to the hot cell, a gamma scan of the entire test train will be performed. A collimated gamma spectrometer in the hot cell will traverse the capsule and record the gamma activity as a function of axial length. Such a measurement is generally qualitative and will provide information to determine whether any fuel compacts have broken or if a significant number of fission products have been released and moved within the capsule.

Following capsule disassembly and removal of the fuel element, the general condition of the fuel will be noted, specimens will be weighed, and dimensional measurements of the specimens will be performed to characterize the shrinkage or swelling that occurred during irradiation.

To examine the physical characteristics of irradiated fuel particle coatings, optical metallography will be performed on cross sections of the fuel pebble or fuel compact. These high magnification examinations offer excellent visual evidence of the condition of the fuel following testing. This technique will be used to investigate layer integrity, possible layer debonding, densification of layers (e.g., buffer) the degree of void formation due to fission gas, the extent of kernel migration and swelling, and the nature and extent of the fission product attack on the SiC. Use of bright field and polarized light and etching are useful techniques to reveal the microstructure of the SiC layer. With proper etching techniques, SiC grain orientation and sizes can be determined. Figure 2-13 is a photograph of optical metallography performed on German fuel following irradiation in the AVR.
Development of a nondestructive tomographic x-ray inspection technique is also under consideration.

Figure 2-13. Photomicrograph of German AVR fuel after irradiation.

Gamma-scanning of capsule components (e.g., graphite bodies) or leaching and gamma counting of capsule components will be used to determine the identity, migration, and distribution of fission products following irradiation.

To identify where the fission products are located within irradiated fuel particles, the fuel element will be deconsolidated to obtain individual particles for examination by electron microscopy to reduce the radiation background. The radiation background is the issue here, not damage to particles or the release of fission products. The reduced background radiation from a single fuel particle is usually required for good measurements by electron microprobe, where one is looking for x-rays characteristic of specific fission products (measured by energy dispersive or wave length diffraction techniques). This technique looks for evidence of fission product accumulation at the IPyC/SiC interface, fission product attack of the SiC, and fission products outside the fuel particles.

For irradiations of fuel compacts or pebbles, there will be a need to measure fuel particle failure fraction independently of the on-line R/B measurements, due to the uncertainty in the R/B measurement for a few particle failures and the inability to measure metallic releases. The most useful technique for fuel particle failure measurements, when the on-line R/B measurements suggest a failure fraction well under 1%, is leach-burn-leach. In this technique, the fuel compact or pebble is leached with acid to remove any fission metals (e.g., cesium) released from defective fuel particles and heavy metal contamination. (Note that on-line measurements during irradiation will only estimate fission gases.) The fuel element is then burned in air to remove all carbon matrix material, the OPyC layers, and also the IPyC/Buffer layers of any particles with failed SiC. Particles that remain are then leached with an acid solution to remove any exposed uranium that had been enclosed by an intact pyrocarbon layer. The measurement of the free uranium is then converted to a SiC defect fraction.

Another technique performed on coated particle fuel is the irradiated microsphere gamma analyzer (IMGA) developed at ORNL. With this technique, fuel particles following deconsolidation are analyzed individually by a gamma spectrometer and catalogued based on the ratio of mobile and immobile fission products measured in the particle. A histogram of such ratios is developed based on all the particles in a sphere or compact and compared to a normal distribution. Variations from normal can easily be seen with such a technique. Metallography following IMGA on the particles that depart from normal can be valuable to tie the microstructure of the anomalous particles to the fission product release. For high-quality fuel with low gas release, this technique may not be required, but for intermediate failure fractions of 10-4 to 10-2, deconsolidation followed by IMGA is useful.

Traditional burnup analysis is also performed as part of the series of post-irradiation examinations. Following deconsolidation, a few particles can be sent for destructive radiochemical assay to determine the concentration of transuranics and minor actinides, from which burnup can be assessed.

2.2.3.2 Safety Testing

An important goal of this program is to determine the performance of the fuel under high-temperature accident conditions, since integrity of the coated particle to high temperature is a crucial part of the safety case for the NGNP. In particular, three environments are of interest: helium, air, and steam. The irradiated TRISO fuel will be exposed to these environments for up to 500 hours. The exact composition of these environments are not known at present, but assumptions are that the test will be run at atmospheric pressure, and steam and air concentrations will be in the range of 10,000 ppm. Some of the early German data of this type is plotted in Figure 2-14, which shows krypton fractional releases as a function of heating time at 1600 °C and burnup. Note that the lower burnup fuel (8-10% FIMA) had little release, but the higher burnup fuel, typical of the burnup expected in NGNP, had much higher releases. Although these data are not directly applicable to the NGNP because of differences in fabrication and particle size, it is illustrative of the need to test fuel at a variety of burnup levels.

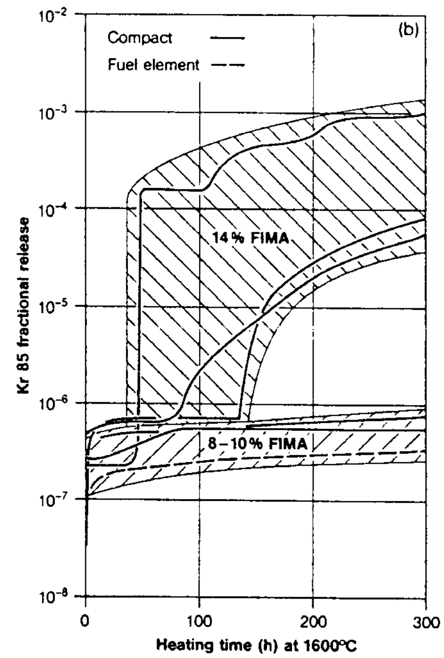


Figure 2-14. Krypton 85 fraction release data versus heating time at 1600 °C and fuel burnup from German heating tests.

The maximum temperature, including a 100 °C uncertainty, predicted for a core conduction cooldown accident in small modular gas cooled reactors is 1600 °C and is reached within ~50 to 100 hours after initiation of the event. Temperatures remain at ~1600 °C for about 25 to 50 hr, followed by a long, slow (hundreds of hours) cooldown. Traditionally, post-irradiation isothermal annealing at temperatures of 1600, 1700, and 1800 °C have been performed for several hundred hours, with continuous collection of released fission products.

Isothermal tests are generally considered to be conservative relative to heatup transient tests, which follow more closely the time-temperature profiles calculated to occur in a core conduction cooldown transient, because more time is spent at the highest temperatures. Thermal gradients are not expected to be significant. Isothermal tests are also easier to analyze than transient tests and, given the long thermal time constant associated with the transients, there is little new information to be gained by conducting transient tests. The experimental facility will consist of a furnace to maintain a fuel specimen at specified temperatures with a cold finger to trap the condensable fission products and a cold trap for trapping fission gases. The cold finger and cold traps are analyzed using traditional gamma spectroscopy. The data needed from safety testing are fission product release, TRISO coating layer integrity, and fission product distribution within fuel particles (corrosion likelihood) and fuel compacts.

The release behavior of the fission products is somewhat different than in other nuclear fuels. Silver (Ag-110m) is released first because of its greater mobility through the SiC coating on TRISO particle fuel. This is followed by Cs (134Cs and 137Cs), which can diffuse through the PyC and SiC layers after long times at these temperatures. Lastly, the fission gases (85Kr) are released.

Postheating test activities include characterization of the TRISO coating layer integrity by optical metallography including looking for evidence of SiC layer thinning and decomposition, chemical attack of the SiC, and the mechanical condition and microstructures of the SiC and PyC layers. Other procedures discussed earlier for irradiated fuel may also be applied. Detailed test matrices will be developed as the program evolves. Nondestructive x-ray tomography (if developed) will also be applicable.

2.2.4 Fuel Performance Modeling

The high temperature gas reactor TRISO coated fuel performance computer codes and models will be further developed and validated as necessary to support the fuel fabrication process development and the NGNP design and licensing activities. The fuel performance modeling will address the structural, thermal, and chemical processes that can lead to coated-particle failures. The models will address the release of fission products from the fuel particle and the effects of fission product chemical interactions with the coatings, which can lead to degradation of the coated-particle properties.

Compared to light water reactor and liquid metal reactor fuel forms, the behavior of coated-particle fuel is inherently more multidimensional. Moreover, modeling of fuel behavior is made more difficult because of statistical variations in fuel physical dimensions and component properties, from particle to particle and around the circumference of any given particle due to the nature of the chemical vapor deposition fabrication process. Previous attempts to model this fuel form have attacked different aspects of the problem. Simple one-dimensional models exist to describe the structural response of the fuel particle. Models or correlations exist to describe the fission product behavior in the fuel, though the database may not be complete owing to the changes in fuel design that have occurred over the last 25 years. Significant effort has gone into modeling the statistical nature of fuel particles. However, under pressure to perform over one million simulations with the computing power available in the 1970s and 1980s, the structural response of the particle was simplified to improve speed of calculation.

New models are currently being developed in the United States that represent a first-principles-based mechanistic, integrated, thermal-mechanical-physio-chemical-irradiation performance model for particle fuel, which has the proper dimensionality yet captures the statistical nature and loading of the fuel. The mechanistic model for coated-particle fuel considers both structural and physio-chemical behavior of a particle-coated fuel system during irradiation. The INL model, called PARFUME, includes the following important phenomena:

- Anisotropic response of the pyrolytic carbon layers to irradiation (shrinkage, swelling, and creep that are functions of temperature, fluence, and orientation/direction in the carbon).
- Failure of a SiC ceramic in the coating system (using the classic Weibull formulation for a brittle material), either by traditional pressure vessel failure or by mechanisms such as particle asphericity (see Figure 2-15), or pyrocarbon layer cracking (see Figure 2-16), or debonding and subsequent stress concentrations in the SiC layer.
- Chemical changes of the fuel kernel during irradiation (changes in carbon/oxygen, carbon/metal and/or oxygen/metal ratios, depending on the kernel fuel type, and production of CO/CO_2 gas) and its influence on fission product and/or kernel attack on the particle coatings.
- Thermo-mechanical response of the kernel and buffer as a result of buffer densification, kernel swelling, and gas generation (fission gases and CO), including development of gaps between the buffer and the TRISO-coating layers as a function of burnup, fast fluence, and temperature.
- Attack of the SiC layer by Pd and other fission products, and by kernel migration.
- Transport of key fission products (Kr, Ag, Sr, Cs) from the kernel and through each layer of the particle.

- Statistical variations of key properties of the particle associated with the production process, requiring Monte Carlo analysis of a very large number of particles to understand the aggregate behavior. Fabricated particles will exhibit statistical distributions for not only the physical dimensions of the individual coatings but also for the mechanical properties of these layers.

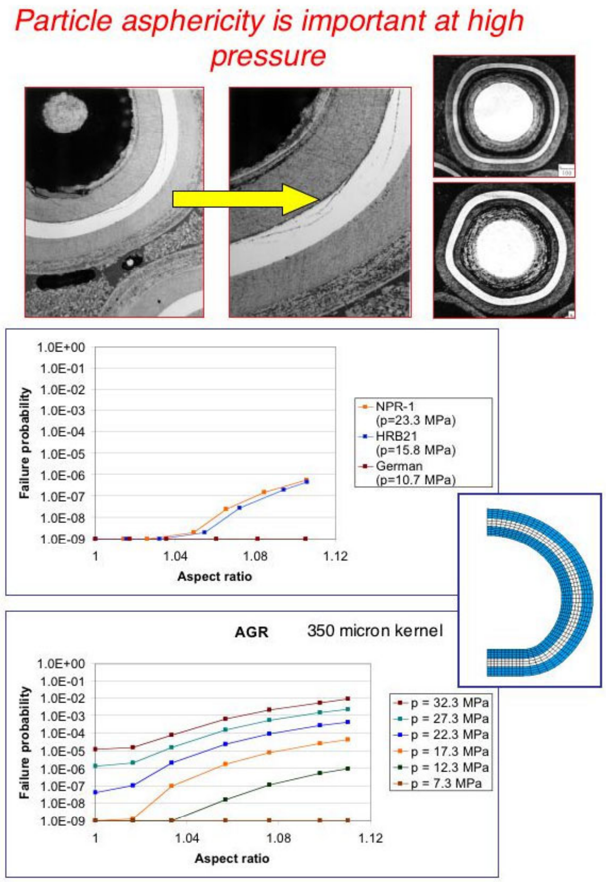


Figure 2-15. Effect of particle asphericity on failure probability.

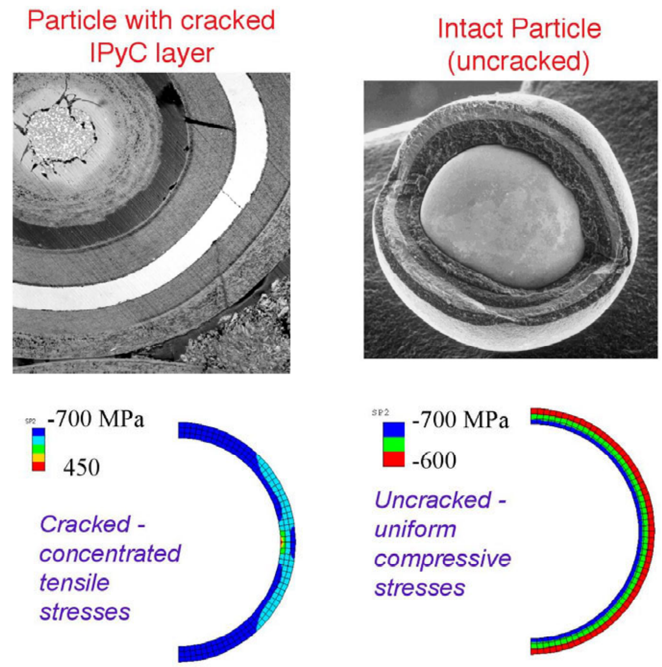


Figure 2-16. Cracked inner pyrolytic carbon layers could lead to SiC layer failure.

These models have had some success in predicting fuel failure mechanisms and rates in the U.S. fuel tested over the last decade, thereby facilitating a better understanding of TRISO coated fuel behavior. Such a tool can be very useful for both pretest and posttest predictions for any experiment performed in this program. In addition, sensitivity studies with the model can be used to identify critical materials properties data and constitutive relations whose uncertainty needs to be reduced because they drive the predicted performance of the coated fuel particle. Furthermore, use of piggyback cells (small encapsulated fuel samples outside the compacts) in the irradiation capsules can be used to study those key individual phenomena in coated particles that have high uncertainty (e.g., shrinkage and swelling of pyrocarbon, fission product release behavior in a purposely defective or initially failed particle). Moreover, some of the post-irradiation examination techniques can provide maps of fission products through the particle, which can be compared with model predictions of fission product transport through the coatings. All of this type of data will eventually be needed to validate the overall TRISO coated fuel performance model. Such fuel performance models will eventually be needed to provide some understanding of fuel behavior inside the operations and safety envelope defined by the irradiation and safety testing (i.e., interpolation) and outside these envelopes where the margins of failure of the fuel may be approached (i.e., extrapolation). Finally, a validated fuel performance model can be used to help evaluate and guide potential future changes in the next-generation coated particle fuel.

The importance of fuel performance modeling has been recognized internationally. The United States is part of the IAEA Coordinated Research Project on coated particle fuel technology. A key task is associated with benchmarking coated particle fuel performance models under both normal and off-normal conditions. The fuel behavior models under development by the AGR program are part of the international benchmark.

2.2.5 Fission Product Transport and Source Term Modeling

Transport of fission products produced within the coated particles will be modeled to obtain a technical basis for source terms for advanced gas reactors under normal and accident conditions. The design methods (computer models) will be validated by experimental data as necessary to support plant design and licensing.

The NRC will require validated computer models that accurately predict the following phenomena:

- Fission product release from the kernel
- Transport through failed coatings
- Deposition fraction of the released fission products in the compact or sphere matrix
- Deposition fraction of what gets through the compact on fuel element graphite (prismatic variant only)
- Deposition fraction of what gets out of the fuel element onto graphite dust and metallic surfaces in the primary circuit
- Re-entrainment of deposited fission products during an elevated temperature accident, or depressurization event
- Transport of fission products on dust particles, and subsequent release to the environment if the primary circuit is breached.

Each of the phenomena listed above is complex and difficult to model. It is also difficult to design and conduct experiments that can cover the multitude of variables that affect the physical situation. The AGR program has developed a research and development plan that, when the work is successfully completed, will produce a technical basis for source terms under normal and accident conditions for advanced gas-cooled reactors. The program consists of irradiations to provide data on fission gas and fission metal release from the kernel and transport through failed coatings (AGR-3), fission product transport behavior in the fuel element matrix and graphite block (AGR-4), out of pile experiments to characterize plateout, and reentrainment of fission products during accident conditions. The program also contains an irradiation (AGR-8) that will be used to validate computer models that describe the in-vessel gas reactor source term.

2.3 Schedule

Figure 2-17 is a high-level schedule for the entire program leading to fuel qualification for the NGNP. Figure 2-18 is a high-level schedule leading up to the AGR-1 irradiation. Note that the years shown in these schedules are calendar years.

ID	Task Name	Start	Finish
1	**Fuel Development**	**Oct 1 '04**	**Dec 29 '17**
2	**Fuel Supply Fabrication Development**	**Oct 1 '04**	**Oct 1 '12**
3	Fuel Fabrication Small Coater for AGR 1	Oct 1 '04	Sep 1 '06
4	Coater Scaleup and Fuel Fabriation for AGR 2,3,4	Jun 6 '06	Sep 3 '08
5	Fuel Fabrication AGR5-8	Jan 2 '12	Oct 1 '12
6	**Fuels and Materials Irradiations**	**Jan 12 '07**	**Oct 14 '16**
7	AGR 1 Irradiation - Shakedown Small Coater Fuel	Jan 12 '07	Jul 14 '09
8	AGR 2 Irradiation - Large Coater fuel Performance	Jan 30 '09	Aug 2 '11
9	AGR 3 Irradiation - Fission Product Transport Data	Aug 3 '09	Feb 1 '12
10	AGR 4 Irradiation - Fission Product Transport Data	Jul 30 '10	Jan 29 '13
11	AGR 5 & 6 Irradiation - Fuel Qualification	Oct 1 '13	Mar 31 '16
12	AGR 7 & 8 Irradiation - Fuel Performance & Fission Product	Apr 16 '14	Oct 14 '16
13	Post Irradiation Examination and Safety Testing	Jun 24 '09	Mar 24 '17
14	Fuel Performance Modeling - Code Development	Oct 1 '04	Jul 3 '15
15	V&V of Fuel Performance Code	Jul 1 '15	Dec 29 '17
16	Fission Product Transport and Source Term	Oct 1 '04	Jul 3 '14
17	Fission Product Transport & Source Term Code V&V	Jul 1 '09	Mar 31 '17
18	Preliminary NRC Topical on fuel Performance	Dec 31 '14	Dec 31 '14
19	Final NRC Topical on Fuel Performance	Jul 1 '16	Jul 1 '16
20	NGNP source Term Validated	Mar 31 '17	Mar 31 '17
21	Fuel Performance Model Validated	Dec 29 '17	Dec 29 '17
22	First Core Fuel Fabrication	Oct 2 '14	Oct 5 '16
23	Complete the NGNP Fuel Qualification Program	Dec 29 '17	Dec 29 '17

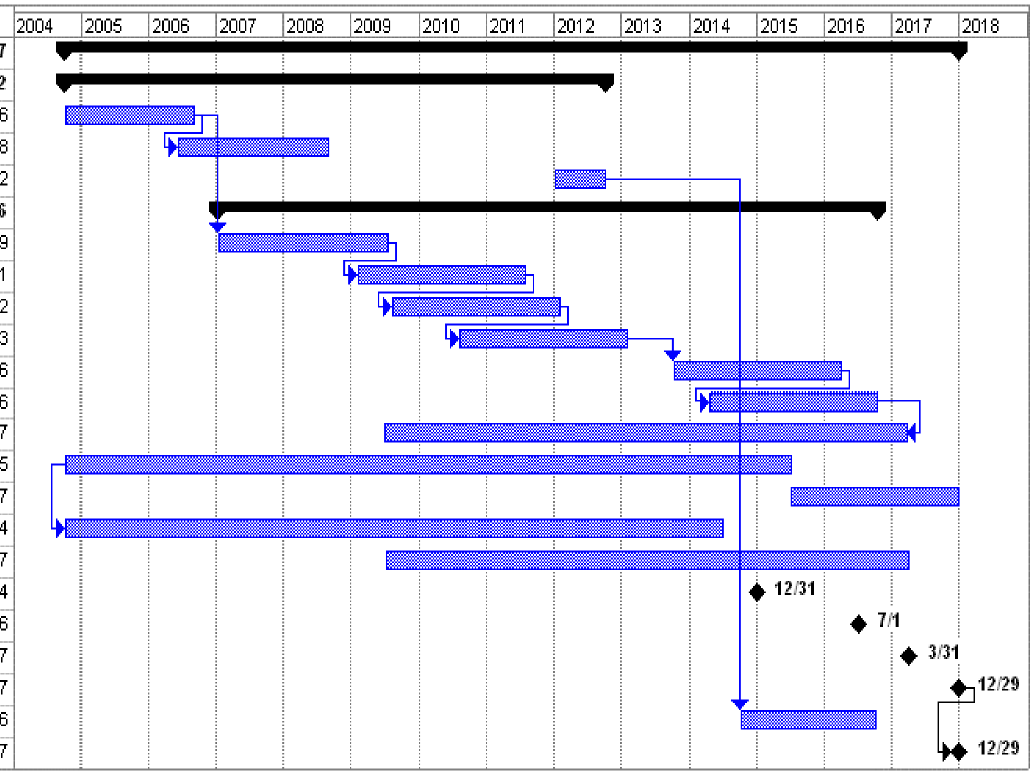


Figure 2-17. AGR high-level schedule leading to qualification of fuel and source term for NGNP.

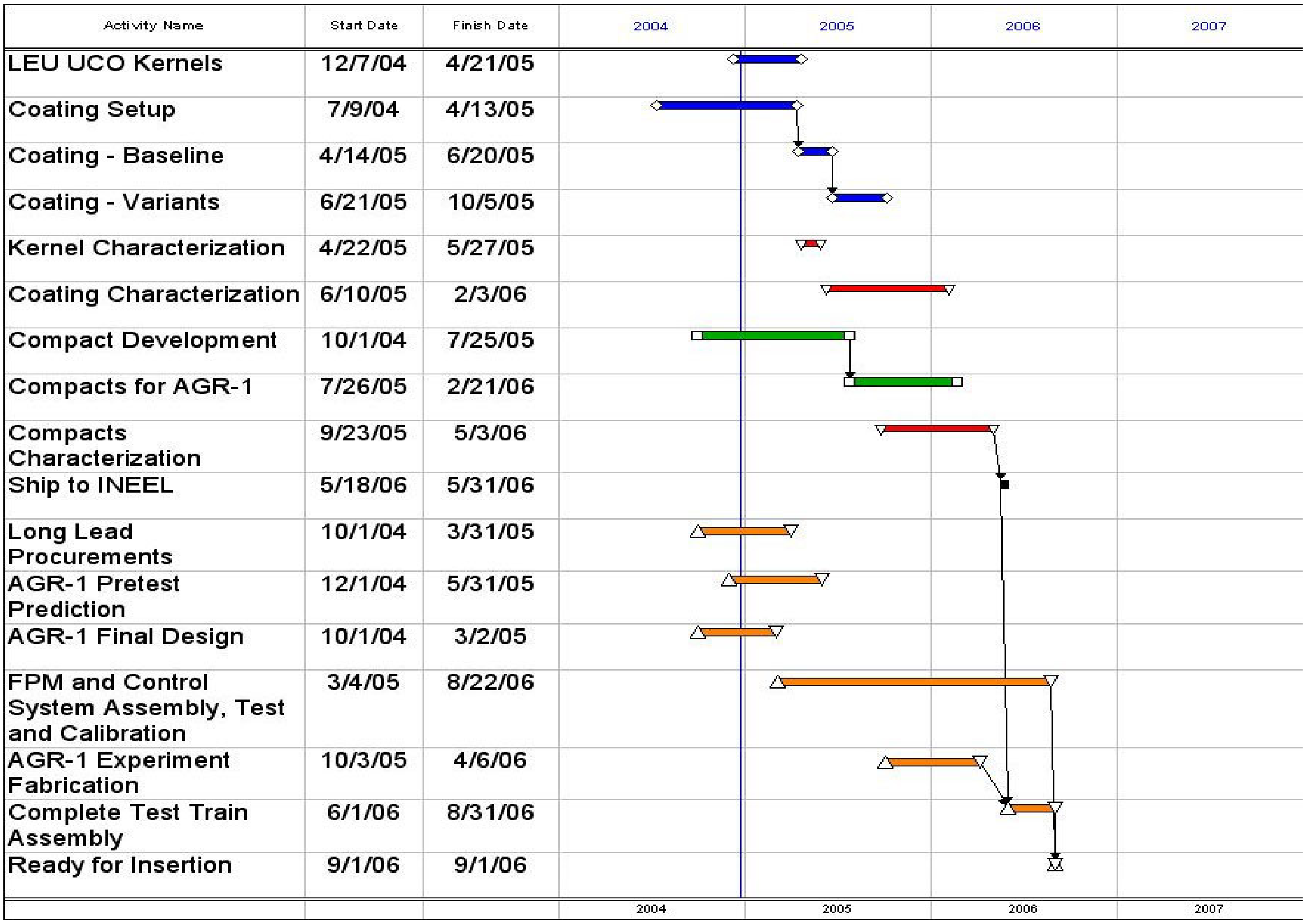


Figure 2-18. Schedule of activities leading to AGR-1 irradiation.

2.4 Summary and Conclusions

The NGNP AGR Fuel Development and Qualification Program consists of five elements:

- Fuel manufacture,
- Fuel and materials irradiations,
- Safety testing and post-irradiation examinations,
- Fuel performance modeling, and
- Fission product transport and source term modeling.

The goal is to qualify the fuel form for use in the NGNP to the following[b]:

- Burnup of 15–20% FIMA,

[b] Specific fuel service conditions subject to change as the NGNP core design advances.

- Volume average time average temperature of 1150 °C,
- Time average peak temperature of 1250 °C, and
- Fast neutron fluence of 5×10^{25} n/m^2 (E>0.18 mev),
- High fission product retentiveness for hundreds of hours at 1600 °C.

The fuel form is based on reference UCO, SiC TRISO particles bonded by a matrix of graphite flour and carbonized thermosetting resin, incorporating past German fabrication experience.

An underlying theme for the fuel development work is the need to develop a more complete fundamental understanding of the relationship between the fuel fabrication process, key fuel properties, the irradiation performance of the fuel, and the release and transport of fission products in the NGNP primary coolant system during both normal operation and any conceivable accident. The logic of the program is structured such that there are multiple feedback loops and opportunities for improvement in the fabrication process based on early results. The fuel performance modeling and analysis of the fission product behavior in the primary circuit are important aspects of this work. The performance models are considered essential for several reasons, including guidance for the plant designer in establishing the core design and operating limits, and demonstration to the licensing authority that the applicant has a thorough understanding of the in-service behavior of the fuel system. The fission product behavior task will also provide primary source term data needed for licensing.

3. MATERIALS RESEARCH AND DEVELOPMENT

3.1 Introduction

The NGNP Materials R&D Program will focus on testing and qualification of the key materials commonly used in VHTRs. The program will address the materials needs for the NGNP reactor, power conversion unit, intermediate heat exchanger, and associated balance of plant. Materials for hydrogen production will be addressed by the DOE's Nuclear Hydrogen Initiative [DOE 2004]. Revision 1 of the NGNP Materials R&D Program Plan [Hayner et al. 2004] was issued in September 2004. The R&D discussed in this document is based on that plan.

The current organizational structure for management of the Materials R&D program is shown in Figure 3-1. The NGNP Program is the primary interface with DOE-NE and the industry partner (following selection). The NGNP Materials R&D Program Manager interfaces with the Generation IV Materials National Technical Director and the NGNP Program System Integration Manager to establish program elements. Input, interface, and recommendations are also obtained from the INL Materials Review Committee (MRC), the Materials Quality Assurance Program (QAP), and the Generation IV VHTR Materials and Components Project Management Board (PMB). Work Packages and detailed program elements are based on DOE guidance and available funding.

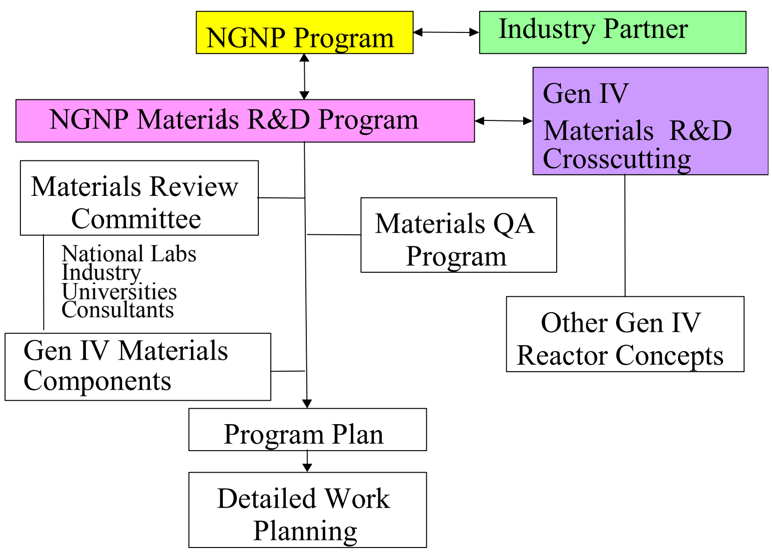


Figure 3-1. NGNP Materials Organization Structure.

The NGNP MRC is a senior independent review body for the materials R&D program. Russell Jones from Pacific Northwest National Laboratory (PNNL) chairs the committee. The MRC provides objective technical review of key selected materials program activities, including test materials selection decisions, test program content, test results, etc.

The Generation IV Reactors Materials Program within the Generation IV Initiative has responsibility for establishing and executing an integrated plan that addresses crosscutting and reactor-specific research needs in a coordinated and prioritized manner. The Generation IV Reactors Materials Cross-cutting and the NGNP Materials R&D Program are both part of the integrated Generation IV Materials Program. The NGNP Program is currently the highest priority reactor concept within the Generation IV Program. Consequently, the Generation IV Materials National Technical Director and the NGNP Materials R&D

Program Manager work closely to correctly define materials R&D program area priorities and detailed work scope to be performed.

The National Technical Director of the Generation IV Reactor Materials Cross-cutting Program will ensure that crosscutting materials R&D activities:

- Support the NGNP Materials R&D Program activities with a minimum of duplication and overlap
- Support the NGNP product team efforts to ensure integration of product requirements into the R&D activities.

There is no actual design for the NGNP; however, Table 3-1 compares nominal parameters of VHTR designs currently being developed with the Advanced Boiling Water Reactor (ABWR) design. Note that the actual design selected for the NGNP could be different from the information presented. Therefore, the information in Table 3-1 should be viewed as illustrative, not specific of the NGNP.

Table 3-1. Comparison of nominal parameters for prismatic and pebble bed design and ABWR.

Reactor Pressure Vessel Parameter	ABWR	GT-MHR	GA-Prismatic [b]	NGNP Prismatic [a]	PBMR [c]	NGNP PBR [a]
Nominal gas outlet temperature (°C)		850	950	1000	900	1000
Nominal gas inlet temperature (°C)		491	590	490	482	490 [d]
Normal operating temperature (°C)	284	495	350	470 [e]	300	465 [f]
Worst case accident temperature (°C)		565	530	560 [a]	506	560 [f]
Inlet gas pressure (MPa)		7.07	7.07	7.07	8.9	7
Outlet gas pressure (MPa)		7.02	7.02	7.02	8.6	6.5?
External diameter (meters)	7.2	8.2	8.2	8.2	7.02	7.06
Nominal wall thickness (mm)	175	100–300	100–300	100–300	120–220	120–220
Nominal height (meters)	21	23.7	23.7	24	22	19
Maximum radiation fast fluence over 60 years (n/cm^2)		3×10^{18} [g]		1×10^{19} [a,h]	4.5×10^{19} [a]	3.0×10^{19}

a. MacDonald et al. 2003. b. Richards et al. 2004. c. DOE 2003. d. 490 °C, based on recent analysis still pending publication. e. If the temperature is 490 °C on the inside, then a temperature drop is assumed to reach about 470 °C. f. 490 °C, based on recent analysis still pending publication. g. GEN IV 2003. h. Core barrel, Neutron Energy Group 2 ($5.105\ 10^9 \times 3600 \times 24 \times 365 \times 60 = 10^{19}$).

All work performed to support the technical program for the NGNP Materials R&D Program will utilize the national consensus standard ASME NQA 1997, "QA Program Requirements for Nuclear Facilities Applications," and Subpart 4.2 of ASME NQA 2000, “Guidance on Graded Application of Quality Assurance (QA) for Nuclear-Related Research and Development,” for project-specific materials development activities.

The QA requirements for specific projects under the NGNP Materials R&D Program will be specified in project-specific Quality Plans and project-specific Technical Specifications. The project-specific quality plans will include the necessary management controls commensurate with the project work scope and importance to the program. There are currently two project-specific Quality Plans, one at Oak Ridge National Laboratory (ORNL), the other at Idaho National Laboratory; both are DOE national laboratories. At ORNL, the quality plan is titled Quality Assurance Plan for the Next Generation Nuclear Plant Materials Program at Oak Ridge National Laboratory, QAP-ORNL-NGNP-01, Rev. 1. The INL Quality Plan is titled Quality Program Plan for the INL NGNP Materials R&D Program, PLN-1792. The

DOE Nuclear Energy Research Initiative university work activities may be managed under the umbrella of either the ORNL or INL Quality Plan.

3.2 ASME Codification

Once appropriate materials have been designated for NGNP use, it will, of course, be necessary to gain ASME boiler and pressure vessel (B&PV) Code acceptance of those materials at the desired operating conditions.

Section II of the ASME B&PV Code is developed and maintained by the subcommittee on materials; it addresses materials approved for use by the construction subcommittees. Besides various specifications for ferrous, nonferrous, and welding materials, Section II also contains material properties such as Young's modulus, thermal conductivity, allowable stresses and stress intensities, etc. To achieve B&PV Code acceptance, specific material information must be submitted to the appropriate subcommittees. Satisfying the requirements of Appendix 5 of Part D, Section II, of the ASME B&PV Code may require significant effort. The NGNP's higher temperatures and operating environment may require even further efforts. Once the material is accepted in Section II, it must also be submitted for construction approval in Section III, Subsection NH, as discussed in the next few paragraphs.

The design rules of subsection NH for Class 1 elevated-temperature components consist of:

1. Load-controlled (primary) stress limits (Section III, Div I-NH Appendix I)
2. Strain, deformation, and fatigue limits (Section III, Div I-NH Appendix T).

The load-controlled stress limits are in the form of time-dependent allowable stresses based on both short-time tensile test results and long-term creep test results. Allowable stress reduction factors for weldments are given, as are reduction factors to account for the degrading effects of prior service. Only elastic analysis results are required to satisfy the primary stress limits.

The second category of design rules—strain, deformation, and fatigue limits—are much more problematic. These rules deal with complex behavior resulting from primary plus cyclic secondary and peak stresses. They aim at preventing failures due to excessive deformation, creep-fatigue damage, and inelastic buckling, and they generally require inelastic design analysis results to satisfy them. The rules include strain accumulation limits, creep-fatigue criteria,c buckling limits, and special limits for welds. The materials currently covered, allowable life times, and maximum allowable temperatures are limited in Subsection NH, as shown in Table 3-2. Only the temperature limits for Alloy 800H come close to those required for the NGNP vessels. Coverage is inadequate for any of the materials in the very-high-temperature NGNP components.

Aside from the fact that many preliminary candidate NGNP materials are not included in Subsection NH, there are several generic shortcomings that will require resolution. First, the maximum temperatures permitted will have to be significantly increased. Second, allowable time-dependent stresses will have to be extended beyond the current 300,000 h maximum to 600,000 h. Third, environmental effects (impure helium) need to be incorporated into the failure criteria, particularly creep-fatigue. Fourth, creep-fatigue damage accumulation rules need to be revised to more accurately predict actual material failure conditions.

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[c] As currently formulated in Subsection NH, the creep-fatigue rules are based on a linear damage accumulation rule, an interaction diagram to account for the synergistic effects (and for environmental effects in the case of ferritic steels), and multiaxial strength theories for both fatigue and creep rupture.

Table 3-2. Current Subsection NH materials and maximum allowable times and temperatures.[a]

Material	Temperature (°C) [b] Primary Stress Limits and Ratcheting Rules	Fatigue Curves
304 stainless steel	816	704
316 stainless steel	816	704
2 1/4 Cr – 1 Mo steel	593[b]	593
Alloy 800 H	760	760
Modified 9 Cr – 1Mo steel (Grade 91)[c]	593[b]	538

a. Allowable stresses extend to 300,000 h (34 years) unless otherwise noted.

b. Temperatures up to 649 °C are allowed for not more than 1000 h.

c. Draft form only, waiting for approval.

Time-dependent structural tests will provide data that either validates the high-temperature design methodology (HTDM) or leads to changes in inelastic design analysis guidelines or Code rules. The role of structural tests will be even more important for the NGNP materials because of the lack of long-term service experience. Very-high-temperature, time-dependent tests of structural models (1) provide better understanding of structural behavior and failure modes, (2) validate inelastic analysis methods, and (3) provide some applications feedback to the Code.

It should be emphasized that the structural tests to be performed in this Materials Program are not tests of NGNP component structures. Rather, they are tests of carefully chosen, simple, but representative, geometrical and metallurgical features subjected to time-varying thermal and mechanical loadings. The tests are contrived to explore key features or problem areas of the methodology. Past examples include beams, plates, thick-walled cylinders subjected to thermal gradients, capped cylindrical shells, and nozzle attachments. Cylinders and plates with notch-like discontinuities and with axial or circumferential welds were included. The latter two types of tests will be particularly important to NGNP because of the two major NRC concerns of weldments and discontinuities.

While not strictly a part of the design methodology, the safety assessments required for licensing depend on much of the same materials and structures database. A particular need is for a flaw assessment procedure capable of reliably predicting crack-induced failures and the size and growth of the resulting opening in the pressure boundary. High-temperature flaw assessment guides have been developed in France, Japan, and the United Kingdom, and work on elements of a procedure is currently underway in the United States under Pressure Vessel Research Council sponsorship. An overall proven procedure does not exist however, which will require inelastic analysis of flawed components, characterization of sub-critical creep and fatigue crack growth, and a structural failure criterion. These will be developed for the NGNP materials.

Four current ASME B&PV Code cases and a draft Code case are relevant to the HTDM project:

1. Case N-499 was developed for HTGRs. It permits Class 1 components fabricated in accordance with ASME SA-533, Grade B /SA Grade 508 steels to exceed the normal 371 °C low-temperature design limit for short periods for Levels B, C, and D events. A similar case might be developed for the NGNP vessel material under off-normal conditions.
2. Case N-201 provides rules for construction of core support structures made of ferritic steels, austenitic stainless steels, and high-nickel alloys, and having metal temperatures not exceeding those in Section II, Part D. This Case, with modifications, might be useful for the metallic core internals of NGNP. The basis for the Case is the same high-temperature structural design methodology as that on which Subsection NH is based.

3. Code Case N-253 provides rules for Class 2 and 3 components for elevated temperature service. Unless exemption rules are met, the case essentially defaults to the criteria of Subsection NH.
4. Code Case N-290, which covers expansion joints in Class 1 liquid-metal piping, can serve as a starting point for criteria and design methods for the NGNP bellows.
5. A draft Code case developed in the 1980s for design of Inconel 617 to 982 °C is directly pertinent to NGNP [Corum and Blass 1991]. The original request for the case came from DOE and General Electric. The specific gas-cooled reactor component of primary interest was a steam-methane reformer, which was to be part of the reactor primary pressure boundary. Materials of potential interest included Alloys 800H, X, and 617. Alloy 617 was chosen for the case because it was a leading choice of designers, and a reasonable database of material properties existed. The case was developed by an ad hoc group of the Subgroup on Elevated Temperatures Design (SG-ETD). The case was subsequently approved by SG-ETD and submitted to its parent group, the Subcommittee on Design, for approval. However, further action on the case was suspended when the DOE project was canceled. The case is of value to NGNP because it can serve as a springboard for establishing NGNP Code rules. It was the result of a five-year effort of experienced high-temperature materials and structures engineers, as well as gas-cooled reactor project participants. It also had the participation and input of researchers from the Japanese Atomic Energy Research Institute (JAERI) and the Institute for Chemical Technology (KFA) in Germany. The draft case, while having the same framework as Subsection NH, has several unique features that are ramifications of the very-high-temperature material behavior. This behavior includes (1) lack of clear distinction between time-independent and time-dependent behavior, (2) high dependence of flow stress on strain rate, (3) softening with time, temperature and strain. Therefore, the design rules of Subsection NH that are based on the separation of time- and rate-independent response, or on strain-hardening idealizations of material behavior require careful reconsideration in the case. For example, the case specifies that inelastic design analyses for temperatures above 649 °C must be based on unified constitutive equations, which do not distinguish between time-independent plasticity and time-dependent creep.[d] The draft case also recognizes that significant environmental effects on Alloy 617 could exist, and it recognizes that extended exposure at elevated temperature may cause a significant reduction in fracture toughness of Alloy 617, thus introducing an additional failure mode—brittle fracture—to be considered. Finally, because of the uncertainties in data extrapolation and the lack of experience in designing to such high temperatures, where allowable stresses are very low, the draft case is limited to design lives of just 100,000 h or less.

3.3 Component Candidate Materials

A variety of options have been identified for potential use of materials in the NGNP reactor and balance of plant components. These options originated through an initial look at the materials issues for a very high-temperature reactor in January 2003 [Baccaglini et al. 2003] and a much larger, focused NGNP materials options identification activity that included meetings at INL and ORNL in July 2003. The information shown in this section is a summary of the options identified because of these activities and any others that have been identified since the July meeting.

3.3.1 Reactor Core Graphite, Reflector, and Supports

Graphite will be the major structural component and nuclear moderator in the NGNP core. The graphite used previously in the high-temperature gas reactor programs in the United States, H-451, is no longer in production, and thus replacement graphites must be found. Hence, it will be necessary to qualify new grades of graphite for use in the NGNP. Fortunately, likely potential candidates currently exist, including fine-grained isotropic, molded or isostatically pressed, high-strength graphite suitable for

[d] This is also the case for the high-alloy ferritics (e.g., 9Cr – 1Mo steel) at the upper end of their useful temperature range.

core support structures, fuel elements, and replaceable reactor components, and near isotropic, extruded, nuclear graphite suitable for the above-mentioned structures and for the large permanent reflector components. These candidates would meet the requirements of the draft ASTM materials specification for the Nuclear Grade Graphite.

3.3.2 Reactor Internals

The reactor internals that need to be addressed will be design specific, but most likely include a core barrel, inside shroud, core support floor, upper core restraint, and shutdown cooling system shell and tubes.

For the very-high-temperature components (>760 °C), the most likely material candidates include variants or restricted chemistry versions of Alloys 617, X, XR, 230, 602CA, and variants of Alloy 800H. Alloys 617, X, and XR were developed for earlier, gas-cooled reactor projects. Alloy 617 has the significant advantage in the United States of having gone through ASME Code deliberations that culminated in the draft code case discussed above, and the body of experts that developed the case simultaneously identified what must be done before the Code case could be approved. Alloys X and XR have a significant database and body of experience in Japan. Alloy 602CA is a relatively new high-temperature alloy that has been approved for Section VIII, Division I, construction to 1800 ºF. Alloy 230 has good high-temperature and environmental resistance properties and is approved for Section VIII, Division I, Construction to 1650 ºF. Alloy 800H is in Subsection NH, and would be the leading candidate for the intermediate temperature range of 600–760 ºC. Interest exists to extend its application beyond 760 ºC.

However, the upper limit of these materials is judged to be 1000 °C. Any component that could experience excursions above 1000 °C would need greater very-high-temperature strength and corrosion resistance capabilities. C_f/C or SiC_f/SiC composites would then be the leading choices for materials available in the near future for service that might experience temperature excursions up to 1200 °C. For similar high-temperature service at some later point in the future, oxide dispersion strengthened alloys may be an alternative. Compatibility of the metals with the helium coolant and irradiation resistance of the potential candidate materials needs to be addressed.

3.3.3 Primary Coolant Pressure Boundary System

Several possible primary coolant pressure boundary systems are envisioned for the NGNP. These comprise a large reactor pressure vessel containing the core and internals, a second vessel containing an intermediate heat exchanger and circulator (or a power conversion unit), and a pressure containing cross-vessel joining the two vessels. Because of the wide range of material thickness in the primary coolant pressure boundary system, it will be constructed in a segmented configuration. The three vessels will be exposed to air on the outside and helium on the inside, with emissivity of the chosen material an important factor regarding radiation of heat from the component to the surrounding air to ensure adequate cooling were accident conditions to develop.

The primary coolant boundary system will either use conventional materials as listed within the ASME SA 508/SA 533/SA182 specifications, or it will be fabricated from materials never used previously for a nuclear reactor in the United States. If the temperature can be maintained to less than 375 ºC by cooling or other means, conventional materials can be used. However, if the pressure boundary is in the range of 375–500 °C, advanced materials will be required. The advanced materials tentatively selected for further investigation for the gas-cooled primary coolant pressure boundary system service are ferritic/martensitic steels, alloyed primarily with chromium and molybdenum. The two most promising classes of commercially available steels are:

- **9Cr-1MoVNb (SA182).** This class of materials has the most industrially mature high-strength database. For example, the 9Cr-1Mo-V (Grade 91) alloy is ASME Code-approved to 649 °C for

Section III, Classes 2 and 3, components and is in the final stages of approval for inclusion in Subsection NH for Class 1 applications. Within this class of alloys, it seems prudent to consider variants such as 9Cr-1MoWV (Grade 911, Grade 92, etc.), because available research data show significantly improved high-temperature strength for those alloys relative to Grade 91.

- **2.25Cr-1Mo (SA508).** There is an extensive database for this alloy, including data in different operating environments, such as helium. Another advantage is the extensive industrial experience with this alloy in many different applications around the world. However, its high-temperature strength is significantly lower than the alloy class discussed above and, as such, is only applicable for substantially lower vessel temperature, such as in the case of the HTTR at JAERI.

3.3.4 Control Rod and Composite Structures

A number of structural composites were identified for potential use in control rods and other composite structural applications in the NGNP. The components and potential materials are shown in Table 3-3. The reason that composites of either carbon/carbon (C_f/C) or SiC_f/SiC must be considered for these applications is long-term exposure to temperatures greater than 1000 °C. At these temperatures, most metallic alloys are ineffective.

Table 3-3. Potential structural composite applications.

	Graphite	C_f-C	SiC_f-SiC
Hot duct		X	X
Core support pedestal	X		
Fuel blocks	X		
Replaceable outer/Inner reflector blocks	X		
Top/bottom insulation blocks	X		
Upper plenum block	X		
Floor block	X	X	X
Upper core restraint and upper plenum shroud (Structural liner and insulation)		X	X
Control rods and guides		X	X

Future qualification tests will be required to delineate which of the composites are the best choice for a given component, based on the response of the composite to exposure conditions expected within the reactor. C_f/SiC composites are not included in the table because they might exhibit cracking problems due to the use of dissimilar materials. Use of C_f/C composites appears to be desirable for many applications within the reactor because of their strength retention at high temperatures. Ceramic composites made from silicon carbide fibers and silicon carbide matrices (SiC_f/SiC) are also promising for nuclear applications because of the excellent radiation resistance of the β phase of SiC and their excellent high-temperature fracture, creep, corrosion, and thermal shock resistance. In addition, there is some evidence that SiC_f/SiC composites have the potential to be lifetime components (no change-out required) within the high radiation environment within the core. Unfortunately, the SiC_f/SiC composites have not been as well characterized as C_f/C composites, so there is more uncertainty in the applicability. Therefore, it will be necessary to carefully evaluate both C_f/C and SiC_f/SiC for the control rod material.

3.3.5 Intermediate Heat Exchanger

An intermediate heat exchanger will be needed for hydrogen production and other process heat applications. It may also be desirable to use an indirect cycle for electricity production. The reactor coolant system pressure will be about 7 MPa, and the difference from the primary to secondary circuit pressure may be small (0.1 MPa) if helium is used for the intermediate heat transfer loop or it may be larger if a liquid salt is used for the intermediate heat transfer loop. If liquid salts are used, the intermediate heat transfer loop can be operated at any desired pressure, including the option of maintaining the pressure half way between reactor coolant pressure and the pressure in the chemical plant heat exchangers.

The intermediate heat exchanger will be contained within a pressure vessel. The leading intermediate heat exchanger design for this cycle is a compact counter-flow configuration that involves channels passing through diffusion-bonded metallic plates. Transient thermal loadings could be a problem, but the details needed to identify the materials performance requirements will depend on the design selected. Environmentally induced degradation of the metals from impurities in the helium or flow induced erosion is a concern. Aging effects are a concern for very-long-time thermal exposure, since embrittlement could affect the performance of the intermediate heat exchanger during thermal transients. Welding/brazing and fabrication issues exist that will depend on the intermediate heat exchanger design details. Again, the leading potential candidates for service at temperatures of 900 to 1000 °C are Alloy 617, Alloy X, and Alloy XR. Other nickel-base alloys such as CCA617, Alloy 740, and Alloy 230, could be considered. Possibly, the compact intermediate heat exchanger could be fabricated from a C_f/C composite. Alternate intermediate heat exchanger designs such as tube-and-shell intermediate heat exchangers introduce concerns that can only be addressed when more is known about the performance requirements.

3.3.6 Power Conversion System

The key components of the NGNP power conversion unit will include turbines, generators, and various types of recuperators or heat exchangers. Considerable materials work may be involved in both the turbine and the generator components, and existing component manufacturers are an excellent source of the needed materials information. As such, much of the turbine and generator materials efforts will be performed via subcontracts to existing manufacturers. However, early efforts should be conducted to identify the materials preferred by various manufacturers and to assess the performance potential of these materials under operating conditions representative of the NGNP.

The recuperator may be a modular counter-flow helium-to-helium heat exchanger; its most likely design has corrugated-plate heat exchange surfaces. Recuperator technology for the temperatures and pressures of operation is relatively mature. For gas turbine applications, tube-on-plate and primary surface units are often fabricated from fine-grained 300 series stainless steels. Recuperators in which the corrugated plate surfaces are sealed by brazing have suffered from thermal fatigue when pushed to higher temperatures, but the NGNP operating conditions may not subject the recuperator to severe cycling.

3.3.7 Valves, Bearings, and Seals

A few valves may be required in the primary or secondary piping systems for this plant, and a flapper valve may be used in the shutdown cooling system. Bearing surfaces exist between the reactor pressure vessel and the core barrel. Seals may be required in a variety of locations. However, insufficient information relating to the specific requirements and issues relating to valves, bearings, and seals is available at this time to initiate R&D activities.

3.4 Materials Qualification Testing Program

The following discussion follows an order of priority established for the materials R&D work. The projects funded in FY-05, in order of priority, include:

1. Test and qualify core graphite materials
2. Develop an improved high-temperature design methodology for use of selected metals at very high temperatures

3. Develop ASME and ASTM codes and standards
4. Perform environmental testing and thermal aging of selected high-temperature metals
5. Reactor pressure vessel materials irradiation testing and qualification
6. Develop and qualify composites
7. Resolve reactor pressure vessel fabrication and transportation issuesThe focus of each project described in this document is on work to be performed in the first two to three years of the program, before completion of the NGNP conceptual design. The content of the projects is specifically designed to envelop the high-priority and long-lead materials R&D information anticipated to be required regardless of the NGNP system design chosen. Subsequent detailed project content will be aligned with design chosen for the NGNP by the industry partner. Note that not all of the projects discussed below are currently funded.

3.4.1 Graphite Testing and Qualification Project

Significant quantities of graphite have been used in nuclear reactors, and the general effects of neutron irradiation on graphite are reasonably well understood. A photograph of an early French gas reactor core is shown in Figure 3.2. However, models relating structure at the micro and macro level to irradiation behavior are not well developed. Also, as mentioned above, much of the past work was specific to a graphite known as H-451, which is no longer available.

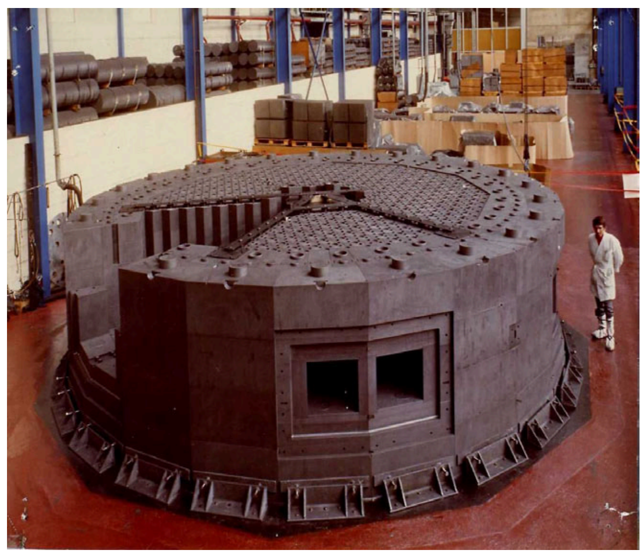


Figure 3-2. Photograph of an early French gas reactor core.

3.4.1.1 Graphite Selection Strategy

Several candidate graphites have been identified for components within the NGNP (Table 3-6). In selecting candidate graphites for the major components of the NGNP, several factors must be considered. Inclusion of all graphites in the materials R&D program is clearly cost prohibitive. Consequently, the scope of the NGNP graphite program will take into account the other activities within the GIF for graphite database development (especially irradiation experiments) and the graphite needs of the prospective reactor suppliers. Moreover, the criteria for selecting graphites will consider whether the particular graphite can satisfy multiple reactor vendor design requirements and whether there are sustainable precursors for extended production runs over the reactor's lifetime. Cost and schedule for this effort will be reduced by limiting the amount of material that needs to be irradiated and subjected to testing.

A strategy for the selection process, acquisition process, and material receipt and storage requirements for the purchased graphite is being developed. GIF members and potential reactor vendors have been solicited for input. A draft GIF collaboration plan for graphite procurement and testing should be available for discussion in December 2004. A series of meetings in Europe in early 2005 will finalize the selection and acquisition aspects.

Table 3-6. Candidate graphites for the core components of the NGNP.

NGNP Concept	Component Description	Candidate Grades
Prismatic block	Fuel element and replaceable reflector	Graftek PCEA SGL Carbon NBG-10,17,18 Toyo Tanso IG-110
Prismatic block	Large permanent reflector	Graftek PGX SGL Carbon HLM
Prismatic block	Core support pedestals and blocks	Graftek PCEA SGL Carbon NBG-10,17,18 Carbone USA 2020 Toyo Tanso IG-110
Prismatic block	Floor blocks and insulation blocks	Graftek PCEA SGL Carbon NBG-10,17,18
Pebble bed	Reflector structure	Graftek PCEA SGL Carbon NBG-10,17,18 Toyo Tanso IG-110
Pebble bed	Insulation blocks	Graftek PCEA SGL Carbon NBG-10,17,18

3.4.1.2 Graphite Irradiation Creep Capsule Design and Planning

The graphite fuel and moderator blocks are subjected to compressive stress due to the mass of the core and tensile and compressive stresses because of thermal gradients and irradiation-induced graphite dimensional changes. When the reactor shuts down, the stresses generally reappear in the opposite (tensile) direction, and block failure may occur. Figure 3-3 shows some of the creep irradiation data obtained for H-451 in the 1970s. Note that significant graphite irradiation creep can occur at the temperatures and stresses of interest to the NGNP. Similar data needs to be obtained for the nuclear grade graphites available today.

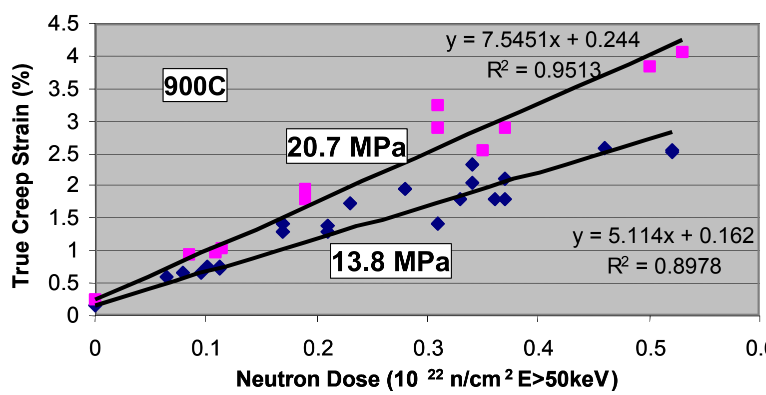


Figure 3-3. H-451 graphite creep strain at 900 °C as a function of neutron dose and compressive stress.

Engineers at the INL, in consultation with graphite experts at ORNL, have started an ATR creep capsule design. Prior Oak Ridge Research Reactor (ORR) and Idaho Engineering Test Reactor (ETR) graphite creep test capsule designs are being used as the basis for the new design (the previous ORR test capsule design used to produce the data in Figure 3-3 is shown in Figure 3-4). The graphite samples will be loaded under compressive stress and irradiated at representative temperatures. In addition to creep rate data, postirradiation examination of the control samples will yield valuable irradiation effects data. Early planning is underway to establish the irradiation goals and test parameters. The graphite samples will be selected from multiple vendors and grades of graphite. The capsule will be designed, all necessary QA documentation prepared, and an experimental plan prepared in FY-05. Capsule construction and bench testing will commence in FY-06. It is anticipated that creep irradiations will be completed in FY-07 or -08.

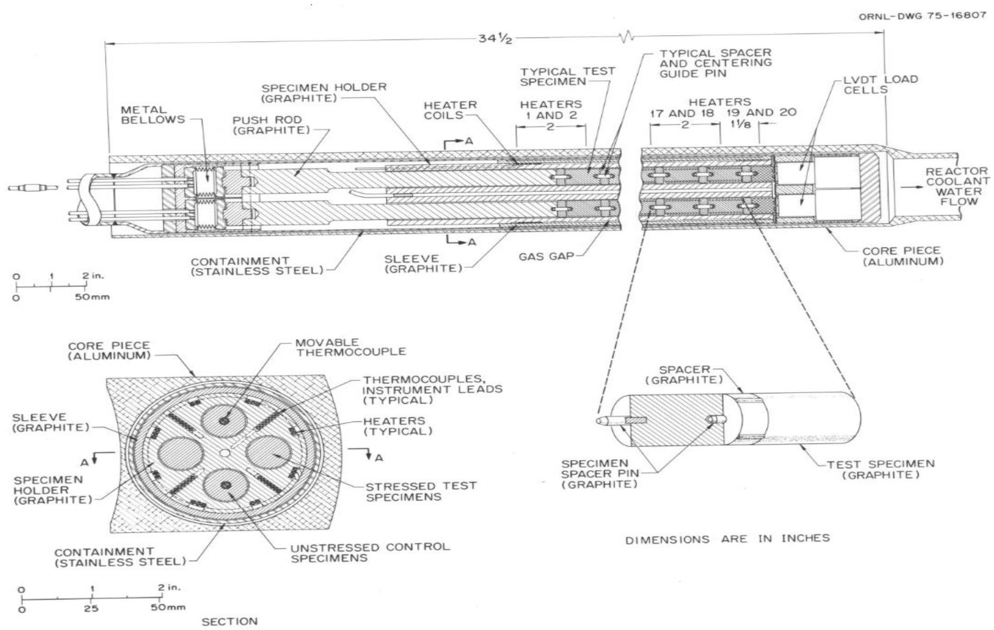


Figure 3-4. Oak Ridge Research Reactor graphite creep capsule design.

Design of the capsule requires 4 to 7 columns of 12-mm-diameter graphite rods be irradiated in a temperature-controlled environment. The graphite rods will consist of multiple specimens 25 mm in length. The specimens will alternate in each stack between different nuclear grade graphites from different vendors. The stressed samples will be subjected to compressive loads by a pressurized metallic bellows exerting a constant force. A load cell will be located in the test train of each of the columns to allow real time determination of the load on the stressed specimens. The load cell will be located above the core at a sufficient distance to minimize damage from heat and radiation. The capsule will have a mixture of helium/argon gas flowing through the capsule to control the temperature and the gas pressurizing the bellows. A PC controller will record the multiple thermocouple temperatures and load cell data. The temperature and column loading will be automatically controlled within specified limits. It is estimated that 170 days of irradiation will be required in the south flux trap position in ATR to achieve the desired fluence of 4.1×10^{21} n/cm^2 for E> 0.18 MeV.

3.4.1.3 Graphite Model Development for Predicting Irradiation Effects

Mathematical models must be developed that describe and predict the behavior of nuclear graphite under neutron irradiation. Such models should be based on physically sound principles and reflect known structural and microstructural changes occurring in graphites during fast neutron irradiation, such as changes in crystallinity, pore shape, coefficient of thermal expansion (bulk and single crystal), etc. Models for the graphite irradiation dimensional changes and irradiation creep behavior are a priority. Existing irradiation data may be used for model development, but validation of the models must be conducted using irradiation data obtained on the newer nuclear graphites being considered for the NGNP. Input data for such models must be obtained from the NGNP candidate graphites. Several modeling approaches will be explored. For example, models will be considered based on microstructural changes described by bulk and crystal coefficient of thermal expansion changes, or on fundamental atom-displacement models linked to finite element codes.

Initial discussions have centered on setting up a computer model test bed that would be used to model the entire graphite core behavior with the material properties developed from ongoing postirradiation examination and future irradiation tests. The test bed will be finite element based. Initial work on this project will determine which of the commercial finite element packages will be used to support the work.

The OECD/NEA Expert Group meeting on Modeling of Microstructure Property Relationships in Irradiated Graphite, SiC, and C/C Composites at High Temperatures at Manchester University in the UK during January 2005 should provide a suitable format to establish an initial modeling approach that is coordinated with our international partners.

3.4.1.4 HFIR Rabbit Capsule Postirradiation Examination

Significant dimension and material property changes can occur in graphite subjected to neutron irradiation. Figure 3.5 shows the dimensional changes that occur in H451 as a function of orientation, temperature and dose. Therefore, a series of 36 NGB-10 nuclear graphite bend-bar samples were irradiated in rabbit capsules in the High-Flux Isotope Reactor (HFIR) at ORNL during FY-04. Each of the 18 rabbit capsules contained a SiC temperature monitor. Post-irradiation examination of the samples will begin at ORNL in FY-05 and will include determination of the following irradiation effects data:

- Volume change
- Dimensional change (parallel and perpendicular to extrusion)
- Dynamic modulus (parallel to extrusion)
- Flexural strength (parallel to extrusion)
- Thermal diffusivity and conductivity (parallel and perpendicular to extrusion)
- Structural change (via scanning electron microscope examination)
- High-temperature annealing study (~1500 °C)

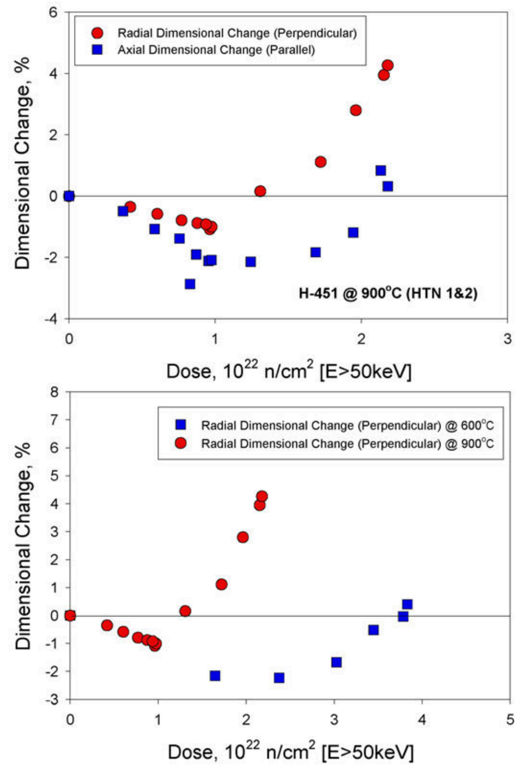


Figure 3-5. H-451 graphite dimensional changes as a function of orientation and temperature.

3.4.1.5 High-temperature Graphite Irradiation Experiments

There are few data for the irradiation behavior of graphite at temperatures >1000 °C. Hence, a high-temperature graphite irradiation capsule for use in HFIR will be designed that will be capable of irradiating graphite samples at temperatures up to 1200 °C. Evaluation will determine the most appropriate HFIR vehicle for these irradiations based on capsule size limitations, ease of attaining the desired temperatures, and availability of space in the HFIR (e.g., rabbit capsule, target capsule, or reflector capsule). The first capsule will be designed, along with an experimental plan and required QA documentation, in FY-05. Irradiation data to be determined on the candidate graphite(s) will include the items listed above in Section 3.4.1.4. The pre- and post-irradiation examination will be conducted at ORNL.

3.4.1.6 GIF Graphite Irradiation Review

This task (currently unfunded) involves reviewing both historical and ongoing graphite irradiation data available through the International Atomic Energy Agency (IAEA). This will be a joint effort between INL and ORNL, requiring foreign travel to IAEA contributor's sites for discussion with principal investigators and physical collection of data. An NGNP white paper will outline directions for requirements for future irradiation capsules, follow on post-irradiation examination work, and requirements for future data collection.

3.4.2 High-Temperature Design Methodology Project

The High-temperature Design Methodology (HTDM) project will develop the data and simplified models required by the ASME B&PV Code subcommittees to formulate time-dependent failure criteria that will ensure adequate life. This project will also develop the experimentally based constitutive models that will be the foundation of the inelastic design analyses specifically required by ASME B&PV Section III, Division I, Subsection NH.

The HTDM project will produce test data, analyze results, and develop constitutive models for high-temperature alloys. Equations are needed to characterize the time-varying thermal and mechanical loadings of the design. Test data are needed to build the equations. The project will directly support the reactor designers on the implications of time-dependent failure modes and time and rate-dependent deformation behaviors. The project will also develop data for regulatory acceptance of the NGNP designs. Safety assessments required by NRC will depend on time-dependent flaw growth and the resulting leak rates from postulated pressure-boundary breaks. This requires a flaw assessment procedure capable of reliably predicting crack-induced failures and the size and growth of the resulting opening in the pressure boundary. Identifying an overall proven procedure is a part of this project.

The HTDM effort is required because the current high-temperature metal alloys design rules are based on the separation of time- and rate-independent response, or on strain-hardening idealizations of material behavior. Components operating at high temperature respond to thermal and mechanical loadings inelastically. At the lower temperature end of a material's useful elevated-temperature operating range, the inelastic response can usually be separated into time-independent plasticity and time-dependent creep. Each can exhibit complex, history-dependent hardening or softening, and the two types of response can interact with one another (i.e., prior plastic strains affect subsequent creep responses, and vice versa.). At higher temperatures, the distinction between rate-dependent plasticity and time-dependent creep blurs for many materials (e.g., modified 9Cr – 1Mo steel and Alloy 617), and the separation between behaviors is no longer valid. The response becomes rate and time dependent, and both strain and cyclic softening occur. This high-temperature inelastic response has significant implications for structural design, involving plasticity, creep, viscoplastic behavior, and inelastic ratcheting.

The use of linear damage fractions and the current linear damage accumulation design rule for creep fatigue is inadequate at higher temperatures and longer operating times. Various improvements in damage methodology, such as those based on ductility exhaustion and damage rate, have been proposed, but sufficient work to allow for their acceptance as a replacement for the linear damage accumulation rule in B&PV Code Section III, Subsection NH, has not occurred.

Most high-temperature structural failures occur at weldments. Welded pipe, for example, has failed in high-temperature fossil plants after many years of operation. Reliably guarding against weldment failures is particularly challenging at high temperatures, where variations in inelastic response of the constituent parts of the weldment (i.e., weld metal, heat-affected zone, and base metal) can result in a strong metallurgical discontinuity. In the hearings for a construction permit for the Clinch River Breeder Reactor Project, NRC identified early weldment cracking as the foremost structural integrity concern. The NRC believed that designers should have better understanding of the metallurgical interactions that take place in weldments and their effects on weldment life. The CRBRP committed to a five-year development program to address these issues before issuance of a plant-operating license. The program was never carried out because of the subsequent demise of the project.

As with metallurgical discontinuities, geometric discontinuities (i.e., notches and other local structural discontinuities) are sources of component failure initiation. The adequacy of the methodology to handle such discontinuities will likely be reliability and licensing issues, particularly when heat-to-heat variability, strain hardening/softening, and cyclic loadings are considered. In priority, this was the second highest unresolved issue (after weldments) in the CRBRP licensing hearings, and, again, the NRC required a development program. Reviewers believed that the effects of stress gradients were not reflected in creep-fatigue design limits, and that general notch weakening and loss of ductility under long-term cyclic loadings were not well understood.

The Alloys 617 and Grade-91 steel have been selected for use in initial improved high-temperature design methodology development. The primary reasons for this selection include:

- Pending ASME Section III draft code cases
- Known high-temperature mechanical properties of the alloys
- Extensive use of the Alloy 617 in non-nuclear very high-temperature applications.

This development is initially being directed toward Alloy 617 joints in the intermediate heat exchanger because we already have an existing Alloy base metal study underway, the joints are a critical weak link in high-temperature structures, and weldment fatigue data was identified as needed in earlier ASME and NRC reviews of Alloy 617 knowledge.

It is recognized that Alloy 617 is a very mature, high-temperature alloy, but it still has a number of issues that must be addressed to allow its longtime usage under the environmental and loading conditions envisioned. The ASME and the NRC have identified major shortcomings in understanding the interactions of creep, fatigue, and environment in these alloys and their weldments. Resolving these issues for Alloy 617 will develop both a technical approach to apply to other high-temperature alloys and reinvigorate the ASME activities needed for their codification within ASME Section III, Subsection NH.

The proposed program will begin to address these deficiencies by studying rate-dependent stress-strain behavior at relatively short times, creep, and creep-fatigue-environment interactions in Alloy 617, leveraging the results of existing programs on Alloy 617 base and weld metal and providing early data needed to complete development of high-temperature design methods required for its codification for nuclear service. Specific near-term activities are described in more detail in the tasks that follow. Other alloys will be added to the program based on need and funding provided.

3.4.2.1 Alloy 617 Specification

The standard specification for Alloy 617 may have to be modified to:

- Optimize long-term properties at elevated temperatures
- Minimize environmental effects caused by exposure to VHTR He impurities at elevated temperatures.

Therefore, we may need to develop a controlled chemistry specification for Alloy 617. It is expected that the specification will be developed based on literature review and consultation with Special Metals and the French. The controlled chemistry Alloy 617 specification previously developed for fossil program will also be being evaluated. The evaluation will allow procurement of Alloy 617 in FY-06.

3.4.2.2 Alloy 617 Joint Characterization Experiments at the INL

A new servo-hydraulic load frame was ordered at the INL to support this work during FY-04. Procurement and checkout of an environmental chamber for this new servo-hydraulic load frame so that creep-fatigue testing can be performed in a controlled environment will be completed in FY-05. In addition, the Alloy 617 fusion welds will be characterized and the basic microstructural properties and strength characteristics of the welds will be determined. As with the base metal testing, the subsequent high-temperature testing will incorporate extensive microstructural study of the damage development to determine the micro-mechanisms by which loading and environment interact in the welds, thereby achieving a better theoretical underpinning for component lifetime models and high-temperature structural design methodology. The results may also reveal unforeseen synergisms between weld microstructures, loading, and environmental exposure.

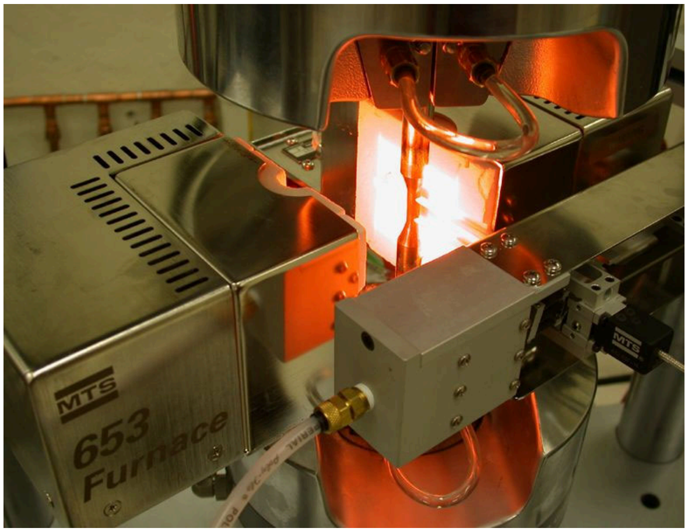


Figure 3-6. Creep-fatigue test in progress in air at 1000 °C at the INL (furnace opened to show specimen).

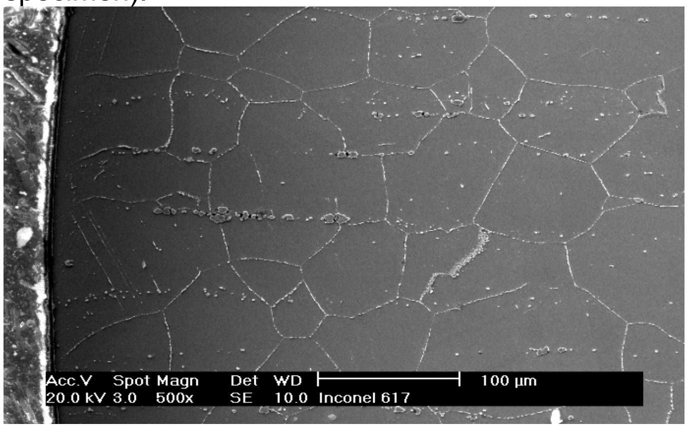


Figure 3-7. Microstructure of Alloy 617 aged 100 hours at 1000 °C with course carbide on grain and twin boundaries.

The creep-fatigue testing at the INL will be performed on Alloy 617 specimens with fusion welds in impure helium at 800 to 1000 °C. The program will leverage programs currently in progress, including work on Alloy 617 within the Ultra-Supercritical Steam Boiler (USCSB) program at ORNL and the Materials for Energy Research (MER) program at the INL. The objective MER work is to study fundamental creep-fatigue-environment interactions in Alloy 617 and clearly distinguish environmental and mechanical damage components, and also compare behavior in aggressive and inert environments. The project has three components:

1. Creep-fatigue (C-F) testing and microstructural analysis
2. Development of laser ultrasonic NDE techniques
3. Phenomenological and atomistic modeling

The preliminary test variables include:

–Environments: air and high-purity He (inert)

–Temperatures: 800 and 1000 °C

Figure 3-6 shows one of the MER tests in progress in air at 1000 °C. Figure 3-7 shows the microstructure of Alloy 617 aged 100 hours at 1000 °C. Note the course carbides on the grain and twin boundaries.

3.4.2.3 Testing and Constitutive Equation Development at ORNL

Aging of inert atmosphere encapsulated base alloy and welded samples for 10,000 h and 1000 °C will also be performed at ORNL. Post-exposure characterization will include microstructural examination, limited creep-fatigue testing, and fracture toughness testing. The characterization will provide time-dependent input for constitutive equation development and a baseline of thermal aging effects in the absence of environmental effects relating to impure helium exposure.

Scoping tests of Alloy 617 creep in helium environments and stress-strain evolution will be performed on base metal and welded specimens. Creep and creep-fatigue tests will be concentrated in the temperature range from 800 to 1000 ^{0}C to complement the lower temperature testing being conducted in the USCSG program. A limited set of initial test conditions will be determined, based on base metal data to complement those in the existing USCSB and MER programs. These conditions will include tests in impure helium and inert or oxidizing environments. Some of the ORNL equipment to be used for this work is shown in Figure 3-8.

Figure 3-8. ORNL environmental creep machines used in former high-temperature gas-cooled reactor projects will be refurbished to support the NGNP program.

Alloy X, XR, 800, and 800H evaluations will also be performed, which will include the procurement of these materials, from which fusion welds will be produced and characterized.

3.4.2.4 Alloy 617 Data Base

As a companion activity to the high-temperature scoping tests and before the substantial effort needed to generate the large database of mechanical property data needed for codification, a thorough assessment and compilation of existing Alloy 617 data are required. This effort will assess the validity of existing test data (e.g., is the helium environment in which data were generated indicative of NGNP conditions?) Substantial advantages in terms of time and cost will be gained by leveraging the USCSG database. A partial database from the ASME Materials Property Council may exist and may be acquired as well. The results of the proposed scoping tests and the review of existing materials database on Alloy 617, including the USCSG's database, will allow researchers to construct a well-defined and focused materials test plan. An in-depth survey of literature will be conducted of components at very high temperature. This will include constitutive equations for stress-strain evolution under various loading conditions for Alloy 617 and Alloy X/XR, efforts at addressing multi-axial effects on damage, and extrapolation of relatively short creep data for use in designing a reactor for a 60-year life. The information will be integrated into the Generation IV materials database.

3.4.3 Support for the ASTM and ASME Code

There are a number of areas relating to ASTM standard method development and ASME B&PV Code development that must be pursued to meet the NGNP goals. The NGNP Materials R&D Program must initiate a presence at ASTM and ASME B&PV Code meetings at the relevant committee and subcommittee level in order to incorporate new materials and/or extend the application of materials presently in the Code and/or further develop test standards.

3.4.3.1 Development of Elevated Temperature Design Rules for Metallic Alloys

Much of this effort will provide required technological support and recommendations to the Subgroup on Elevated Temperature Design (NH) as they develop methods for use of Alloy 617 at very high temperatures. In addition to inheriting the known shortcomings of Subsection NH, the Alloy 617 draft ASME Code case has a number of gaps and shortcomings that must be overcome before it can be satisfactorily and reliably applied. Therefore, a new ASME Code case needs to be written. The following tasks were identified as the Code case was being developed:

- Alloy 617 must be added to the low-temperature rules of ASME Section III
- Weldment stress rupture factors must be added
- Thermal expansion coefficients must be added
- Additional isochronous stress-strain curves must be added
- Create simplified methods.

In addition to the Alloy 617 related code work, code work will proceed on extending the usage of 2-1/4 Cr-1 Mo and 9 Cr-1 Mo (Grade 91) reactor pressure vessel steel. ***ASME and ASTM Activities in Support of Graphite and Carbon Composites***

ASME design code development is also required for the graphite core support structures of the NGNP and later for the C_f/C composites structures of the core. A project team under Section III of ASME is currently undertaking these activities. Participation of both ORNL and INL staff is anticipated in this activity. Standard test methods are also required to generate data that may be used in the design code. The ASTM DO2-F committee on Manufactured Carbons and Graphites is currently engaged in the final stages of developing a Standard Materials Specification for Nuclear Grade Graphite, and is also developing several standard test methods for graphites (crystallinity by x-ray diffraction, surface area, thermal expansion, fracture toughness, and graphite oxidation, for example). Participation of ORNL and INL staff in the DO2-F committee work will continue. For example, a round robin evaluation of the oxidation method will be conducted. Similarly, ORNL will lead an assessment to determine the applicability of the existing ASTM method, the Brunauer Emmett Teller (BET) method, for measuring the effective surface area of graphite (information needed when assessing the effects of potential air or water ingress). The method will then be adopted into ASTM C-781 (Standard Practice for Testing Graphite and Boronated Graphite Components for High-Temperature Gas-Cooled Nuclear Reactors).

Also, the ASTM Committee DO2-F has identified a test method for determining the fracture toughness (K_{ic}), based on existing Standard C1421 (for advanced ceramics at ambient temperatures). This standard will be modified to apply to graphite, and ruggedness tests will be performed using several different graphites. Once a modified version of the standard test method has been established, round-robin testing will begin. ORNL will analyze the K_{Ic} data and develop the ASTM-required research report with precision and bias data. A standard test method for determining the K_{Ic} value of graphite will be prepared and approved through the ASTM DO2-F committee.

3.4.3.3 Working Group on Composites Testing

INL and ORNL will support the formation of an ASTM working group on SiC_f/SiC composite testing development and ensure that guidelines for testing of tubular SiC_f/SiC structures proceeds. INL and ORNL will interface with the design community to ensure that the appropriate properties are being targeted.

3.4.4 Environmental Testing and Thermal Aging of High-Temperature Metals

The three primary factors that will most affect the properties of the metallic structural materials from which the NGNP components will be fabricated are the effects of irradiation, high-temperature exposure, and interactions with the gaseous environment to which they are exposed. This work is focused on assessing the property changes of the metallic alloys as a function of exposure to the high-temperature and impure gas environments expected in the NGNP. The information below describes the overall work that needs to be performed.

3.4.4.1 Aging Tests

Procedures for the evaluation of aged and "service-exposed" specimens will be developed. Properties evaluation will be performed on a limited number of materials, including Alloy 617, Alloy 800H, and Alloy X, that have been aged at temperatures as high as 870 °C for long times in helium. It is expected that aging exposures of more materials will be performed to at least 25,000 h. Mechanical and microstructural properties of bulk and weld structures will be evaluated, and the determined experimental properties will also serve as input to and checks on the computational continuum damage modeling activity for predicting high-temperature life. Results of mechanical testing and microstructural evaluations of candidate alloys aged 1000, 3000, and 10,000 h will serve as additional input to computational continuum damage models. The predictions of these models will be compared to results of testing of materials aged to at least 25,000 h to provide for validation of these models. The mechanical and microstructural data will also provide input into code rules for accounting for aging effects.

The specimens for long-term aging studies of Alloy 617 will be fabricated at INL. A detailed long-term materials test matrix will be prepared to examine aging and environmental effects at very high temperature on Alloy 617.

A review will be performed of the extensive body of work on Alloy 617 and two other candidate materials to document the applicability of the available thermal aging effects data/information in the temperature range of interest to the NGNP. This review will also serve to highlight the areas where additional information is needed.

3.4.4.2 Evaluation of Helium Environments

The out-gassing of even nuclear grade graphites at very high temperatures may release significant impurities (H_2O, CH_4, CO_2, CO, N, and H_2) into the helium coolant of the NGNP. The overall stability of the NGNP helium environment must be evaluated to ensure the testing proposed in various parts of the program are performed in environments having consistent chemical potentials. In addition, the corrosion of metals and nonmetals will be evaluated to establish baseline data where it does not exist. These tests will be performed at temperatures to include at least 50 °C above the proposed operating temperature.

3.4.4.3 Helium Loops

Design and construction of a recirculating low-velocity helium loop with gas cleanup is proceeding at INL. Special emphasis is being placed on the gas clean up system, which will serve as the prototype for a high-velocity loop. The system will be designed to operate using vacuum or inert gas as the reference atmosphere, with capacity to mix ppm levels of impurities (e.g., H_2 or CO_2 or water vapor) designed to simulate the NGNP environment.

ORNL will also restart two recirculating low-velocity helium loops (Figure 3-9) and initiate gas/gas studies to establish the dynamic stability of selected metals exposed to a very high-temperature impure helium environment.

The existing data/information on the environmental effects of impure helium on Alloy 617 will also be reviewed to document the applicability of existing data for the range of temperature and helium compositions of interest to the NGNP. This review will also delineate the ranges in which additional data are needed.

In future years, long-term creep testing capabilities will be designed and/or augmented as needed. Existing creep facilities will be refurbished, and additional creep-fatigue equipment procured as necessary to meet the need for high-velocity and long-term testing of materials in potentially contaminated helium environments. A new test loop will be designed and constructed for performing the required testing. It is envisioned that the test capability will be helium, with controlled impurity levels at temperatures up to 1100 ℃, 7.5 MPa pressure, and flow rate up to 50 m/s. Despite the additional complexity and cost, it is apparent that a closed loop with gas clean-up capability and compressor will be the only feasible design. System testing will be in two stages. The first stage will evaluate helium with chemical modifications only. The second stage will modify the test loop so that particle erosion on test coupons will be added to the flow of controlled chemistry hot helium. Capability will be added for generating particulate and characterizing the density and velocity of particles. An important system capability for the erosion test loop will be the capability to control the angle of impingement of the particles on the surface.

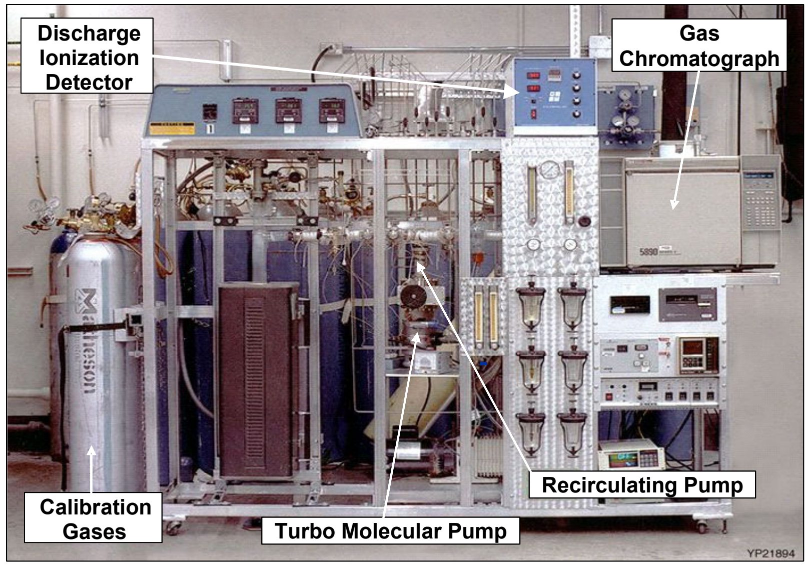


Figure 3-9. Low velocity helium loop at ORNL previously used for gas-gas studies to establish the dynamic stability of selected metal in very high temperature environments.

In addition, reactor pressure vessel alloy specimens will be prepared for thermal aging in air. The materials will be aged for 1000, 3000, and 10,000 h at 650 ℃. These experiments will yield relatively early indication of each material's response to long-time high-temperature exposure in air, a condition applicable to the uncoated outer surface of the reactor pressure vessel. The aged materials will then be tested for tensile, creep, and toughness behavior, and characterized microstructurally. Candidate materials and weldments will also be aged in the impure helium environment for the same times, mechanically tested, and microscopically examined. In addition, portions of the candidate materials and weldments will remain under thermal aging in both air and in helium until at least 25,000 h and tested to provide longer time data to allow for comparisons with predictive models. Finally, thermal aging of the prime candidate alloys at the reactor pressure vessel operating temperature will continue for more years to accumulate data for very long times.

3.4.5 Testing and Qualification of Reactor Pressure Vessel Materials

Some VHTR designs assume the use of higher alloy steel than currently used for LWR pressure vessels. The irradiation damage and property changes of these materials must be measured. Therefore, an irradiation facility that can accommodate a relatively large complement of mechanical test specimens will be designed and fabricated for placement in a material test reactor. This facility will replace the irradiation facility shown in Figure 3-10 and shut down last year at the Ford Test Reactor at the University of Michigan. The facility will, of course, include temperature control to allow for irradiation at the temperatures of interest, and operate at a flux low enough to provide results representative of the conditions anticipated for the NGNP reactor pressure vessel.

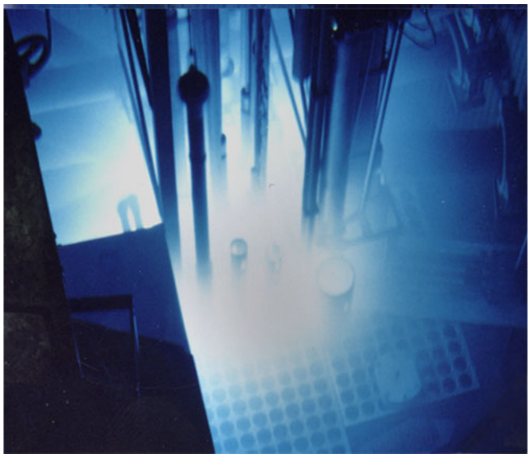


Figure 3-10. Reusable capsules for reactor pressure vessel materials testing at the University of Michigan test reactor.

The new irradiation facility is anticipated to be a joint DOE- NRC facility. Preliminary design concept envisions two separate and independent operating capsules in the facility, one for the NRC-funded Heavy-Section Steel Irradiation Program and the other for the Generation IV Reactor Materials Cross-Cutting and NGNP Programs. The capsules can be readily designed and fabricated to operate from 250 to 650 °C, with a preliminary fast neutron flux of about 1 to 2 × 10^{12} n/cm^2·s (>1 MeV). The next steps in this effort involve executing a Memorandum of Understand (MOU) between DOE and NRC, issuing a request for proposal (RFP) to potential irradiation facilities, and site selection. Useful hardware will be retrieved from the Ford Test Facility, and redesign and fabrication will be performed for the irradiation hardware.

Although the operating temperature of the reactor pressure vessel and cross-vessel may change with evolution of the design, it is currently planned to irradiate mechanical test specimens at about 350 to 600 °C. The choice of these temperatures is based on the assumptions that (1) 600 °C is the highest possible operating temperature that can be envisaged for the reactor pressure vessel and cross vessel at this time because of creep, (2) 350 °C is in the range of the lowest operating temperature that would likely require higher alloy pressure vessel material than the current SA508, which has been extensively tested, and (3) the range between these temperatures would likely provide sufficient information for design and operation of the reactor pressure vessel at any intermediate temperature with respect to irradiation effects. Irradiations of the preliminary candidate materials, both base metals and weldments, will begin in later years, with the choice of materials to be based on results of a literature review, as well as the baseline and aging tests completed at the time. For purposes of this plan, specimens to be irradiated will include those for tensile, hardness, creep and stress rupture, Charpy impact, fracture toughness, and fatigue crack growth testing. The currently estimated maximum exposure is about $1{\times}10^{19}$ n/cm^2 (>0.1 MeV) and 0.075 dpa. The specimens will be irradiated to an exposure about 50% greater to accommodate uncertainties in the exposure estimates. A limited number of irradiated specimens will be aged in the impure helium environment for up to 10,000 h, tested, and examined by optical and electron microscopy.

A decision to conduct further test reactor irradiations beyond those noted above will be based on the results of the initial testing. As currently required by 10-CFR-50, Appendix H, and prudence, the NGNP will incorporate a surveillance program. The specific design of the surveillance program will be based on the results obtained from the test program discussed above, but will likely include, as a minimum, tensile, Charpy impact, fracture toughness, and creep specimens. Because the NGNP is a demonstration reactor, the surveillance program will be more extensive than required by the regulatory authority, such that it could serve as a test bed for irradiation experiments of more advanced materials that may be developed as NGNP operations progress.

The fluences accumulated in the metallic core internal materials are expected to be low relative to the tolerances of those structural alloys. Nevertheless, some consideration of the irradiation effects in those materials is thought prudent. The radiation effects on the metallic reactor internal components will be reviewed. Exposures and evaluation of the irradiated materials will include evaluation of the radiation-induced changes in microstructure, hardness, and ductility.

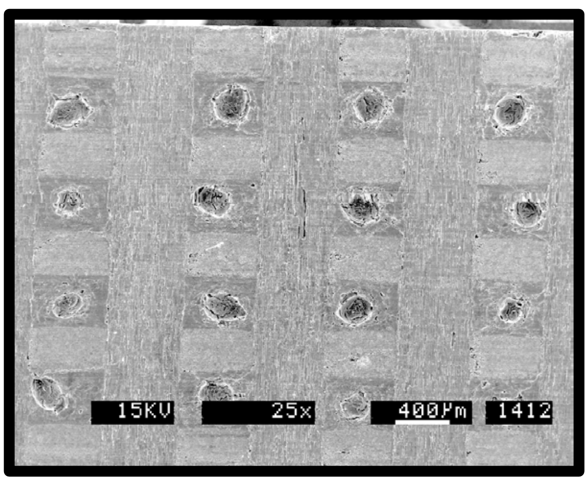
3.4.6 Composites Development, Testing, and Qualification

This program is directed at the development of C/C and SiC/SiC composites for use in selected very high-temperature/very high neutron fluence applications such as control rod cladding and guide tubes (30 dpa projected lifetime dose) where metallic alloy are not feasible. It is believed that SiC/SiC composites have the potential to achieve a 60-year lifetime under these conditions. The usable life of the C/C composities will be less, but their costs are also significantly less. The program will eventually include a cost comparison between periodic replacement of C/C materials and use of SiC/SiC composites.

Composite materials are being considered for these applications because they are superior "engineering" materials compared to monolithics. In particular, they have:

- •Higher strength, especially in tension
- •Higher Weibull modulus (more uniform failure)
- •Much higher damage tolerance (fracture toughness)

Figure 3-11 shows a photograph of a typical SiC/SiC composite cross-section along with a scanning electron microscope photograph of the fiber-matrix interface.



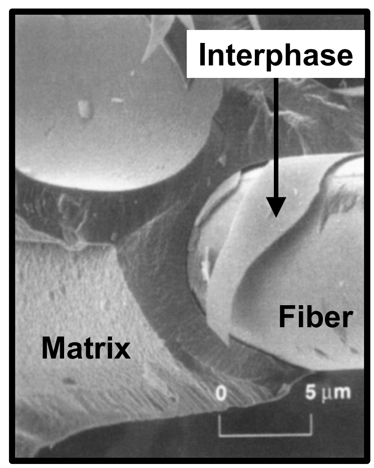


Figure 3-11. Photograph of a typical SiC/SiC composite cross-section (left), and scanning electron microscope photograph of the fiber-matrix interface (right).

3.4.6.1 Initial SiC/SiC Composite Irradiation Studies

Currently, SiC_f/SiC composites have only been irradiated to fairly low (8 dpa) levels. At this irradiation dose, the composites are stable and do not show much degradation after about 1 dpa. SiC_f/SiC composites may be stable out to at least 30 dpa without much degradation; however, this assumption needs to be validated.

ORNL currently has high purity SiC_f/SiC samples being irradiated to higher irradiation levels in HFIR in FY-05. It is expected that the specimens will reach about 10 dpa in FY-05 and 20 dpa in FY-06. Based on these preliminary results, the irradiation stability of SiC_f/SiC composites versus C_f/C composites at higher doses should be resolved. Assuming that SiC_f/SiC composites are more stable, ORNL will continue irradiating specimens to 30 dpa levels in HFIR over the next few years. This part of the research is simply to prove that SiC_f/SiC composites can survive in an irradiation environment without significant reduction in strength or structural stability (i.e., without "falling apart").

Post-irradiation testing in established hot-cell facilities at ORNL will be initiated in FY-05. Testing will include, but not limited to, the following: thermal conductivity, irradiation-induced dimensional change, sonic elastic modulus, room-temperature bend strength, burst strength of tubes, slow crack growth testing of irradiated bars in simulated NGNP gas, and scanning and transmission electron microscopy of irradiated materials.

In future years, ORNL will select appropriate tube architecture composed of Nicalon Type S fibers. Nicalon Type-S fibers are being used because of their previous, excellent radiation performance. Infiltration of the Nicalon Type S fiber matrix will be performed using chemical vapor infiltration with high-purity SiC. When completed this will be a multiplayer SiC interphase composite. Both flat plate and tubular geometries will be fabricated. Details of the architecture to be manufactured will be studied in the initial phases of this project to determine the optimum approach. The matrix will be fully crystalline beta-SiC.

3.4.6.2 Test Methods for SiCf/SiC Composities

A real problem exists for scaleup of composite materials. Unlike monolithic materials, these composites are engineered from two distinct materials using complicated vapor infiltration techniques. The material properties may be affected when the component geometry or size is changed significantly. This is a major consideration, since small sample sizes and more suitable geometries are required for test samples. It must be shown that the test samples adequately represent the true response of larger SiC_f/SiC tubes used for control rod applications.

Representative samples from these tubes need to be irradiated and fit into ATR irradiation positions. Test samples much smaller than the actual control rod diameters (about 1.25 to 4 in.) will be required. In addition, in order to simplify the test rig in the ATR, "dog-bone" shaped flat tensile specimens have been proposed. This would potentially allow use of one of the "A-hole" positions instead of the larger and more expensive flux trap positions in the center of the core. However, before these smaller dog-bone flat tensile specimens can be used, it needs to be established that they are truly representative of the large tubes that would be used for the control rods. The project will work with the ASTM to establish a proper test methodology to determine the size and geometry effects of these smaller test samples. A test matrix encompassing all sizes and shapes to be used will be established. A round robin testing program will be initiated for all laboratories (ORNL, INL, PNNL, and possibly others) with the appropriate number of specimens from each category. Before the ASTM testing, the following sample parameters will be established:

- Determine appropriate sizes for test specimens. Currently, sample sizes of 3/8, 1/2, and 3/4 in. have been suggested. These sizes will be verified with the appropriate ASTM subcommittee members.
- Determine appropriate dog-bone tensile specimen size. These samples are strictly intended to determine whether flat samples accurately represent right cylindrical tubular samples. One size only will be used, and results will be compared to the tubular samples.
- Determine a statistically accurate sample number for each sample type.

Once the sample matrix has been established, the participating laboratories will test the samples using similar testing methods. The results will be fed back to the appropriate ASTM subcommittee (or working group) and analyzed. From these data, the size and geometry effects on composite testing will then be established.

3.4.6.3 INL Out-of-Pile C/C and SiC/SiC Composite Creep Studies

INL has the lead in performing irradiation creep studies on the SiC_f/SiC composites. This task is designed to prepare INL for performing both out-of-pile and in-pile testing of composite creep samples (both SiC_f/SiC and C_f/C composites). Specific issues that must be addressed include:

- Design and modification of the existing INL creep test stands to accommodate inert atmosphere testing.
- Potential modification of some INL creep stands to accommodate very high temperatures (i.e., 1400 °C) for off-normal events.
- Purchase of additional necessary equipment to perform thermal creep studies for ceramic composite structures.
- Out-of-pile creep testing for baseline thermal creep results.
- Design, development, and coordination of SiCf/SiC, Cf/C, and graphite creep capsules where applicable.

Basically, the main purpose of this task is to prepare INL to perform ceramic composite creep studies. This includes building/rebuilding the high-temperature testing infrastructure at INL and providing the experience necessary to perform the higher-temperature creep experiments. Some of the anticipated infrastructure changes include modifying the existing creep frames at INL to accommodate ceramic tube specimens, designing/building new sample grips, building/purchasing new environmental creep chambers, purchasing new creep monitoring probes, purchasing new furnace elements (high-temperature elements), purchasing new furnaces, etc.

Preliminary thermal creep studies will be conducted once the appropriate equipment infrastructure is in place. Ceramic tubes (either tubes from Hypertherm or another supplier) will be tested in the modified/new creep frames. It is anticipated that small equipment modifications or changes to the samples/grips will need to be implemented before achieving optimal testing conditions. Experience gained from these out-of-pile thermal creep tests will be applied to the design of in-pile irradiation creep tests.

3.4.6.4 INL In-Pile SiCf/SiC and Cf/C Creep Tests

Composite samples representative of the control rod tube architectures will be irradiated within the ATR to fluence levels similar for a full-lifetime dose (i.e., 30 dpa). These irradiation test samples will necessarily be much smaller than the actual control rod diameters (of approximately 1.25 to 4 inches) to accommodate the small irradiation ports within the ATR. In addition, in order to simplify the test rig in the ATR, small, flat tensile specimens ("dog-bone" shaped) will be irradiated within one of the "A-hole" positions instead of the larger flux trap positions in the center of the core. This will provide a significant cost and time reduction in the composite testing.

A statistically representative number of flat, dog-bone shaped composite samples will be fabricated using fiber architectures similar to those utilized in the baseline mechanical tests in previous years. Irradiation creep is defined as the dimensional difference between a stressed and unstressed sample within the same irradiation field. Correspondingly, half of the samples will be loaded in tension to load levels anticipated for a nominal control rod. The remaining samples will have no loads placed upon them and will be used as the creep baseline. All samples will be subjected to dose levels of approximately 30 dpa at temperatures of 800 °C (through gamma heating) within the ATR.

After irradiation, all samples will be examined and differences noted between the samples. Dimensional changes between the stressed and unstressed samples will provide a total dimensional change (i.e., total amount of creep). Since in-situ monitoring of the samples is not possible during irradiation, the data available to determine the rate at which these materials creep will be limited at best. These initial irradiation creep tests are designed to provide a total or cumulative dimensional change to the control rod material over a lifetime equivalent dose within the NGNP reactor.

3.4.6.5 Environmental Effects on C/C and SiC/SiC Composites

It is assumed that the fundamental irradiation response will be similar for all composite architectures and geometries. However, using different composite architectures (i.e., weave angles, fiber tow counts, weave structures, etc.) can lead to differences in the engineered materials due to infiltration efficiency, fiber bending stresses, or matrix/fiber interface characteristics. The environmental conditions these materials will be subjected to may change the overall creep response of the composite (i.e., creep crack growth for fiber-reinforced materials).

PNNL has extensive experience in environmental degradation of SiC. They have developed a creep crack growth model to predict the environmental factors on the overall creep of the SiC_f/SiC composite structures. PNNL will expand this model to include flat, thin specimens (i.e., simulate flat dog-bone shaped tensile specimens). It is anticipated that the model may be further expanded to include the 3-dimensional tubular geometry if applicable or desirable later.

To improve the accuracy of the model predictions, PNNL will determine a limiting environment for elevated temperature tests. Most likely, the limiting environmental species in the helium environment will be the H_2/H_2O ratio. Assuming these species are the most damaging to the composites, PNNL will determine the degradation potential for various H_2/H_2O ratios using both modeling and experimental tests.

3.4.6.6 Cf/C Composites

The C_f/C composites have performance issues similar to the SiC_f/SiC composite structures for control rod applications. A typical C_f/C composite cross-section and weave pattern is shown in Figure 3-12.

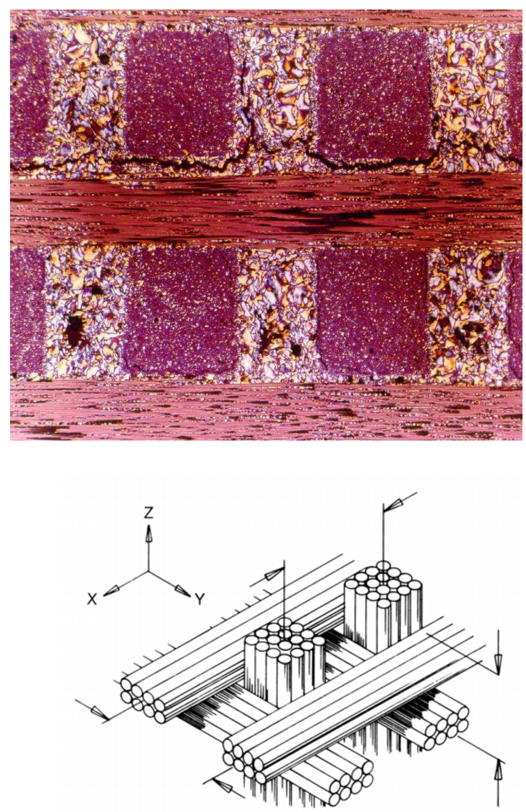


Figure 3-12. Typical C_f/C composite cross-section and weave pattern.

A survey of potential vendors will be conducted (domestic and foreign) to ascertain which vendors have the capability to fabricate complex architecture C_f/C composite components and what sizes can be processed. For the control rod assemblies, where neutron damage is a concern, consideration must be given to the ease of processing of the preferred fibers (mesophase pitch derived), which tend to have high modulus and are thus very difficult to weave. Heat treatment capabilities and furnace sizes/availability will be determined. NGNP designers will require this information in order to size the larger C_f/C components of the NGNP. ORNL personnel will conduct this study and issue a white paper report.

Candidate C_f/C composite materials for NGNP control rod applications will be purchased against a materials specification. The materials will be typical of those used in the NGNP components in terms of their fiber and matrix selections, and processing conditions. It is anticipated that a review of New Production Reactor literature and R&D activities in this area will be conducted before placement of a purchase order. Existing 3D C_f/C materials will be evaluated for the control rod application. Irradiation program needs will be evaluated.

3.4.6.7 C/C and SiC/SiC Composites I-NERI

The United States and France have agreed to fund a three-year I-NERI directed at the composites R&D needs for the VHTR. INL will coordinate the project with the other entities involved: PNNL, CEA, LCTS (University of Bordeaux), and SNECMA/GE Energy. The objectives of this project are to:

- Develop tubular SiC/SiC composite material for control rod and guide tube structure applications with requisite thermal, mechanical, and irradiation resistance properties
- Optimize the tubular SiC/SiC properties with advanced material synthesis methods using high-purity materials
- Generate a property database for the optimized materials using standard test methods (ASTM)
- Compare the results obtained for tubular SiC/SiC composites and flat plate composites made by the current state of the art
- Irradiate SiC/SiC tubes at high flux levels for long times at elevated temperature and perform post-irradiation examinations.

SNECMA will develop innovative 2D SiC/SiC technologies using multilayered interphased (combined with Hi-Nicalon) fibers, with support from LCTS. GE Energy will manufacture the very long SiC/SiC tubes necessary for qualification testing for control rod applications in the NGNP.

High-temperature testing of SiC/SiC tubes in impure He gas will include:

- Tensile tests (LCTS, INL)
- Creep experiments (PNNL)
- Delayed fracture and crack growth resistance (PNNL)
- Failure analysis as a function of composite architecture, including interface design (LCTS)
- Thermal conductivity versus temperature (PNNL).
- Specifics of the irradiation testing and the post-irradiation examination will be defined later. The irradiation testing will be performed at INL. The following deliverables are planned:
- First SiC/SiC tubes delivered to PNNL and INL (FY-05, LCTS)
- Optimized material design report (FY-05, LCTS)
- Failure analysis report (FY-06, LCTS).

Other deliverables will be determined following an initial planning and review meeting to be held in Cocoa Beach, Florida in January 2005.

3.4.7 Data Management and Handbook

The organizational structure for the preparation, control, etc., of NGNP data needs will be finalized for incorporation into the *Gen IV Materials Handbook* being developed in the Materials Crosscutting Program. Existing materials handbooks will be examined to determine what information might be extracted and incorporated into the *Gen IV Materials Handbook*. The primary documents to be reviewed will be the DOE-funded *Nuclear Systems Materials Handbook* and the *AFCI Materials Handbook*, followed by relevant portions of other ASME, Pressure Vessel Research Committee, American Society for Metals, etc., documents.

A *Gen IV Materials Handbook* plan will be prepared to identify needed management structures, advisory groups, working bodies, etc. This will establish the details of the handbook's scope and format, including what materials to include (at least initially), what properties to incorporate, and how these are to be presented. It may be that hands-on physical preparation and maintenance of the handbook will best be by an outside organization familiar with preparation of similar documents. This task will assess this possibility and, if appropriate, identify and down-select among the qualified outside sources.

A *Gen IV Materials Handbook* implementation plan will be prepared to detail the purpose, preparation, publication, distribution, and control of the handbook. It will also prescribe records required, QA, and review and approval responsibility and authority. Once fully implemented, the Handbook will become the repository for the NGNP materials data and serve as a single source for researchers, designers, vendors, codes and standards bodies, and regulatory agencies. It is also planned to evaluate the potential for including similar data from GIF international partners. Near-term activities in this area will include assembling and inputting existing data on materials of interest to NGNP.

3.4.8 Reactor Pressure Vessel Transportation and Fabrication Assessment

Reactor pressure vessel heavy section fabrication is a major issue that needs to be evaluated. Several potential candidate pressure vessel steels have been identified for the reactor pressure vessel and cross-vessel (see Section 3.3.3). It is unlikely that manufacturing of the NGNP reactor pressure vessel will take place in the United States. Preliminary considerations and discussions indicate that Japan Steel Works is the most likely source of forgings of the required size. The physical size of even the largest required forging appears to be within their range of capability; however, the specific material selection is critical in that very large forgings of most of the potential candidate alloys (other than SA508) have not been manufactured, including the 9Cr-1Mo-V alloy.

The main issue will be attaining the required through-thickness properties of the higher-alloy steels in such thick sections. In addition, welding of the steels in thick sections is also an issue. Therefore, fabrication and inspection will be major considerations in the selection of materials. Besides the technical issues, transportation of the completed reactor pressure vessel or large ring forgings from the vendor facilities to the reactor site may be an issue. It is possible that the reactor pressure vessel will require field fabrication, meaning welding of the ring forgings, heads, etc., onsite. In this case, the conduct of the post-weld heat treatment takes on more significance in that the post-weld heat treatment is more difficult to conduct and control than when performed in the shop.

This review will enlist the assistance of consultants with expertise in large vessel fabrication, particularly with low-alloy and medium-level chromium ferritic steels. If high-chrome low alloy steels are retained as the prime candidate materials, they will require developing fabrication, heavy-section welding, and post-weld heat treatment. Production of such forgings with the potential candidate alloys will be evaluated during the review. The assessment will also include transportation of individual ring forgings or a partially completed reactor pressure vessel to the United States and a fully completed reactor pressure vessel to the construction site in Idaho. The assessment will include evaluation of domestic welding and heat-treating capabilities for the potential case of final fabrication of the reactor pressure vessel in the United States and transport of the completed reactor pressure vessel to the construction site.

In future years, a fabricator will be chosen to fabricate forgings of sufficient size to represent the largest and thickest one required for the reactor pressure vessel. These forgings would be evaluated with mechanical testing and microstructural characterization. As a part of this task, a review will be conducted of nondestructive examination (NDE) procedures for the preliminary candidate materials. If the review indicates need to develop procedures specific to those materials, NDE procedures will be developed with a view toward satisfying the requirements of the ASME Code and the NRC, and incorporate the procedures in the required in-service inspection program.

3.4.9 Reactor Pressure Vessel Emissivity (Unfunded in FY-05)

Emissivity data are needed on the various potential candidate materials for the reactor pressure vessel. These are necessary because cooling of the reactor pressure vessel during normal operation and, most important, during a pressurized or depressurized conduction cooldown accident occurs partially by radiation from the outer surface to the air and reactor cavity cooling system in the cavity between the reactor pressure vessel and surrounding concrete. It is therefore necessary to have a stable, high emissivity on the external surface of the pressure vessel at elevated temperatures. Depending on the emissivity of the selected material, it may be necessary to incorporate a high emissivity coating on the outer surface of the reactor pressure vessel.

Preliminary emissivity screening testing of the potential candidate materials will be performed to determine the detailed experimental program needed for developing a stable surface with minimum emissivity required for adequate cooling of the reactor pressure vessel. Concurrent with that testing, a surface treatment/coatings program will be conducted to investigate the efficacy of various potential concepts for either increasing the emissivity of the reactor pressure vessel materials or providing a coating that will have the required emissivity.

3.4.10 Internals Materials Testing and Qualification (Unfunded in FY-05)

The existing database for candidate alloys will be assembled, analyzed, and evaluated with respect to the design and operating requirements for the reactor internals. Principal topics for review will include high-temperature strength, stability, and long-time performance under irradiation of the materials, effects of impure helium on the mechanical and physical properties of the materials, codification status, prospects, and needs. The status of the joining technology will be reviewed. The weld metal and weldment database will be collected for the candidate alloys. And the technology behind the weld strength factors under development by the ASME and other international codes will be reviewed in collaboration with activities on design methodology. The neutron fluences accumulated in the metallic core internal materials are expected to be low relative to the tolerances of the structural alloys. Nevertheless, these will be reviewed and details developed for confirmatory testing and evaluation. Based on the results of the review, details of the program will be developed to evaluate the mechanical and fracture properties of the leading candidates, along with their environmental and irradiation response.

Joining technology will be developed and experimental work started. Weldments will be produced for mechanical testing, aging studies, and microstructural characterization. Creep-rupture and creep crack growth testing will be started. Environmental testing and creep-fatigue will be performed, and computational models will be used to predict weld microstructures. Microstructural evaluations will be completed on aged materials. Microstructural parameters will be quantified for use in damage prediction models. Weld strength reduction factors will be preliminarily estimated. Candidate weld metals will be ranked for performance. Data will be provided to the design methodology activity to explore the constitutive behavior of weld metal relative to base metal. Weldment test data required for the efforts on design methodology will be produced, and testing of welds will establish confidence in the modeling efforts and the code rules developed from testing and modeling.

3.4.11 Intermediate Heat Exchanger Fabrication Testing (Unfunded in FY-05)

The leading potential candidate alloys will be identified in the course of a detailed assessment. Most likely, these materials will be Alloy 617, Alloy XR, and Alloy X. New alloys, such as CCA617, Alloy 740, and Alloy 230, will be considered as alternates. Assessment will also be undertaken of the potential of C_f/C composites for the compact intermediate heat exchanger. The baseline materials data generation program for the intermediate heat exchanger will focus on characterizing the material of construction as it is influenced by the specific fabrication procedures needed to produce the compact intermediate heat exchanger configuration. The material performance requirements will be developed, and a list of leading candidates will be identified. It will be necessary to decide whether the fabrication processes selected should produce a material of optimum metallurgical condition or whether an off-optimum material condition is satisfactory. At 1000 °C, most of the wrought nickel base alloys require relatively coarse grain size for good creep strength, but fatigue resistance is best for fine grain size.

Exploratory testing will undertake to establish the effect of fabrication variables on the subsequent creep and fatigue properties. Bench testing small models of the intermediate heat exchanger will be performed to add confidence to life prediction methodologies. Manufacturing issues relating to the compact counter-flow intermediate heat exchanger will be addressed as part of the research and testing. It has yet to be demonstrated that such a unit can be manufactured from the leading candidates of high-temperature alloys, so it is clear that the manufacturing of such a unit will produce several issues to be resolved. Issues include production of a high-integrity diffusion bond between the sheets of metal used to build the module, control of conditions that result in an optimum grain size in the metal ligament, development of methods for NDE of the unit, and design and fabrication of joints between the unit and the inlet and outlet piping systems. A review will be undertaken of German and Japanese experience with materials in “more conventional” intermediate heat exchanger units for gas-cooled reactors.

3.4.12 Hot Duct Liner and Insulations Test (Unfunded in FY-05)

Data on the performance of fibrous insulation are needed to ensure that the selected materials are capable of lasting for the life of the plant. The data include physical properties (heat resistance, heat conductivity, and heat capacity), long-term thermal and compositional stability, mechanical strength at temperature, resistance to pressure drop, vibrations and acoustic loads, radiation resistance, corrosion resistance to moisture and air-helium mixtures, stability to dust release and gas release, thermal creep, and manufacturing tolerances and mounting characteristics. Acquisition of these data requires testing of insulation specimens or small assemblies of thermal insulation panels and application of appropriate ASTM standards. This standards development work will be supported within this program. Moreover, application of current nondestructive evaluation techniques, especially in support of the monolithic insulators, is included within this test plan. Specific test rigs and facility requirements include helium flow, vibration, and acoustic test equipment as well as an irradiation facility and hot cell. The testing of prototype assemblies will not include neutron irradiation.

3.4.13 Power Conversion Equipment (Turbine, Generator, Recuperator, etc.) Materials Testing and Qualification (Unfunded in FY-05)

The design information needed to plan an R&D program in this area is insufficient at this time. Also, it is expected that the industrial participants in the NGNP program will do work in this area, and the DOE laboratories will address only selected topics in support of the industrial work.

3.4.14 Valves, Bearings, and Seals Qualification Testing (Unfunded in FY-05)

The design information needed to plan an R&D program in this area is insufficient at this time. Also, it is expected that the industrial participants in the NGNP program will do work in this area, and the DOE laboratories will address only selected topics in support of the industrial work.

3.5 International Collaborations

The Gen IV International Forum (GIF) is the primary mechanism for international collaboration for materials R&D activities in support of the VHTR. The GIF is an international effort to advance nuclear energy to meet future energy needs of ten countries—Argentina, Brazil, Canada, France, Japan, the Republic of Korea, the Republic of South Africa, Switzerland, the United Kingdom, and the United States—and the European Union.

The primary mechanism for collaboration of materials R&D for the NGNP is through the GIF VHTR Materials and Components Project Management Board (PMB). This board is currently composed of members from France, Switzerland, Japan, Korea, South Africa, the United Kingdom, the United States, and the European Union. It meets on a nominal quarterly basis in various locations in the world. The board will be addressing each materials R&D program area noted in this document and will develop detailed collaboration plans for each of these areas. The plans are being developed in about the same order of priority as noted in Section 3.4. It is currently envisioned that this process will not be fully developed and implemented until the end of 2006; however, as each plan is developed, implementation of collaboration activities will begin immediately. Currently, the collaboration plan for nuclear graphite R&D is being developed and should be available by April 2005. This will allow further discussion and development of this plan at the next Materials and Components PMB meeting at ORNL scheduled at that time.

It is currently envisioned that collaboration will involve the establishment of coordinated test and irradiation programs, coordinated purchase of testing materials, coordinated use of special testing facilities, coordinated support for establishment of an integrated Generation IV materials database, and coordinated support of codes and standards committees. It is expected that these collaboration activities will result in a spirit of cooperation between the participating countries, the acceleration of design and licensing activities of VHTR systems, and the reduction of the cost for the NGNP materials R&D.

3.6 NGNP Materials R&D Program Schedule Estimates

Figures 3-13 and 3-14 present the summary and detailed schedules for the NGNP materials R&D over the next 10 years.

Figure 3-13. NGNP Material R&D Program summary schedule.

ID	Task Name	Duration	Start	Finish
1	Graphite	84 mons	10/4/04	12/30/11
2	Select Graphite	4 mons	10/4/04	2/4/05
3	Procure Graphite	3 mons	2/7/05	5/11/05
4	Design ATR test	12 mons	10/4/04	10/14/05
5	Irradiation Creep Test in ATR	72 mons	10/17/05	12/30/11
6	Model Development	24 mons	2/7/05	3/2/07
7	Graphite Design Methodology	24 mons	3/5/07	3/27/09
8	Post Irradiation Evaluation	24 mons	10/4/04	10/27/06
9	High Temp Irradiation Tests	60 mons	10/30/06	12/30/11
10	GIF Irradiation Review	26 mons	10/4/05	12/31/07
11				
12	High Temp Design Methods	119.2 mons	10/4/04	1/13/15
13	Data Assembly	12 mons	10/4/04	10/14/05
14	Testing of Alloy 617	96 mons	10/4/04	1/11/13
15	Testing of Alloy 617 welds	95 mons	10/4/04	12/12/12
16	Develop Methods	36 mons	10/4/05	11/10/08
17	Develop Simplified Methods	36 mons	10/4/05	11/10/08
18	Testing Other Alloys	96 mons	10/4/06	1/13/15
19	Testing of Other Alloy Welds	95 mons	10/4/06	12/12/14
20				
21	Codes & Standards	84 mons	10/4/04	12/30/11
22	ASME High Temp Metals	84 mons	10/4/04	12/30/11
23	ASME Graphite	84 mons	10/4/04	12/30/11
24	Air Oxidation	24 mons	10/4/05	10/29/07
25	BET Surface Method	24 mons	10/4/05	10/29/07
26	ASTM SiC/SiC	84 mons	10/4/04	12/30/11
27	ASTM Fracture Toughness	24 mons	10/4/04	10/27/06
28				
29	Environ Test & Aging	84 mons	10/4/04	12/30/11
30	Evaluate existing data	9 mons	10/4/04	7/13/05
31	Refurbish creep-fatigue equipment	9 mons	10/4/04	7/13/05
32	High Velocity Helium Testing	36 mons	10/4/04	11/9/07
33	Low Velocity Helium Testing	36 mons	10/4/04	11/9/07
34	High Temp Long Term Aging Tests	84 mons	10/4/04	12/30/11
35	Long Term Creep Testing in Helium	84 mons	10/4/04	12/30/11
36				
37	RPV Materials Irradiation Test & Qua	84 mons	10/4/05	12/31/12
38	Select facility	12 mons	10/4/05	10/16/06
39	Fabricate Testing Hardware	24 mons	10/4/05	10/29/07
40	Irradiation of RPV materials	60 mons	10/30/07	12/31/12
41				
42	Composites	72 mons	10/4/04	12/17/10
43	Helium Environment on SiC/SiC	12 mons	10/4/04	10/14/05
44	Tube Geometry Tests	12 mons	10/4/04	10/14/05
45	SiC/SiC Control Rod Tests	60 mons	10/17/05	12/17/10
46	C/C Control Rod Testing	36 mons	10/4/04	11/9/07
47	C/C Other Structures Testing	36 mons	11/12/07	12/17/10
48				
49	Database & Handbook	72 mons	10/4/05	12/19/11
50	Other Components	60 mons	10/4/05	12/6/10
51	RPV Transport & Fab	72 mons	10/4/04	12/17/10
52	RPV Emmissivity	24 mons	10/4/05	10/29/07
53	Metallic Core Internals	24 mons	10/4/05	10/29/07
54	Hot Duct & Insulation	60 mons	10/4/07	12/5/12
55	IHX & Piping	72 mons	10/6/08	12/19/14
56	IHX Pressure Vessel	72 mons	10/6/08	12/19/14
57	Administration	120 mons	10/4/04	2/6/15

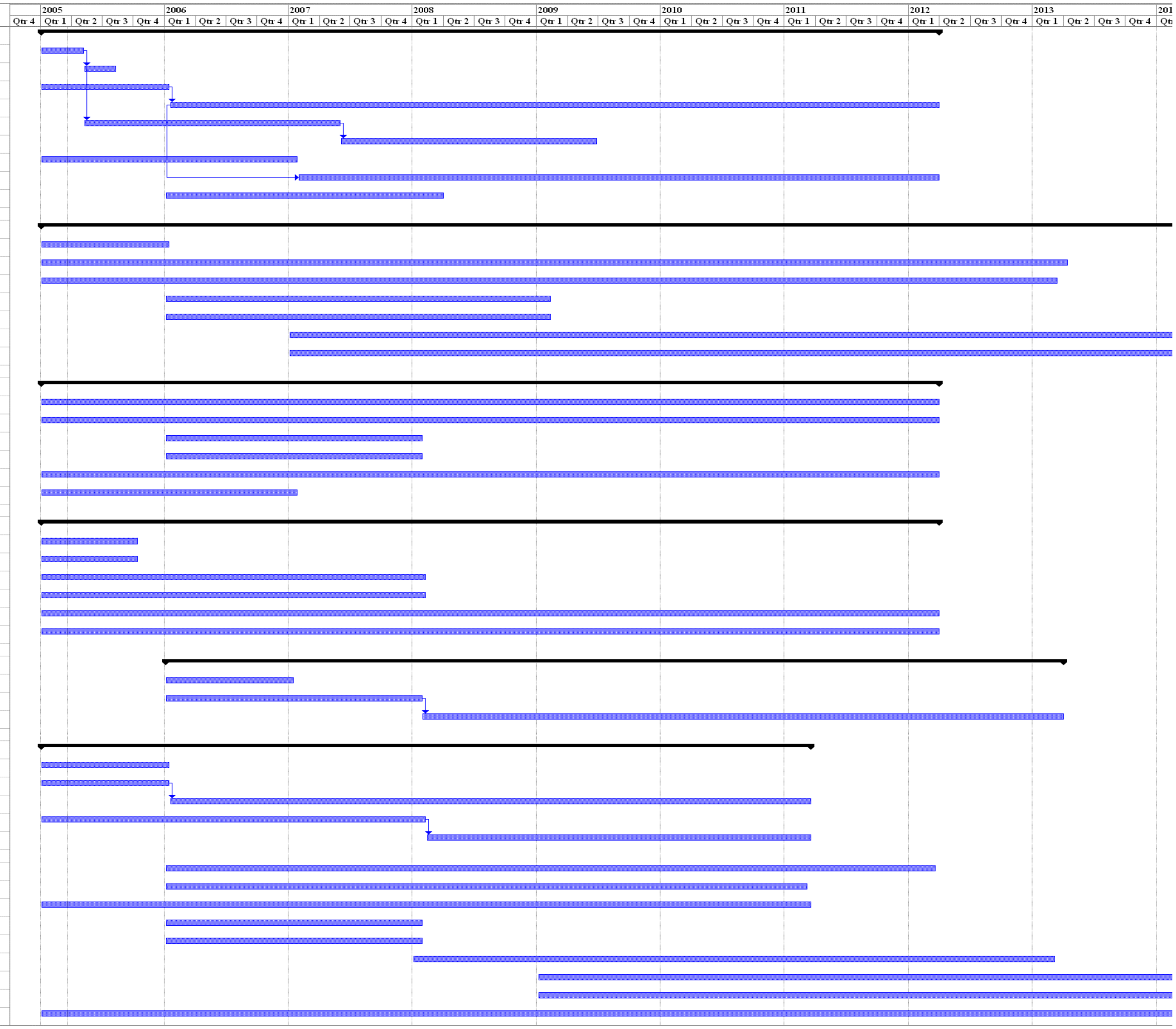


Figure 3-14. NGNP Material R&D Program detailed schedule.

4. NGNP DESIGN METHODS DEVELOPMENT & VALIDATION

This section outlines an ongoing, NGNP Design Methods Development & Validation Program and an implementation strategy designed to govern the selection, validation, and use of the software analysis tools and supporting input data required to calculate the behavior of the NGNP system during all normal and off-normal scenarios. The software tools discussed here include those necessary to calculate the neutronic behavior, the thermal-hydraulic behavior, the interactions between neutronics and thermal-hydraulics, and the structural behavior where necessary. The fuel performance and fission product transport modeling efforts are discussed in Section 2 of this document. The material in this section is based on the material in the document: *Next Generation Nuclear Plant – Design Methods Development and Validation research and Development Program Plan* [Schultz et al. 2004]. The NGNP Methods Program is designed to be interactive across the appropriate parts of the DOE complex as well as with other university and industrial nuclear community stakeholders and will include their feedback through a peer-review process.

Following selection of the NGNP pre-conceptual design concept, the design will undergo a series of three evolutions (conceptual, preliminary, and final design). The NGNP Project software must be capable of analyzing the NGNP design behavior for each of the latter three design stages. The plans outlined herein are designed to achieve this objective.

The Design Methods Development and Validation R&D implementation methodology is shown in Figure 41.

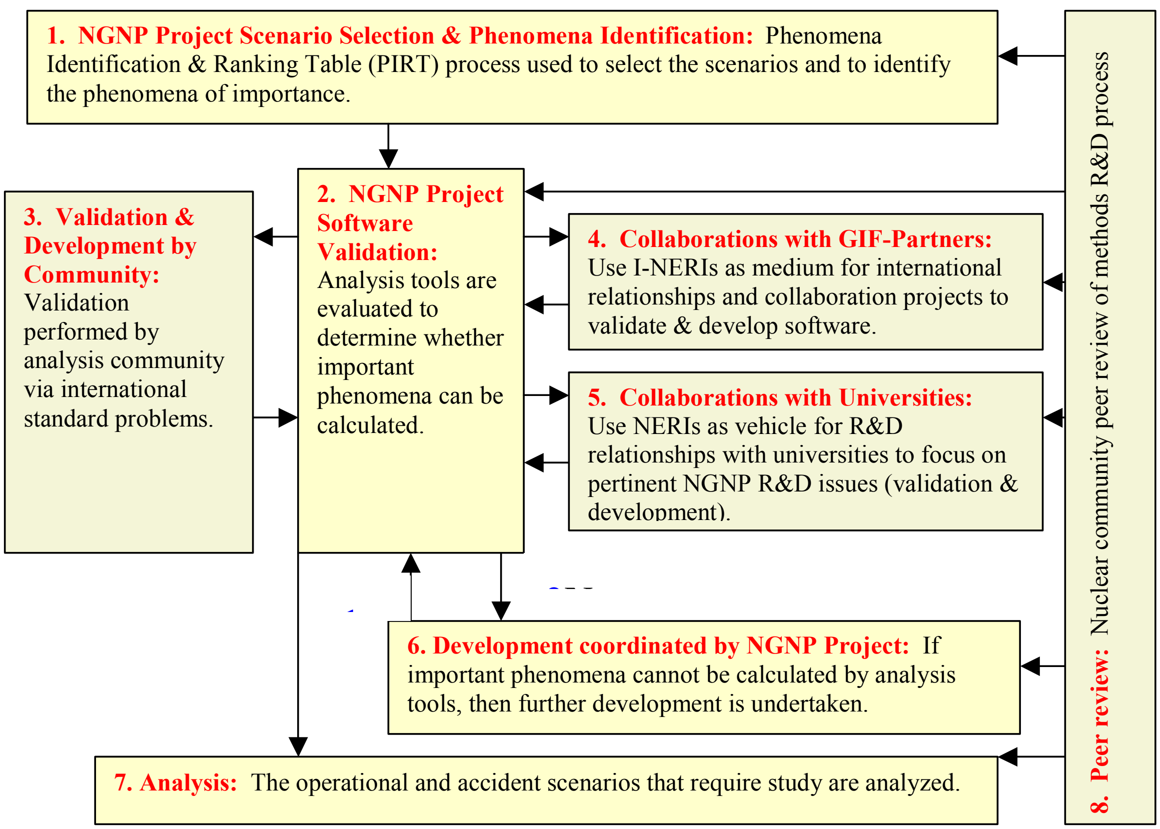


Figure 4-1. Methods R&D process.

The Design Methods Development and Validation R&D implementation methodology consists of eight interacting activities:

1. Selection of the most challenging scenarios together with the dominant phenomena in each,
2. Internal validation of the software tools and data required to calculate the NGNP behavior in each scenario,
3. External validation of the software tools via non-NGNP Project nuclear engineering community participation in international standard problems,
4. R&D performed through GIF-member & NGNP Project collaborations centered in International Nuclear Engineering Research Initiatives,
5. R&D performed through university & NGNP Project collaborations centered in Nuclear Engineering Research Initiatives or GIF Project Management Board agreements,
6. Software development, when validation findings show that certain models are inadequate,
7. Analysis of the operational and accident scenarios, and finally,
8. Review of the global process, and the process ingredients, using experts outside the program.

The ultimate objective of this effort is to ensure the software tools and data used to analyze the desired NGNP behavior in Activity 7 of Figure 4-1 are capable of meeting the NGNP analysis requirements to achieve the necessary objectives. The analysis requirements are defined by identifying the important phenomena and processes, using the Phenomena Identification & Ranking Tables (PIRT) process [Boyack et al 1990], for each of the challenging scenarios that require analysis.

The analysis requirements can only be achieved by using a spectrum of software tools and associated data libraries. The calculational process that satisfies the analysis requirements identified above is broken into seven steps, as shown in Figure 4-2. The seven steps are summarized in paragraphs a through g below. Figure 4-3 identifies the software currently associated with each of the steps in Figure 4-2.

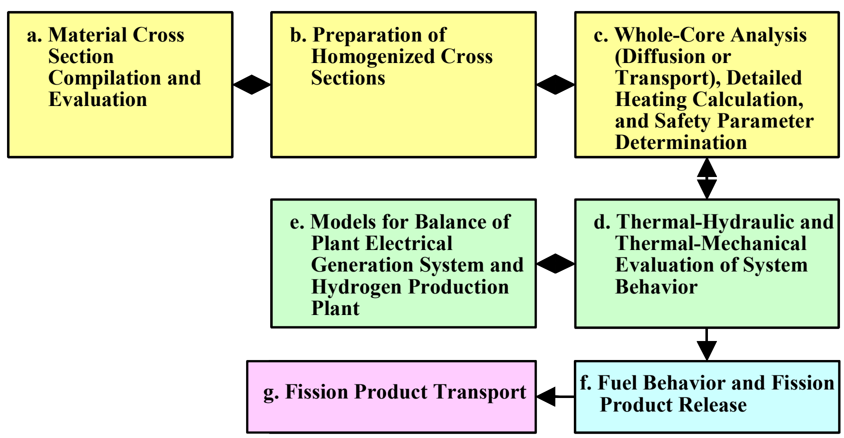


Figure 4-2. Calculation process.

a. ***Material cross section compilation and evaluation.*** Nuclear interaction cross sections are among the most basic fundamental engineering data required for design, licensing, and operation of nuclear systems. Compared to current light-water reactors, any of the proposed NGNP configurations will feature a somewhat harder neutron spectrum, a more complex fuel form, and two to three times greater burnup. Studies show that there is a near-term need for improved cross section measurements in certain neutron energy ranges for some isotopes to support the extensive computational modeling that will be required for the NGNP design, regardless of the specific basic reactor configuration that is ultimately selected. The isotopes ^{240}Pu, ^{241}Pu, and ^{242}Pu are particularly important at high burnup. Improved cross section data are ultimately incorporated into the Evaluated Nuclear Data Files (ENDF) maintained by the US National Nuclear Data Center. These data are subsequently processed to produce input libraries useful in reactor analysis software.

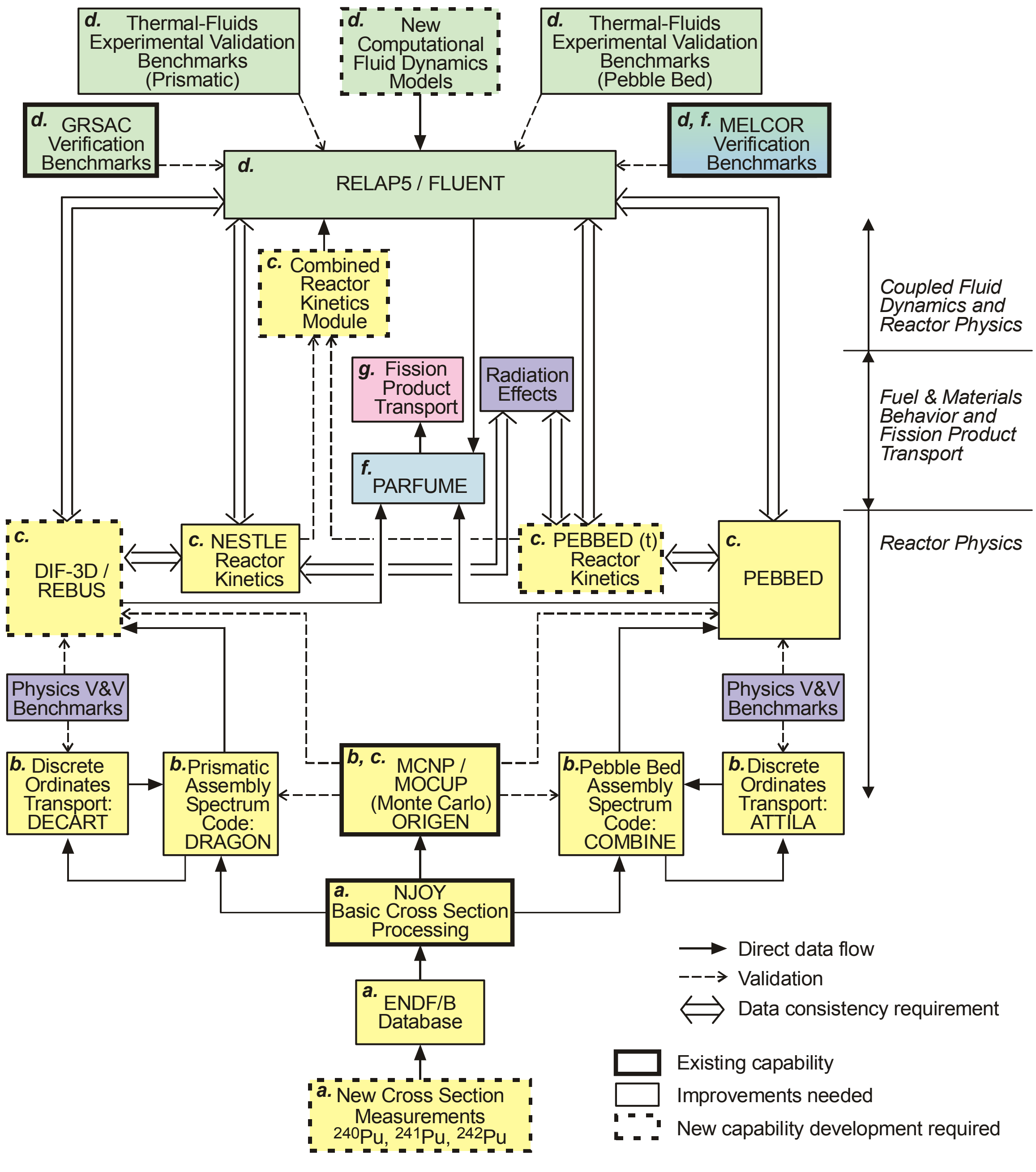


Figure 4-3. Application of process to block-type and pebble-bed candidate designs for NGNP—with applicable software.

b. Preparation of homogenized cross-sections. Before it can be used for a specific reactor application, the ENDF data, as processed into a general format by NJOY or a similar tool, must be further processed into a case-specific form using local cell and assembly modeling codes. The basic physical data are processed for case-specific resonance shielding and then weighted with characteristic energy and spatial flux profiles generated from unit cell or super-cell models. This step is performed using software that approximates the neutron transport equation using P_N or B_N transport codes for the energy flux calculation and a one- or two-dimensional transport code for the spatial flux. [In the advanced lattice codes, spatial resolution is typically done using integral transport methods (collision probability or method of characteristics approaches.)] Software that will be initially evaluated for this function includes COMBINE, BONAMI/NITAWL, MICROX-2, WIMS-8, HELIOS, and DRAGON. An appropriate suite of codes will be implemented and validated according to accepted standards. The geometric aspects of this process are significantly different in the prismatic and pebble-bed concepts, so two computational paths are shown in Figure 4-3. For additional assurance that the computational results obtained using diffusion theory codes are accurate, higher order deterministic transport methods should be employed to perform selective benchmark checks. Representative software that might serve this function is ATTILA, TWODANT, THREEDANT, or DORT/TORT. These transport packages are also used as part of the assembly cross-section preparation process.

c. ***Whole-core analysis (diffusion or transport), detailed heating calculations, and safety parameter determination.*** Nodal diffusion-theory codes, such as DIF3D and an INL-developed code, PEBBED, which is designed specifically for pebble-bed reactor simulation, will be the centerpiece production codes to perform NGNP reactor core analysis. Steady-state eigenvalues, energy and spatial flux profiles, reaction rates, reactivity changes (burnup and control rod movement), etc., will be calculated with the nodal diffusion-theory codes. Multi-group cross section data generated in the reactor assembly cross section preparation step (Step b above) will be provided to the nodal diffusion code. The DIF3D code also contains a nodal transport option (VARIANT) based on the variational transport approach. To consider the power behavior as a function of fuel depletion, additional capabilities are required. This function is usually performed by the REBUS code in conjunction with DIF3D, whereas it is internal to the PEBBED code for the pebble bed reactor case. All of these software packages will be verified against alternate computational models, especially models based on the well known MCNP stochastic simulation (Monte Carlo) code as shown in the center of Figure 4-3, and various deterministic approaches. In addition, all of the reactor physics models will be validated against various suitable experimental benchmarks. A preliminary assessment of appropriate validation benchmarks pertinent to the current gas-cooled NGNP reactor concepts has in fact been completed by INL and ANL and more detailed benchmark evaluations are now underway. Output from the nodal diffusion codes will not only provide the steady-state operational physics parameters for each operational analysis conducted, but it will also be used as the initial condition for reactor kinetics calculations required as part of the overall system analyses performed in Steps d and e below. Spatial changes in flux and power level as functions of time during postulated transients, predicted by the kinetics module, will provide the energy source term required for the overall thermal-hydraulics systems code computations at each time step during each transient. This process permits full coupling of thermal and neutronics computations, consistent with modern practice for nuclear systems analysis. The NESTLE code, a subroutine in the RELAP5-3D systems analysis thermal-hydraulics code, will serve this purpose for the prismatic reactor concept, and a time-dependent implementation of the PEBBED code will be used for the pebble-bed concept.

d. ***Thermal-hydraulic and thermal-mechanical evaluations of system behavior.*** The fluid behavior, and interactions with the neutronics, will be calculated using a systems analysis code, or perhaps a coupled systems analysis/computational fluid dynamics (CFD) code. Examples of two systems analysis codes and a CFD code are RELAP5-3D, GRSAC, and Fluent. In such a coupling, systems analysis software is used to perform calculations of the overall system behavior considering the interactions between all the parts, e.g., the core, the plenums, the hot exit duct, the turbine, and the remainder of the plant. CFD codes, such as Fluent, are used to calculate the detailed three-dimensional fluid behavior in a region of the reactor such as a plenum. In some cases, where one code has been validated extensively, it can be used for limited validation of a second code. An example of this is shown in Figure 4-3 where GRSAC may be used to partially validate RELAP5-3D. In addition to analyzing the fluid behavior under a spectrum of operating and accident conditions, the thermal-hydraulic tools also will be used to investigate the significance of material geometric tolerance variations due to manufacturing, thermal responses, and irradiation effects such as graphite swelling. The need to examine factors that affect thermal-mechanical influence on fluid and heat transfer behavior will be included in the tool selection and evaluation process.

e. ***Models for balance of plant electrical generation system and hydrogen production plant.*** The behavior of the balance-of-plant systems will be modeled using a systems analysis code such as RELAP5-3D or Aspen. The balance-of-plant models are important to include in the analysis process to account for the important interactions that affect the system efficiency during normal operational conditions, but also to account for the equipment interactions that may lead to undesirable conditions such as turbine over-speed, loss of net positive suction head for auxiliary systems, or oscillatory conditions that may lead to equipment damage. Interactions between the reactor system and its balance-of-plant components lead to boundary conditions that will determine whether fuel-damaging conditions are likely (see item f).

f. ***Fuel behavior and fission product release.*** The performance of fuel particles under irradiation is modeled to determine whether fuel failure will occur, with the subsequent release of fission products, and whether subsequent migration of fission products throughout the system must be considered. The INL software designed to perform this function is called PARFUME. In addition to the physical description of the fuel, an operation history generated by physics and thermal analysis codes (consisting of fuel temperature, burnup and fast neutron fluence) is used as input to PARFUME. The code models the mechanical and physico-chemical behavior of the fuel and calculates the fraction of the fuel particle inventory that may fail. Several potential failure mechanisms are analyzed, including cracking of structural particle layers, debonding of the inner pyrolytic carbon layer from the silicon carbide (SiC) layer, buildup of internal fission gas pressure, kernel migration (amoeba effect) to the SiC layer, and thinning of the SiC layer by fission product interactions. PARFUME also calculates the fraction of selected fission product gases released from failed particles and from fission of uranium contamination in the matrix material surrounding the fuel particles. Calculation of the release of selected metallic fission products is currently under development. The fuel and fission product modeling activities are described in Section 2 of this document.

g. ***Fission product transport.*** If a loss-of-coolant accident has occurred, such that the fission products may migrate or be impelled into the confinement/containment building with perhaps subsequent release to the environment, then the final calculational step is the prediction of the fission product movement into the environment and its environmental distribution. Software tools that may be used for this purpose include MELCOR, RELAP5-3D/SCDAP in conjunction with VICTORIA, and perhaps a CFD code with an appropriate user defined function.

The process described in items “a” through “g” is shown in the flow chart of Figure 4-2. The complete calculation process illustrated in Figure 4-2 is only exercised in its entirety for a few scenarios. Most scenarios would require the use of only a fraction of the calculations represented in Stages a through e. For example, scenarios that do not include a loss of coolant, i.e., a pipe break, usually would not require calculation of fission gas transport (Stage g). In addition, if the neutronics has been thoroughly calculated for the reactor system operating condition (Stages a through c), then a multitude of reactor system calculations can be performed using the evaluated reactor power state at time zero, and hence the Stage a through c calculations may only need to be performed once for a desired operating condition. Thereafter, for such scenarios that assume reactor scram (requiring no reactor kinetics), a multitude of calculations can be performed using only the software tools developed for Stages d and e.

A rigorous PIRT analysis of the NGNP has not been performed since the design has not yet been identified. However, based on the accumulated knowledge of the advanced gas-cooled reactor vendor community, a “first-cut” PIRT has been defined and used to specify the FY-05 R&D and also to formulate the fundamental R&D progression for subsequent years. Once the design is specified, the design methods development and validation R&D requirements will be aligned with the design.

The “first-cut” PIRT used to specify the FY-05 R&D and to formulate the fundamental R&D progression for subsequent years is shown in Table 4-1. The following discussion briefly outlines R&D needs, as listed in Table

4-1, for normal operational conditions as well as for accident conditions that are anticipated to be most challenging in that they place the greatest requirements and limits on system design and operation such as the depressurized conduction cooldown (DCC) and pressurized conduction cooldown (PCC) scenarios. The following phenomena have been determined to be important: neutronics behavior, core hot channel characterization, bypass analysis, mixing, laminar-turbulent transition flow and forced-natural mixed convection flow, air-water ingress, and fission product transport.

Table 4-1. "First-Cut" PIRT for normal operation, PCC and DCC scenarios.

Scenario	Inlet Plenum	Core	RCCS	Outlet Plenum
Normal operation		i. Neutronic behavior ii. Bypass flow iii. Hot channel characteristics		i. Mixing
DCC		i. Thermal radiation and conduction of heat across the core ii. Axial heat conduction and radiation iii. Natural circulation in the reactor pressure vessel iv. Air & water ingress v. Potential fission product transport	i. Laminar-turbulent transition flow ii. Forced-natural mixed convection flow	
PCC	Mixing	i. Neutronic behavior ii. Bypass iii. Laminar-turbulent transition flow iv. Forced-natural mixed convection flow v. Hot channel characteristics at operational conditions	i. Laminar-turbulent transition flow ii. Forced-natural mixed convection flow	i. Mixing

Neutronic behavior. As noted previously, the current NGNP design candidates have somewhat different neutronic properties than standard light-water reactors, specifically, a higher thermal flux component with the peak shifted to higher energy, a more complex fuel geometry, and a fuel cycle with two to three times the burnup. Some of the global reactor physics issues associated with the current VHTR core designs include:

- Strong thermal flux and power peaking at the reflectors, especially the inner reflector,
- Proper accountability for double heterogeneity of the fuel,
- Complexity of the 3-dimensional core depletion and reload analyses, and
- Accurate coupled thermal-hydraulic/neutronics/thermo-mechanical kinetics with appropriate representation of irradiation effects.

Furthermore, there are some additional, more fundamental physics issues. At the very high burnups expected for the NGNP, the higher isotopes of plutonium contribute a significant amount of fission energy and resonance capture, both of which affect the basic operating characteristics. Yet it is generally acknowledged that the necessary cross-section information for these isotopes requires improvement. In addition to improvements in the cross-section data to increase the accuracy of the neutronics calculations, improvements in cross-section processing methods are needed in the treatment of resonances in the thermal energy range in graphite-moderated reactors where up scattering is significant. The inability to account properly for this effect could lead to substantial errors in the harder spectrum of a graphite-moderated reactor. The double heterogeneity noted above is another fundamental physics aspect requiring attention. The two scales of heterogeneity involved here are the fine scale, associated with the fuel particles, and the coarse scale, which must properly model the fuel-pebbles or fuel-compacts. The improvement in cross section generation will require valid treatment of resonance interactions.

Core hot channel characterization. The characteristics of the hottest cooling channels at operational conditions are considered a key calculational need since the hot channel temperature distribution defines the hottest initial condition for the fuel and surrounding materials. Hence preliminary computational fluid dynamics (CFD) studies have been initiated and validation data are sought.

Bypass. The bypass flow passes through the reflector regions in both pebble-bed and block reactors and, in a block-type reactor, between the blocks. Because the quantity of bypass flow is a direct function of the bypass area, which in turn is a function of the temperature distribution, fluence, and graphite properties, the influence of the bypass on the core temperature distribution may be significant.

The influence of bypass may be assessed in part by performing a series of parametric calculations that differ in the geometric boundary conditions—as defined by the various factors that influence the bypass flow passages such as manufacturing tolerances, misalignments, and geometric distortions.

Mixing. Mixing refers to the degree to which coolant of differing temperatures entering a region mixes to produce a uniform temperature. Mixing is a three-dimensional phenomenon in the inlet and outlet plenums and a function of a number of variables. In the inlet plenum, where it is identified as important in the PCC scenario, mixing occurs during natural convection as helium moves upward through the hottest portion of the core while cooler helium moves downward through the bypass and the cooler regions of the core. In the outlet plenum, mixing occurs between the bottom of the core and the turbine or immediate heat exchanger inlet during normal operation. A preliminary calculation of the temperature variation in the lower plenum indicates that gas temperature variations could exceed 300 °C. Although the specification for temperature variation at the immediate heat exchanger or turbine inlet has not been set, it is thought that the helium temperature variation must be less than ±20 °C. Also, it has been seen that helium has a surprising resistance to thorough mixing [Ball 2004, based on experience of Kunitoni et al. 1986] and that the temperature in the core outlet jet can vary over a considerable range, particularly since the bypass flow may vary between 10% and 25%. Therefore, it is likely that special design features will be required to ensure good mixing and minimal thermal streaking from the lower plenum to the turbine inlet.

Laminar-Turbulent Transition Flow &. Forced-Natural Mixed Convection Flow. During the PCC scenario in the core region and during both the PCC and DCC scenarios in the reactor cavity cooling system (RCCS), there is the potential for having convective cooling in the transition region. Because the convective cooling contribution is an important ingredient in describing the total heat transfer from the core and thus the ultimate peak core and vessel temperatures, these heat transfer phenomena are potentially important.

Air & Water Ingress. For loss-of-coolant scenarios, such as the DCC, there is the potential for air and water ingress into the core in perhaps harmful quantities—depending on the scenario assumptions. Air may be present in the reactor cavity (some designs have a cavity filled with inert gas) and may enter the core by diffusion in a DCC accident. Water is normally present in the air in the form of humidity, but it may enter the core in much greater quantities, with much greater potential effect on reactivity, if the shutdown cooling system suffers a pipe leak or break. Oxidation of graphite in the prismatic core design is also a potential safety issue.

Fission Product Transport. Fission product transport must be calculated for cases where some fraction of the TRISO fuel particles fail prior to or in conjunction with the DCC scenario and because certain fission products such as silver and palladium may diffuse through the TRISO coatings. Dust that will likely contain fission products must also be tracked and accounted for using state-of-the-art calculational tools, particularly for the pebble-bed reactor.

The remainder of this chapter describes the Methods Development and Validation R&D is four basic areas: (1) Nuclear data measurements, integral evaluations, and sensitivity studies, (2) Reactor kinetics and neutronics analysis development, (3) Thermal-hydraulics and (4) methods development and analyses of the molten salt cooled NGNP.

4.1 Nuclear Data Measurements, Integral Evaluations, and Sensitivity Studies

Accurate differential nuclear data libraries and well characterized and accurate integral benchmark information is required for all computational reactor physics tasks associated with NGNP design and operation. Differential nuclear cross section data for all materials used in the reactor are required as input to the physics codes. Furthermore, integral benchmark experiment data for relevant existing critical configurations are required for physics code validation. Finally, rigorous sensitivity studies for representative NGNP core designs are required for prioritizing data needs and for guiding new experimental work in both the differential and integral regimes. The following sections describe the near- and long-term needs and planned activities in these three areas.

4.1.1 Sensitivity Studies

A study will be conducted early in the NGNP R&D effort to quantify uncertainties in computed core physics parameters that result from propagation of uncertainties in the underlying nuclear data used in the various modeling codes. ANL has developed expertise in this area and will have the lead in this effort, which will serve as an aid in further quantifying the need for additional cross-section measurements and/or evaluations for NGNP and as a guide in planning of future measurements and evaluations. INL will collaborate with ANL in the effort at a level that will increase over time during the first three years. The required evaluations will be accomplished by performing formal sensitivity and uncertainty analysis in order to identify the nuclides that contribute to calculational uncertainties and to quantify the propagated uncertainties in the context of the currently anticipated NGNP core designs. The NGNP gas-cooled prismatic core design will be the basis for this initial study [MacDonald et al. 2003]. Subsequent studies will encompass the other candidate concepts. Sensitivity coefficients will be calculated by generalized perturbation theory codes and folded with multi-group covariance data to derive propagated uncertainties in computed integral reactor parameters arising from the nuclear data. Key integral parameters to be evaluated include reactivity, peak power, reaction rate ratios, nuclide inventory, feedback coefficients, etc. The impact of cross section data uncertainty on the accuracy of each parameter will be evaluated, along with the identification of nuclides, cross section types, and energy ranges that have greatest impacts on accuracy of integral parameters. Most of the effort will be conducted during the first three years, FY-05 through FY-07, although the required computational capabilities will be maintained by ANL and INL over the life of the overall NGNP project for use as needed to address additional issues that will undoubtedly surface from time to time.

4.1.2 Integral Neutronic Parameter Evaluations and Assessment of Needs for New Measurements

The computer codes used in NGNP design and safety analyses must be able to model the NGNP configuration accurately. Therefore, these codes must be benchmarked against appropriate experimental data. Various experimental data on the physics of high-temperature gas-cooled reactors (HTGRs) have been measured internationally since the early 1960s. During FY-04, under DOE Generation IV crosscut funding, the INL and ANL studied all the known experimental and prototypical HTGRs and relevant critical facilities in order to assess their potential to be used as benchmarks. For the pebble-bed NGNP concept, the ASTRA, AVR, CESAR II, GROG, HTR-10, HTR-PROTEUS, KAHTER, SAR, and THTR facilities were assessed. For the prismatic NGNP concept, DRAGON, Fort St. Vrain, Gulf General Atomic (GGA) criticals, HITREX-1, HTLTR, HTTR, MARIUS IV, the Peach Bottom Reactor and criticals, SHE, U.K. NESTOR and HECTOR lattices, and VHTRC were assessed.

As a result of the assessment, the HTR-10 (Figure 4-4) was chosen as the most promising facility for the first pebble bed reactor benchmark that will undergo full evaluation, and HTTR and VHTRC were chosen as the most promising facilities for the first block-reactor benchmarks. Recent NGNP program strategic developments have also resulted in the need for an additional assessment exercise to include any available molten salt cooled concepts that may be suitable as benchmarks. Current plans are to conduct this assessment, as a collaborative effort of INL, ANL, and ORNL in FY-06 or FY-07.

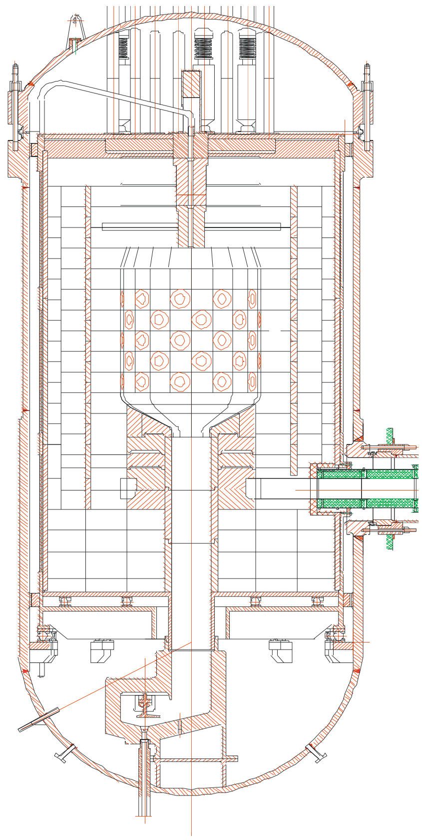


Figure 4-4. Schematic diagram of HTR-10 core and vessel.

The next steps in connection with the assessment study that has been completed for the gas-cooled concepts will involve detailed evaluation and documentation of the identified facilities to provide benchmark specifications accepted by the community and regulators for validation of physics modeling codes. The work will be conducted under the International Reactor Physics Evaluation Project (IRPhEP), an international effort endorsed by the Organization of Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Nuclear Science Committee (NSC) in June of 2003. The INL and ANL will contribute NGNP-specific benchmarks evaluated under this R&D Plan. Data contributed to the IRPhEP will be published in an OECD Handbook to be made available to all participating countries. Because of the rigorous quality standards in the evaluation process, published IRPhEP benchmarks will have the highest level of international credibility and acceptance.

The INL provides leadership for the IRPhEP Technical Review Group that was organized during FY-2004 / FY-2005, maintains the infrastructure of the IRPhEP, and is responsible for compiling and distributing its annual publications. Through this effort, the IRPhEP will be able to (1) consolidate and preserve the information base that already exists worldwide, (2) retrieve data that is not readily available, (3) identify areas where more data are needed, (4) draw upon the resources of the international reactor physics community to help fill those needs, (5) identify discrepancies between calculations and experiments caused by deficiencies in cross section data, cross section processing codes, and neutronics codes, (6) eliminate a large portion of the tedious and redundant research and processing of reactor physics experiment data, and (7) improve experimental planning, execution and reporting.

The formal benchmark evaluation process is quite rigorous and includes very specific steps. Each draft experiment evaluation undergoes thorough internal review by the evaluator's organization. In addition, each experiment undergoes independent peer review by another IRPhEP Technical Review Group member at a different facility. Starting with the evaluator's submittal in the appropriate format, independent peer reviewers verify (i) that the benchmark specification can be derived from the descriptive information given in the evaluation, (ii) the completeness of the benchmark specification, (iii) the results and conclusions, and (iv) adherence to format. A third review by the assembled IRPhEP Technical Review Group then verifies that the benchmark specification and conclusions are adequately supported.

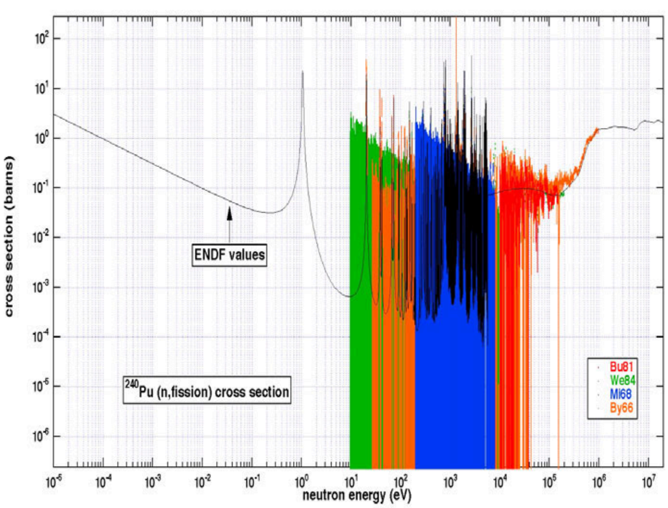
The NGNP integral evaluation activities conducted by the INL under this Plan will be coordinated with the ongoing Generation IV Design and Evaluation Methods Crosscut program to avoid duplication of effort and to maximize funding leverage. There will also be a high degree of international coordination due to the inherent organizational nature of the IRPhEP. For example, the ASTRA facility in Russia is being evaluated by the South African pebble bed reactor development group and their international collaborators. In the first year of this effort, the INL will evaluate the HTR-10 and possibly PROTEUS. ANL will evaluate either HTTR or VHTRC. Other appropriate benchmarks will be evaluated in later stages of the overall integral benchmark effort encompassed by this Plan. It is anticipated that evaluations for most, if not all of the higher priority NGNP-specific gas-cooled facilities identified in the assessment will be completed during the first five years, FY-2005 through FY-2009. Additional evaluations will continue in the later years as specific needs are identified, for example in the case of the molten salt cooled concept if that concept becomes of interest to the industrial partners in the NGNP program.

Where additional experimental measurements are required, attempts will be made to make use of existing facilities worldwide. Conclusions will be made concerning adequacy of available measurements and facilities and will be used as a basis for recommendations relative to the need for new measurements and/or facilities.

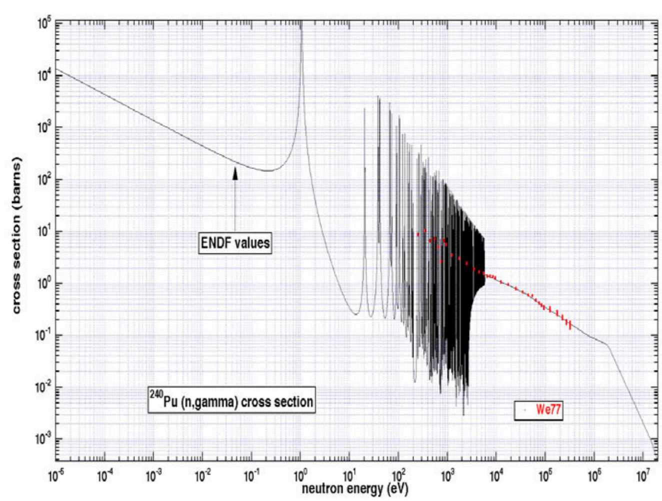
4.1.3 Differential Nuclear Data Assessment and Measurement.

Studies already conducted by INL and others as a part of the NGNP, Generation IV and Advanced Fuel Cycle Initiative (AFCI) programs show that the transuranic nuclides, for which useful cross section data are extremely limited in many cases, will significantly affect the neutronic behavior of some advanced nuclear energy systems of interest. Databases from some key nuclide integral experiment studies [e.g., Mercatali et al. 2004] confirm the sensitivity of computed parameters for minor actinides dominated fast-spectrum system to uncertainties in the cross sections of many of these materials. For the NGNP, the current design will feature a somewhat harder thermal neutron spectrum than is usually found in standard light-water reactors as well as two to three times the burnup of a light water reactor. As a result, improved cross section data might be required in certain neutron energy ranges for some isotopes, in particular ^{240}Pu, ^{241}Pu, and ^{242}Pu. A joint Gen IV/AFCI working group on reactor physics and related nuclear data assessment and improvement needs has been formed to coordinate and prioritize nuclear data activities.

Figure 4-5a shows a plot of the ENDF/B-VI data file values for the ^{240}Pu fission cross-section, the black solid line. Also plotted on 4-5a are the available published direct (differential) measurements over broad energy ranges in the same experiment, shown by the colored vertical lines, with the length of the line as an indicator of the reported uncertainty of the data. Experimental data below ~10 eV are limited to single-point experiments that may not have been done under the same conditions, as discussed later. Thus, in several energy ranges of interest, the ENDF values are heavily based on theoretical models with limited experimental data input, and can be uncertain.



(a) Fission



(b) Capture

Figure 4-5. ENDF/B-VI data file values (black solid line) and available experimental data sets for the ^{240}Pu fission & capture cross sections.

It should also be noted that even where data are available the reported uncertainties are high. Figure 4-5b shows the capture cross section for ^{240}Pu. In this case the experimental data are even more limited and no uncertainties were reported. This capture cross section is particularly important because neutron capture in ^{240}Pu leads to ^{241}Pu, which has large (but also uncertain and in fact difficult to measure) fission and capture cross sections. Recent computations performed at INL show that for a reference prismatic NGNP fuel design, an uncertainty of as little as 10% in this cross section can lead to uncertainties in system reactivity of as much as 500 pcm absolute reactivity because of the propagated uncertainty in the ^{241}Pu buildup. *Initial sensitivity studies conducted early in FY-2005 by ANL in connection with this R&D plan confirm the general direction of this result and indicate that propagation of the estimated uncertainty associated with the ^{240}Pu capture cross section, especially in the low-energy resonance region, is in fact even larger, by perhaps as much as a factor of two, and likely is the most significant contributor to the overall uncertainty of integral reactivity parameters computed for the end of cycle condition.* Furthermore, earlier integral-experiment-based code validation studies performed by INL [Sterbentz 2002, Sterbentz & Wemple 1996] for low-enriched fuel with thermal or slightly hyperthermal neutron spectra representative of typical NGNP designs show that computations of the inventories of the plutonium isotopes of interest here (^{240}Pu, ^{241}Pu, and ^{242}Pu) can vary by as much as 30% from corresponding measurements, at burnups of less than one-third of what is contemplated in a baseline NGNP scenario. Once again such discrepancies can propagate with significant effects on the uncertainty of computed safety-related reactor parameters such as reactivity, Doppler feedback, etc.

A comprehensive standard database, CINDA (Computer Index of Neutron Data), maintained by the National Nuclear Data Program at Brookhaven, was used as the source for experimental data files and references for ^{240}Pu shown in Figure 4-5. In a search of CINDA, 1450 references and data files were found. In these, only one direct measurement of the neutron capture cross section over an extended energy range under self-consistent conditions was found. All other capture cross section information was extracted from ratio measurements relative to other nuclides, based on calculational extractions from total neutron induced reactions on a ^{240}Pu sample, or composed of single-point measurements at one energy or averaged over an energy range to yield a single value. The vast majority of the single-point values were at "thermal" energies, or were integral values.

Thus the roughly 50,000 points in the ENDF data file for the ^{240}Pu capture cross section are the result of one or more nuclear model calculations with what appears to be very limited experimental data as input. There are 18 experimental data files (i.e., there are 18 experimental references in the 1450 CINDA references that represent any experimental measurement), with only one file containing a direct measurement with experimental results over an energy range. The 17 other experimental data files used in compiling the ENDF file are total cross section measurements, ratio measurements, or single-point measurements. As another example, there are 810 references in the CINDA database for the ^{240}Pu fission cross section. Of these, 40 references have experimental data of some form that are used to construct the ENDF evaluated file containing 50546 data points. The four plotted experimental data files represent the only direct, multipoint measurements of the cross section out of the 40 references containing experimental cross section values. The other 36 references of experimental data sets are ratio measurements, single-point measurements, or average values over several broader energy ranges. The single-point values are Maxwellian distributions about some central energy values, generally 0.025 eV. The 810 ^{240}Pu fission cross section references in CINDA also contain experimental data on other parameters associated with fission as well as evaluations, theoretical papers, reports, and other works that do not contain direct data.

The situation for ^{242}Pu is similar to what was described above for ^{240}Pu (although studies do consistently show that the propagation of uncertainty in the case of this isotope is significantly smaller in the NGNP context than for ^{240}Pu). The experimental effort devoted to ^{235}U and ^{239}Pu over the past 50 years has simply not been put into measurements for other actinide isotopes. There is thus also a potential need for new data for essentially all of the approximately 16 heavy actinides that will come into play with the even more advanced Generation IV systems and fuel cycles under study. However, the emphasis in this discussion, specific to the NGNP, is on the plutonium isotopes.

In this portion of the planned NGNP R&D program, the INL, in partnership with ANL and various university and international collaborators, will conduct a research program of measurements for actinides of interest at the

ANL Intense Pulsed Neutron Source (IPNS). The IPNS facility is one of the few accelerator-based neutron sources in the world where it is possible to do relevant neutron-induced measurements. Over the past several years, the INL Nuclear Physics Group has installed the array of detectors shown in Figure 4-6 at IPNS, has implemented the supporting electronics and a data acquisition system based on coincidence techniques developed over the last two decades in nuclear physics, and has been using this array for the study of fundamental aspects of the nuclear fission process. The overall INEEL effort in this area, which includes work at several facilities, has produced over 100 refereed journal papers over the years, and it has established an international collaboration to support the experimental effort through data analysis. In the past year INL has undertaken an effort to upgrade the system in a manner that will allow measurement of absolute nuclear cross sections as well, specifically to support this proposal. The proposed program will be coordinated with, and will complement, related efforts elsewhere, especially the nuclear data measurements under way at the Los Alamos Neutron Science Center (LANSCE) under the AFCI program as well as related efforts in Europe.

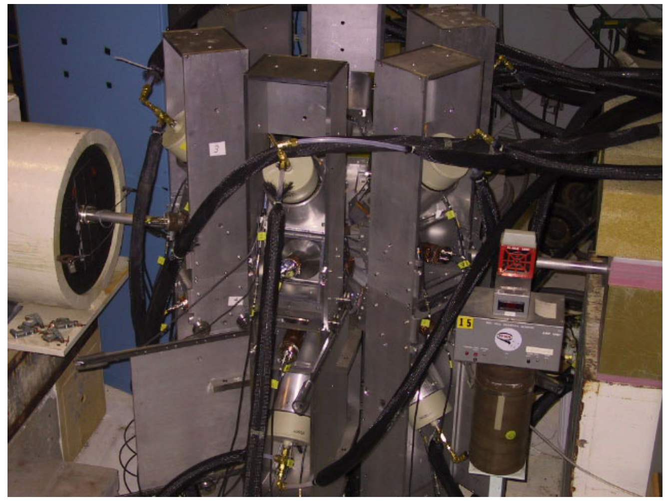


Figure 4-6. INEEL Detector array at ANL/IPNS.

In the first five years, FY-05 through FY-09, the differential data measurement campaign will be focused on development of data for ^{240}Pu and ^{242}Pu, especially in the lower parts of the resonance energy range where the IPNS facility offers good energy resolution coupled with the desirable features of the coincidence based measurement and data analysis techniques that are employed. These advantages include background reduction, model-independence for some interactions, separation of competing interactions, etc. Although the technique offers the greatest advantages for high precision fission measurements (cross section and fission product yield), data for all observable interactions are taken simultaneously. Capture and inelastic scatter interactions may also be determined with the aid of detailed nuclear structure information for the target isotope. This information is quite limited and difficult to obtain in many cases. However, the coincidence methods offer an approach for determining some of the necessary nuclear structure information as an inherent part of part of the overall measurement process itself. Although capture and inelastic measurements with the IPNS approach are not model-independent and the attainable absolute precision of such measurements is likely to be less than is the case for fission, the relative event spectrum data for these interactions, which will be of greater precision than absolute measurements, is of significant value in and of itself. This is because the energy-dependent relative cross section information can be normalized to single point cross section measurements from other experiments to provide a more complete picture of the situation than could be obtained by either type of measurement alone.

The ^{240}Pu and ^{242}Pu measurements will be followed by measurements for ^{241}Pu, assuming that the priorities that are currently recognized by the various nuclear data working groups remain unchanged. After this initial campaign is complete and the data published for inclusion in the ENDF evaluations, attention will turn to the higher actinides with specific priorities set, once again, by the cognizant international nuclear data working groups.

Plans have been made to acquire targets of the needed isotopes from the Federal State Unitary Enterprise State Scientific Center of the Russian Federation Research Institute of Atomic Reactors in Dimitrovgrad, Russia. This is an important aspect of the proposed work, since isotopically pure metallic forms of the targets are not readily available in the United States, although the INL does have a capability for fabrication of some limited types of targets and this expertise will be used where appropriate. It is also important to recognize that a collaboration of university and national laboratory scientists in both the US and Russia exists to analyze and publish the results, once again allowing leveraging of resources.

It should also be noted in closing that the prospective INL/ANL program for nuclear data measurements at IPNS underwent an international scientific peer review in September 2004. As a result of this review, there will be a much stronger emphasis in the first 18-24 months on careful calibration of the instrumentation and proof of principal testing of the experimental protocol using ^{239}Pu standards. This will be followed by measurements for ^{240}Pu with a focus on the low-lying resonances. Furthermore there will also be additional attention over the longer term to improvement of certain design details of the time of flight beam monitor as well as to various experiment design and protocol improvements required to reduce systematic uncertainty in the overall experimental results. These latter efforts will include, 1) additional attention to corrections for neutron scattering in the sample targets as well as corrections for other detector artifacts that can be addressed by careful modeling and, 2) continuous online monitoring of the system over the long run times needed in order to ensure self-consistency of the final data sets. The measurements will be done for the "overlap" between the measurement priorities that the sensitivity studies discussed in Section 4.1.1 above are defining and the specific capabilities of the measurement technique. A joint AFCI-Generation IV physics group has been formed by DOE-NE and it will be the appropriate forum to discuss and determine this overlap.

4.2 Reactor Kinetics and Neutronics Analysis Development

The design and operational analyses of the NGNP requires the ability to carry out the following reactor physics computations: (1) cross section preparation and fuel assembly spectrum calculations to produce effective nuclear parameters for subsequent global reactor analysis, (2) static reactor analysis for core design and fuel management, (3) reactor kinetics and safety analysis (4) material-neutronics interface, (5) validation and verification. This section of the overall methods development plan focuses on the development of a suite of deterministic code systems, including spectrum codes, a lattice physics code, and nodal diffusion codes, that can be used for efficient and accurate design of the NGNP. In order to accomplish the project goal efficiently, existing codes will be used as the basis of the new code suite with the addition of required functionalities for VHTR applications. For applications to global reactor analysis, ANL has selected the DIF3D/REBUS-3 code, which has been successfully used for the reactor physics analyses of fast reactors [Toppel 1990]. The INL is developing the PEBBED code specifically for the global analysis of pebble-bed reactors with circulating fuel [Terry 2002]. Although these codes provide the starting point of this project, a significant amount of development is required to enhance their capabilities. Furthermore it is necessary to equip a group-constant generation system that properly incorporates all of the physics of the two basic NGNP concepts, pebble-bed and prismatic (gas or molten salt cooled). Many of the issues to be addressed in this effort are common to both concepts, yet there are distinguishing features that require parallel developments. However, it may be noted that the analytical tools that work for a prismatic reactor cooled by helium are generally adequate for one cooled by molten salt.

General considerations. Enhancements and improvements in modeling capability are thus needed to address unresolved issues in pebble bed and prismatic high-temperature graphite-moderated reactor physics. One of the most important of these issues is the proper preparation of nuclear cross-section libraries suitable for use in the analysis of in graphite-moderated, helium-cooled reactors. As explained in the specific task descriptions below, existing cross-section preparation methods for graphite-moderated reactors will yield poor agreement with continuous-energy Monte Carlo calculations in the thermal and hyperthermal energy range if the double-heterogeneity of the fuel form and assembly leakage are not carefully addressed. Improved treatments are needed since the basic reactor operating parameters are highly sensitive to any inaccuracies in the effective cross sections in this energy range.

Another major development area is in core simulation; i.e. enhancing PEBBED and DIF3D to provide lifecycle analyses using modern methods. For the prismatic designs (cooled by either helium or molten salt), the DIF3D/REBUS-3 code developed at ANL provides much of the capability required of a core simulator, although it needs additional development (e.g., thermal feedback and de-homogenization models for these reactors). Modern core design and optimization techniques that are used widely in LWR fuel management would have to be applied to DIF3D/REBUS-3 given the largely unexplored parametric space of NGNP fuel and the potential impact on economics of the reactor. For the pebble-bed reactor, the PEBBED code developed at the INL is the only core simulator that models pebble flow and is available to the DOE complex. It possesses an advanced optimization capability but currently lacks the ability to model some distinguishing features of the pebble bed reactor.

A third issue is the modeling of the power deposition. The proper accounting for production and transport of gamma photons should be included as part of the analysis code suites. The power deposition distribution can differ significantly from the fission rate distribution due to the transport of gammas away from their creation sites.

A fourth issue is the extension of the material damage method to high temperatures (for both prismatic and pebble-bed designs). Current material damage modeling techniques assume very low (0 K) material temperatures that have not been shown to be valid at higher temperatures. Furthermore, the effects of high-temperature annealing (during either normal operation or accident conditions) must be characterized as part of the safety case.

The work described herein begins the process of completing the suite of analysis methods to permit the full scope of NGNP design analysis calculations to be performed with state-of-the-art tools. An integral part of the development and testing of the new capabilities will be the assessment of their implications for NGNP design limits. The INL and ANL will be leading the research efforts for these tasks and are cooperating on the identification and development of a code suite that incorporates the techniques required for accurate analyses of all current candidate NGNP concepts. These labs have already established working relationships with a number of universities and international organizations that have an interest in gas reactors. A number of workshops and electronic information exchanges have taken place and more are planned as the overall NGNP project proceeds. It is anticipated that the bulk of the code development effort will be completed in the first five to six years of the overall effort. After that, code maintenance, validation, and application to the ongoing NGNP design effort will continue for the duration of the project.

Special considerations for pebble-bed reactors. Until recently, design analysis methods for pebble-bed reactors have been several generations behind the state of the art for light-water reactor design and analysis. For the past five years, the INL has been engaged in the development of analysis methods for high temperature gas reactors, with a special emphasis on the pebble bed reactor. The laboratory has developed the PEBBED code for reactor physics and fuel cycle analysis and the PARFUME code for fuel materials analysis [Miller et al. 2002]. A sample PEBBED graphical neutron flux output is shown in Figure 4-7. In addition, the INL has developed a method for quantifying material damage in graphite and silicon carbide reactor materials. The availability of these tools has made possible innovations and discoveries that could not have been achieved without them. For example, the INL determined the reason for the success of the German TRISO fuel particles and the failure of other countries' fuel [Petti et al. 2003]. Using a genetic algorithm developed to work in conjunction with PEBBED, the INL optimized design parameters to achieve a passively safe pebble bed reactor design of 600 MWt, a goal that had not previously been attained [Gougar et al. 2004]. Using PEBBED, the INL was also able to propose design enhancements to the pebble bed reactor that increased safety during a potential water ingress accident and improved fuel economy and utilization [Gougar et al. 2003]. All these significant design improvements were attained as incidental results of the verification of the new methods and the testing of their capabilities and the resulting extension of the design limits that can be reached.

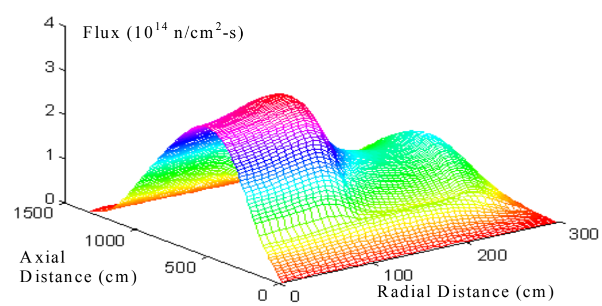


Figure 4-7. Thermal neutron flux profile in the NGNP 600 MWt reactor calculated by PEBBED.

One of the licensing issues for pebble bed reactors is the perception that the semi-stochastic nature of the pebble distribution admits the possibility of collection of relatively reactive pebbles in regions of high neutron flux, so that the local power density could become excessive either in normal operation or in accident scenarios. Some rough estimates of the consequences associated with this phenomenon were performed at the INL in 2003. In this study, various pebble flow scenarios were modeled that resulted in abnormally high concentrations of fresh fuel. The probability of such configurations occurring as a result of stochastic variation is many orders of magnitude lower than the typical beyond-design-basis events normally considered in high temperature gas reactor analysis. Yet the nominal and accident fuel temperatures attained during these events was shown to be only somewhat higher than normal and still lower than those required to induce significant fuel failure. However, in light of the pebble temperature measurements made in the AVR [Baumer et al. 1990], some of which were much higher than expected, a more rigorous analysis is warranted and is possible with advanced Discrete Element techniques and other new modeling tools. The modeling techniques should be employed to characterize and bound the stochastic component of pebble movement and feed these probabilities to a reasonably conservative core safety analysis. Such tools can also be used to develop flow models for pebbles in the discharge and entry regions and to examine pebble-packing issues.

Special considerations for prismatic reactors. In 2003 the INL developed "point designs" for pebble-bed and prismatic-fuel versions of the NGNP [see MacDonald et al. 2003]. The objective of the point design project was to develop a reactor specification with a mixed mean coolant core outlet temperature of 1000 °C, passive safety, and about 600 MWt of power. Sensitivity studies for various block-fuel parameters were performed at ANL and the INL to address design issues critical to this objective and to provide data for developing a fuel specification for the AGR program discussed in Section 2 of this document. For modeling the block-type NGNP with great geometric detail, the Monte Carlo code MCNP [LANL 2000] has proven itself to be a very powerful tool; in conjunction with the ORIGEN depletion code [Croff 1980] with a coupling code such as MOCUP [Babcock et al. 1994], it can follow a block-type core through its fuel cycle and produce accurate burnup and isotopic buildup data in each fuel block. Since MCNP requires long calculation times to produce good statistics, it is not a practical tool for performing large numbers of calculations in design studies, generating detailed core power distribution, or analyzing the effects of small perturbations. For some scoping studies and sensitivity analyses, the ANL codes DIF3D and REBUS-3 have been successfully applied. However, additional developmental work needs to be done to improve computational models and user friendliness.

4.2.1 Fuel Cell and Assembly Spectrum Analysis for Cross Section Homogenization

For NGNP applications, both the DIF3D/REBUS system and PEBBED require cross-section data preparation using specialized techniques that are not implemented in current software in the form that is needed. Cross sections used by PEBBED and DIF3D are calculated externally and passed to these global reactor analysis codes as input. For PEBBED simulations (and DIF3D analyses of the New Production Reactor), these cross sections were calculated by the INL's COMBINE code [Grimesey et al. 1991] or by MICROX-2 [ORNL 1999], neither of which is able to accurately model certain phenomena that are characteristic of graphite-moderated reactors with highly heterogeneous fuel arrangements [Keller et al. 2004].

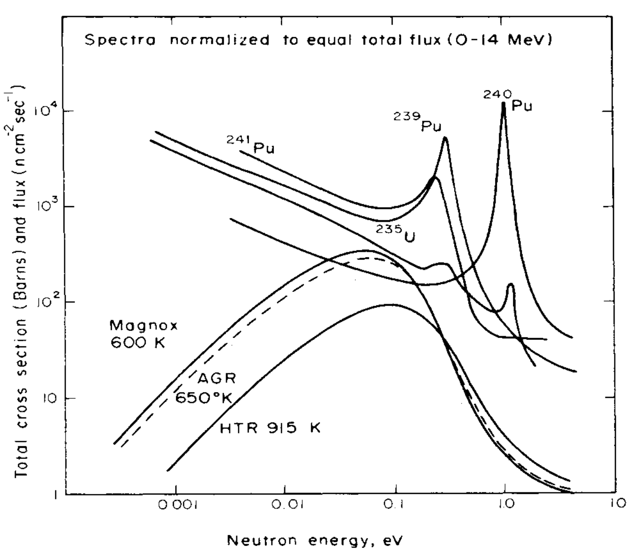


Figure 4-8. Typical VHTR spectrum and some low-lying resonances.

As illustrated in Figure 4-8, thermal neutron spectra in graphite moderated reactors have higher characteristic average energies than is the case for water moderated reactors, especially at the high (900 – 1000 °C) operating temperatures that are anticipated for the NGNP [Massimo 1976]. As a result, special methods are required to account properly for self-shielding of resonances in the thermal energy range. This becomes particularly important in situations with high burnup because of neutron upscattering into the prominent low-lying resonances in plutonium. Core reactivity, temperature coefficients and other related phenomena are all highly dependent on proper modeling of resonance effects. The required computational improvements will thus be developed under this plan and implemented into the spectrum codes as appropriate.

Furthermore, the random distribution of fuel kernels within a compact or pebble is not accurately treated in any of the available codes. This also adversely affects the accuracy of the all-important resonance shielding calculations that are required to produce accurate cross sections. Efforts are under way at the INL and elsewhere to address a range of issues associated with fuel based on the TRISO coated particle and to implement improvements in a widely available code. The modifications to any of these codes could be construed as forming the basis for a future DOE-owned stand-alone modern code for a proper treatment of resonances.

Finally, it should be noted that preparation of required cross section libraries from the basic nuclear data files also involves very sophisticated data processing prior to application of the spectrum and assembly cross section generation calculations described above. Under this R&D Plan, the INL will also maintain the necessary expertise and software tools required for this step in the overall reactor physics analysis sequence. This will include active participation in the National Nuclear Data Center's Cross Section Evaluation Working Group (CSEWG), as well as in corresponding international organizations whose focus is on the key interfaces between basic nuclear data measurements and the final ENDF files that have been evaluated and released for use in subsequent reactor physics applications.

4.2.1.1 Development Of A Method For Improved Treatment Of Double Heterogeneity Using Improved Dancoff Factors.

An important aspect of improving cross sections is to account better for the heterogeneity on two scales in the NGNP: on a fine scale associated with the fuel particles, and on a more coarse scale associated with the pebbles or fuel compacts. For the pebble bed reactor, self-shielding and shadowing effects are important and must be accounted for on both scales. In the continuous-energy Monte Carlo code MCNP, it is possible to model every single fuel grain in the reactor, using the repeated structures feature. However, in deterministic codes like PEBBED or DIF3D, these features are accounted for in the cross sections they receive as input. Such cross sections must be generated using explicit modeling of the heterogeneity or using correcting factors that account for it (the Dancoff factors). Previous studies show that the available Dancoff factors are not sufficiently accurate. Hence, the new method must also include accurate corrections for the effects of double heterogeneity [Rahnema et al. 2004; Hudson et al. 2005].

Furthermore, existing codes used for gas reactor analysis were developed without a full appreciation of the importance of randomness in particle distribution. Recent studies indicate the error introduced by assuming a regular array of fuel lumps. Neither COMBINE nor MICROX gives acceptable agreement with continuous-energy MCNP calculations (Table 4-2). The treatment of randomness in DRAGON may also be inadequate. A

Table 4-2. Spectral indices as computed by MCNP and MICROX-2.

			BCC 61% cold	BCC 61% hot
ρ^{28}	**Epithermal-to-thermal ^{238}U captures**	**MCNP**	5.91	7.50
		MICROX-2	6.77	9.23
		diff (%)	***15***	***23***
δ^{25} x 10^2	**Epithermal-to-thermal ^{235}U fissions**	**MCNP**	9.63	10.99
		MICROX-2	9.99	11.34
		diff(%)	**4**	**3**
δ^{28} x 10^4	**^{238}U fissions to ^{235}U fissions**	**MCNP**	28.08	30.14
		MICROX-2	27.19	30.44
		diff (%)	***-3***	***1***
C^*	**^{238}U captures to ^{235}U fissions**	**MCNP**	0.360	0.463
		MICROX-2	0.404	0.551
		diff (%)	***12***	***19***

rigorous treatment of randomness in the distribution of fuel lumps has been developed at the INL and incorporated into the construction of Dancoff factors. A similar treatment for explicit geometry lattice codes like DRAGON will also be explored.

4.2.1.2 Develop the interface between a spectrum code and a pebble bed reactor core simulator.

In the pebble bed reactor, fuel elements (pebbles) move in a semi-continuous fluid-like manner through the core during operation. Recirculation of partially burned pebbles means that any pebble in the core is surrounded by pebbles with a wide range of burnup. Furthermore, the spectral history of each pebble is unique and can only be approximated (using a code such as PEBBED). Batch-loaded cores (LWR or prismatic gas-cooled reactor) generate cross sections from unit cell burnup calculations assuming fixed boundary and spectral conditions. This approach is not valid for the pebble bed reactor. Instead, cross section and core simulation calculations must be executed simultaneously and iteratively to obtain the proper burnup conditions in each spectral zone (the pebble bed reactor analog to an assembly or block). A simple version of this iterative process has been implemented using PEBBED and MICROX. It uses zone leakage and temperature to obtain cross-sections by interpolation among pre-computed data sets. Because the spectral zone itself is not clearly defined and contains a randomly packed assortment of pebbles, a geometrically rigorous spatial transport calculation (2- or 3-dimensional) is neither wholly effective nor computationally efficient for the algorithm described above. A 1-dimensional (spherical) calculation with appropriate Dancoff factors and isotopics provided from PEBBED can yield cross-sections with the required accuracy.

In a proposed approach, the INL spectrum code COMBINE will be modified to exploit the new Dancoff treatment and be coupled to PEBBED. COMBINE solves the one-dimensional (spherical) B-3 approximation to the transport equation with Bondarenko treatment of unresolved resonances and Nordheim treatment of resolved resonances. It uses Dancoff factors to correct the resonance calculation in the presence of arrays of fuel lumps and has options for self-shielding of cross sections in the thermal range. No specific geometry specification of an assembly or lattice is required and thus it is suitably fast and accurate for the PBR problem described above. Isotopics of the local pebble distribution, leakages from adjacent spectral zones, and zone temperatures will be fed to COMBINE. COMBINE will use these to generate homogenized cross sections for the nodal diffusion calculation and burnup-dependent cross sections for depletion if individual pebble flow streams.

COMBINE results have been compared favorably to analytic benchmarks and other calculations (ANISN and TWODANT). Further validation will be required for NGNP-specific configurations such as:

- Infinite array of identical fuel pebbles (double-heterogeneity)
- Infinite array of pebbles with a limited number of specified burnups (burnup effects)
- Semi-infinite array of fuel pebbles and graphite reflector (leakage effects)
- Full or half-core models of pebble-bed NGNP with specified axial burnup gradient

Corresponding MCNP models can be set up as benchmark cases. COMBINE and other spectrum codes can be tested against these cases and each other to assess the effectiveness of various solution schemes.

4.2.1.3 Identify or Develop an Assembly Code for Prismatic Block Cross Section Generation

The prismatic reactor core, on the other hand, is composed of hexagonal graphite blocks containing coolant channels and fuel compacts. The compacts contain TRISO particles distributed randomly within. A core simulator code such as ANL's DIF3D has as its basic computational element a hexagonal cell for which few-group diffusion coefficients must be computed by a lattice or assembly code once the basic unit cell cross sections have been determined. Previous analyses indicate that an under-prediction of about 3% in k-infinity for a fuel element can occur if the fuel-graphite composite is treated as a homogenized mixture. Therefore, the lattice transport code to be used for group constant generation must be able to treat the double heterogeneity properly and, in addition must account for spectral variations across the basic lattice unit via appropriate neutron transport computations. This capability is available in a few lattice physics codes such as WIMS8, APOLLO2, DELIGHT, and DRAGON [Marleau et al. 2000]. Where such capabilities exist, the codes (e.g., WIMS8 and APOLLO) are typically proprietary and are only available at great cost. In some cases, the source code is not available for release.

This makes the DRAGON code an attractive option, and for this reason further assessment and development of the DRAGON will be a major task in this project. This work will be done in collaboration with the researchers at the Ecole Polytechnique de Montreal who originally developed the code. ANL has organized an information exchange with them (for example, a workshop is to be held at ANL in February 2005). During FY-05, a complete assessment of the DRAGON code deficiencies and identification of the necessary modifications that would make the code attractive for prismatic NGNP applications will be provided. It is also intended to obtain better code documentation, support for an alternative data library, and descriptions of advanced models and capabilities not in the public domain (e.g., methods of characteristics solution, homogenization/de-homogenization, and parallel code version).

This effort will be coordinated with other ongoing and proposed projects. For example, there is already an I-NERI project underway and centered at the INL to develop safety analysis codes with experimental validation for a prismatic VHTR. The collaborators include the University of Michigan and KAIST (Korea). One of the tasks in this project is to use MCNP to develop cross sections for DIF3D in place of a deterministic assembly code. This project complements the DRAGON effort and will provide an essential comparison of methods. While useful for principal cross sections, however, generation of cross sections with Monte Carlo calculations is not very practical for group scattering cross sections given the tremendous computational effort required.

4.2.2 Static Analysis For Evaluations Of Criticality And Power Distribution

Fuel management and design optimization: The fundamental quantity in reactor physics analysis, which determines all other aspects of core behavior, is the neutron flux distribution. Extremely accurate calculations of the neutron flux, accounting for great geometric detail, can be made with Monte Carlo codes such as MCNP. However, Monte Carlo codes are still prohibitively expensive for use in repetitive design and tradeoff calculations or analysis to determine local reaction rate distributions or small reactivity effects. Nor can current coupled Monte-Carlo-depletion codes be applied to the pebble bed reactor (Figure 4-9), because they do not account for fuel movement during operation. Deterministic codes offer much greater computational speed, at the cost of reduced geometric modeling capability. However, the natural geometric configurations of both pebble-bed and prismatic VHTRs lend themselves to accurate modeling by deterministic codes: circular cylindrical geometry for the pebble bed reactor and hexagonal geometry for the prismatic reactor. For prismatic reactors, the hexagonal nodal code DIF3D is a modern deterministic tool. It requires additional work to implement a thermal feedback model and a tabulation scheme of nodal cross sections versus depletion and temperatures. Deterministic code systems available for design and analysis of pebble bed reactors are generally based on older, less accurate methods that do not take advantage of advances in computer capabilities. These old methods have been superseded by more convenient and accurate capabilities, of which the central feature is their application of nodal techniques. DIF3D possesses such a nodal solver. A new nodal technique for cylindrical geometry is being developed with NERI funding and is being implemented in PEBBED [Ougouag 2004]. This NERI draws on support or cooperation from the Georgia Institute of Technology, Penn State University, the University of Arizona, and PBMR

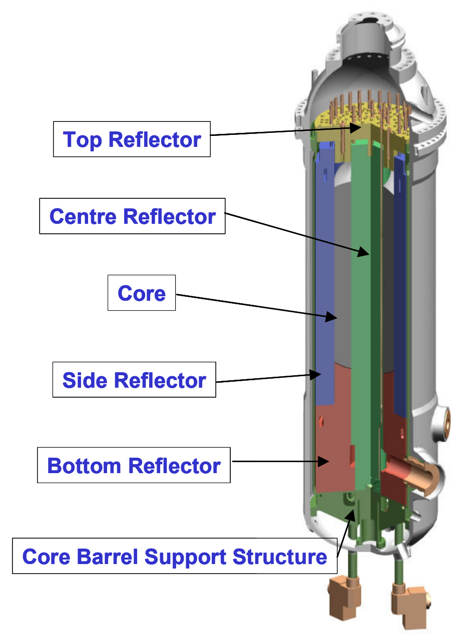


Figure 4-9. Sketch of the PBMR.

(Pty), Ltd. of South Africa.

While PEBBED has been used for some basic equilibrium core design problems, it still required development of some key features that will allow it to be used extensively as a design and analysis tool for the NGNP. Some of the more important ones are listed here.

1. Complete implementation of the in-line cross-section generation capability that accurately treats resonance effects of the doubly heterogeneous fuel, leakage, and temperature effects.
2. Implement the capability to model time-dependent fuel loading cases (non-equilibrium cores).
3. Implement coupled neutronic and thermal-hydraulic transient capability that properly computes fuel and moderator temperatures.
4. Implement a gamma photon transport and adjoint (variational) computational capabilities.

PEBBED does possess an advanced optimization routine (a genetic algorithm) that allows automated searches for optimal core designs and fuel loading patterns. Such methods have been applied to LWR codes for some time but have yet to be developed for the prismatic reactor. This will need to be addressed. Fuel blocks are proposed that may have compacts with differing enrichments, packing fractions, and burnable poison concentrations. The block-refueling pattern may be strictly radial or may have an axial shuffling component as well. Core optimization and fuel loading must be automated to some extent to produce viable cores within practical time limits. There are a number of advanced optimization approaches, including genetic algorithms, simulated annealing, neural networks, Tabu search, and others. One or more of these will be explored and implemented in conjunction with DIF3D/REBUS-3.

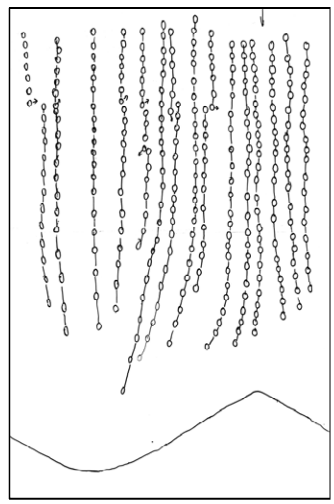
Neutron transport: For some core physics issues, diffusion methods are not appropriate, and transport methods are required. Often, it suffices to use transport methods on a local scale, to incorporate transport effects into diffusion-theory parameters such as cross sections, as discussed in the previous section. For example, in the pebble bed reactor, control rods are required to shut the reactor down rapidly on demand and keep it subcritical at low temperatures. The pebble bed reactor also contains a large gas plenum above the core through which pebbles are dropped. Diffusion theory alone cannot accurately predict neutron transport in these regions, so some sort of transport calculation is necessary. Nor does diffusion theory solve the gamma transport problem and thus the actual heat deposition distribution differs from what it can compute. Various whole-core transport methods and codes are being developed that can address these problems. Partial-core models can also be used to quantify the error resulting from the diffusion approximation. For the prismatic core analysis, DIF3D contains a variational transport solver that can properly treat regions in which diffusion theory is not valid. A NERI project, led by F. Rahnema of the Georgia Institute of Technology in conjunction with the INL, is investigating novel neutron transport techniques that can be used to accurately treat gas plenums and control rod regions in the pebble bed reactor [Rahnema 2004]. In addition to code and model development activities, reactor physics workshops and meetings on this item will be conducted throughout the life of the NGNP project.

Monte Carlo techniques are free from all these VHTR modeling issues if fuel particles are modeled explicitly in the core calculation. However, this detailed modeling is currently unattractive because of the tremendous problem size and because a very large number of neutron histories is required to resolve fuel-element power distribution and small reactivity effects. As a practical matter, the necessary calculations are beyond the current capabilities of even the most sophisticated computers. Furthermore, several important phenomena such as thermal feedback at power generating conditions, flux uncertainty propagation in the depletion calculation, and fission product buildup are not properly addressed in these tools at the present time. To do so with the Monte Carlo technique would increase the computational requirements even further. In contrast, deterministic three-dimensional whole-core transport calculation provides a possibility of resolving all these problems as long as the proper fuel modeling and thermal feedback capability is incorporated in the underlying effective cross section data.

More recently, computational techniques and advances in parallel computing have made feasible more detailed 3D deterministic transport calculations for some applications such as accurate gamma transport and deep shielding. DeCART, Attila [Gougar 2004], and EVENT [De Oliveira 1998] are codes that will be explored for use in the project. It is noteworthy that DeCART perform whole-core transport calculation in fine group level for heterogeneous geometries and thus it avoids the cross section homogenization and condensation steps. Attila also can be run with very high energy resolution. Full-core transport modeling with these codes will still require tremendous computational power and is not practical for scoping or design optimization calculations. Rather, such techniques (Monte Carlo or deterministic) will be useful for benchmarking activities once a design has been rendered using the other approaches described above. With continued advances in computer power and implementation of innovative numerical solution methods, these techniques may in the future provide a practical, high-fidelity capability for routine use in design and analysis.

Isotope depletion. For prismatic reactors, the depletion code REBUS-3 was written to perform depletion calculations in coordination with the DIF3D nodal diffusion code. The DIF3D/REBUS-3 code system is capable of multigroup flux and depletion calculations in hexagonal-Z geometry. This code system uses the DIF3D module as the flux solver and contains both nodal diffusion and transport theory capabilities. Therefore, it can be adapted to prismatic NGNP reactor problems with limited effort compared to other codes. Lumped fission product models, however, need to be developed in order to make the microscopic depletion scheme used in REBUS-3 practical.

PEBBED is a combined diffusion/depletion code conceived to solve self-consistently for the neutron flux and the burnup distribution in a pebble bed reactor with circulating pebbles. As noted above, it was originally written with a finite-difference diffusion solver, but nodal diffusion modules have been installed in the code and are currently undergoing checkout and debugging. An analytical nodal solver using a "moments-stepping" method that allows for variable cross sections within nodes has been developed by an INL researcher. Such advances in burnup calculations will be explored as a complement to the development of the nodal diffusion solver. and will be implemented in PEBBED to complement the nodal solution.



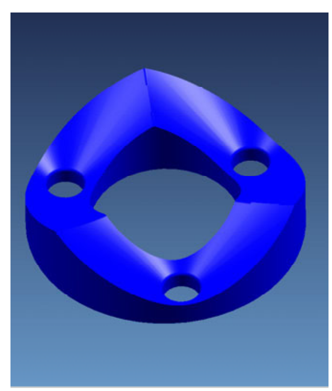


Figure 4-10. Pebble flow lines (top) and sketch of the defueling cones near the bottom of the PBMR core.

Pre-asymptotic core analysis in the pebble bed reactor. PEBBED obtains the asymptotic distributions of neutron flux and burnup directly, without following the time-dependent distributions in the run-in period. This property of the code permits very rapid solution. However, a typical pebble bed reactor may take as many as three years to achieve an asymptotic state. A theoretical formulation for pre-asymptotic core analysis is under way at the INL and will ultimately lead to time-dependent solutions of the coupled pebble-flow/burnup problem.

Non-axial pebble flow in the pebble bed reactor. The flow of pebbles in a pebble bed reactor is not strictly axial, particularly near the discharge tubes (see Figure 4-10). While the neutronic importance in this region is minimal, a method and code must be developed that link together depletion zones along the true flow path of pebbles even for flow lines that are not strictly axial. In this development the axial flow of pebbles is modeled, as in the previous case. In addition, the radial drift of pebbles is also accounted for. Effective pebble flow characteristics are developed and used to link computational coarse nodes systematically. Experiments and some computations have been performed that confirm and characterize the strong deterministic (streamlined) component of pebble flow. A computational approach to pebble flow using a particle dynamics approach has been initiated at the INL and similar efforts are underway at MIT and other universities. Results from these efforts will be used to construct the flow lines over which the depletion equations are solved in PEBBED.

4.2.3 Kinetics, Thermal Module Coupling, and Feedback

Three-dimensional spatial kinetics capabilities have been under development for more than 20 years. Practical tools now exist and include the public versions of NESTLE [EPRI 2003], PARCS [Joo et al. 1998], VARIANT-K, and DIF3DK. These high-fidelity kinetics methods are important for core transients involving significant variations of the flux shape but have not been systematically applied to graphite-moderated, helium-cooled reactors. In the future, integrated thermal-hydraulics and neutronics methods should be extended to enable modeling of a wider range of transients pertinent to the NGNP. Required advances include increasing the efficiency of the coupling approaches and improving the representation of cross section variations.

For the pebble bed reactor, the fixed-source solver in PEBBED will be extracted and implemented into a transient thermal-hydraulics code to calculate the flux and power distribution. Kinetics parameters such as the delayed neutron fraction will also have to be generated for the kinetics code. These are generally obtained from adjoint flux calculations on the core model. DIF3D possesses adjoint capability, but PEBBED currently does not. The analytic nodal equations will be modified to allow for adjoint solution and the solver in PEBBED will be upgraded. Next, time-dependence must be introduced into the PEBBED fixed-source solver and the source term must be reformulated to account separately for prompt and delayed neutrons, for neutron kinetics analysis within a pebble bed reactor systems model for a code such as RELAP5. A cooperative research effort between Penn State University and PBMR (Pty), Ltd of South Africa is underway to develop a coupled neutronics-thermal-hydraulics code for pebble-bed reactor transient and safety analysis starting with the NEM code [Ougouag et al. 2004]. The INL plans to participate; the PEBBED code would be used to generate steady-state conditions to be fed to the transient code. This is a good bridge to a complete steady state and transient code.

Kinetics is the study of very short-term transients in the neutron flux, so it is probably not necessary to account for the slow motion of fuel pebbles in kinetics analyses. The portions of PEBBED that account for fuel motion (the burnup equation solver and the pebble recirculation matrix) are not needed for kinetics calculations, and it would unnecessarily encumber RELAP5 to couple it to the full PEBBED code. Initially, the time dependence in the kinetics equation will be treated with finite-difference techniques, but once this method is working more advanced approaches to treating time dependence will be explored.

Furthermore, steady state thermal hydraulic calculations correctly assume a close coupling of fuel and moderator temperatures. This assumption is invalid in sharp transients during which the kernel temperature can rise dramatically and independently of the surrounding graphite. Whichever transient codes are used for either prismatic or pebble-bed analysis, a proper separation of fuel and moderator temperature effects must be implemented.

Nodal diffusion and transport kinetics capabilities have been developed for the DIF3D code in the past. These capabilities have been successfully applied for transient analysis of thermal reactor systems (e.g., NPR-HWR, RMBK, VVER, and LWR) by integrating them in a system analysis code, SASSYS. Initial estimation indicated that a multigroup analysis (about 20 groups) is required to represent accurately the reactivity effect of spectral change. The multigroup capability of DIF3D would be attractive for integration with a system code, such as RELAP5/ATHENA, that can be utilized for the analysis of the NGNP. Eventually, it can be upgraded with the new kinetics treatment described above.

4.2.4 Material-Neutronics Interface

Optimization of kernel size and packing fraction. One design issue that be studied as the new tools are developed is the diameter of the fuel kernel within the TRISO particle. Changes in this parameter (while otherwise maintaining the overall TRISO structure and overall diameter) are expected to have profound effects on fuel burnup, fission product migration, and fuel particle reliability. Similarly, studies of particle packing fraction have begun and may have significant effects on fuel performance and core design for either the prismatic or pebble-bed concepts. These and related fuel design parameters may have direct impact on economics and safety. Work has begun in this area and should continue [Ougouag et al. 2004].

Irradiation Effects and Annealing Feedback. In traditional neutron kinetics codes, suitable for the analysis of light-water reactors, thermal feedback is usually accounted for. Other forms of feedback are unimportant and are not explicitly modeled. For the NGNP, the situation could be drastically different. Of particular importance is the change in material properties caused by radiation. For example, the thermal conductivity of graphite is degraded gradually as radiation damage accumulates. Similarly, some nuclear properties, such as the scattering cross sections, are altered by the damage. During transients, the increase in temperature may anneal some or all of the damage, resulting in (partial) property recovery. This could, for example, imply that the scattering cross section would increase during a transient, resulting in stronger thermalization properties and an increase in reactivity. Other similar phenomena are believed to occur that also have a potential impact on the safety of the NGNP during extreme transients. The feedback mechanisms just described must be incorporated into the kinetics codes. The characterization of irradiation damage and its effects on material, neutronics, and thermal properties is another important task in this research [Hawari 2004].

4.2.5 Validation, Verification, and Ongoing Improvement of Code Suite

The resulting suite of deterministic codes developed above will be verified against reference solutions obtained using Monte Carlo and deterministic models and against integral experiments. The reference (numerical) solutions will enable the accuracy of specific assumptions and approximations to be tested and verified. The double heterogeneity treatment will be examined for detailed fuel block and pebble problems by comparing the lattice code solutions with continuous-energy Monte Carlo solutions. The whole-core solution scheme will be verified against multi-group Monte Carlo solutions using pre-calculated multi-group cross sections and homogenized fuel-element models. The pebble bed reactor solution will also be compared against results from the code VSOP [Teuchert et al. 1980].

As the improvements and the extensions are implemented, the overall accuracy of the resulting suite of codes will be quantified by analyzing appropriate integral physics experiments. All known reactors, critical facilities, and other experiments of both types have been assessed for suitability as benchmarks [IAEA 2001]. For the prismatic reactor, the HTTR facility in Japan [Fujikawa et al. 2001] possesses large amounts of critical reactor physics data that can be used for validation purposes. For the pebble bed reactor, the HTR-10 [Zhong and Qin 2001], and possibly PROTEUS, may also provide essential data. HTR-10 has in fact been selected as the initial candidate for a full evaluation under the integral benchmark data task as noted previously. In an independent effort, a set of neutronic and thermal-hydraulic benchmarks for the PBMR has just been accepted by the OECD/NEA. This benchmark effort has been a cooperative effort among PBMR, (Pty), Ltd., Purdue University, Penn State University, and the NRG Corporation of the Netherlands. The INL has been invited to join this effort. Formal participation by the INL is expected to begin in June of 2005 when the next PBMR benchmark workshop is held in Paris in conjunction with the OECD/NEA meeting.

Validation and verification of the tools used for these predictions can be accomplished through the collection of a large compendium of relevant in-core critical experiment data into a detailed, peer-reviewed standard format as described previously in connection with the IRPhEP. Such an approach has also been taken by the USDOE-NNSA in handling the validation and verification for stockpile stewardship where computer modeling is also relied upon extensively. In support of this effort, it would be appropriate to ultimately establish and promulgate validation and verification standards, or at least some set of test problems, for the Generation IV systems. If suitable validation and verification data do not exist, experiments will have to be designed and conducted to fill in the gaps.

Monte Carlo simulation itself provides a powerful tool for validation and verification. The recent and continuing growth in computer power motivates the assessment and further development of Monte-Carlo-based analysis capabilities applicable to multiple reactor types. Enhancement of these codes would also be investigated, including the propagation of errors as a function of depletion, provision of temperature interpolation capability, and modeling of thermal-hydraulic feedback.

The need for appropriate nuclear physics experiments (e.g. critical facilities) may emerge from this effort as the current nuclear data picture comes into focus. Every large R&D program leading to the development of a working reactor was accompanied by extensive nuclear data measurements and evaluations. The operating range of the NGNP (temperature and burnup), along with more stringent requirements of nonproliferation and repository performance, will likely require an extensive experimental nuclear physics component to validate the reactor physics calculations, including the Monte Carlo benchmarks.

4.3 Thermal Hydraulics

The flow and heat transfer in the NGNP are characterized by complex physics in complex geometries. Advanced simulation tools are available to simulate turbulent flow and heat transfer in engineered systems. It is desired to validate such tools to determine their usefulness for applications to the NGNP. It is fully expected that advanced computational fluid dynamics (CFD) codes will be needed to simulate regions of complex turbulent flow in the plant. Despite the size and complexity of the plant, it is currently expected that thermal-hydraulics systems analysis codes can be applied, in conjunction with CFD codes, to analyze the plant fully. The distinction between CFD and systems analysis codes stems from the distinctions between the software tools themselves. CFD codes use first-principle based solutions and subdivide a problem domain into cells that are small with respect to the phenomena that requires modeling. Systems analysis codes use field equations that have been simplified (for example by not including the viscous stress terms) and subdivide the problem into a macroscopic structure that does not model phenomena such as turbulent eddies. Of course, neutronics/fluid behavior interaction will also be important to analyze in the NGNP.

The methodology applied to ensure that the thermal-hydraulic software can be used with confidence to calculate the behavior of the NGNP is outlined in Figure 4-11. However, it is useful to outline how the methodology will be applied specifically for the thermal-hydraulic R&D outlined in this section—since the thermal-hydraulics R&D, including the following summaries, stems directly from this methodology.

The R&D process is progressing as follows:

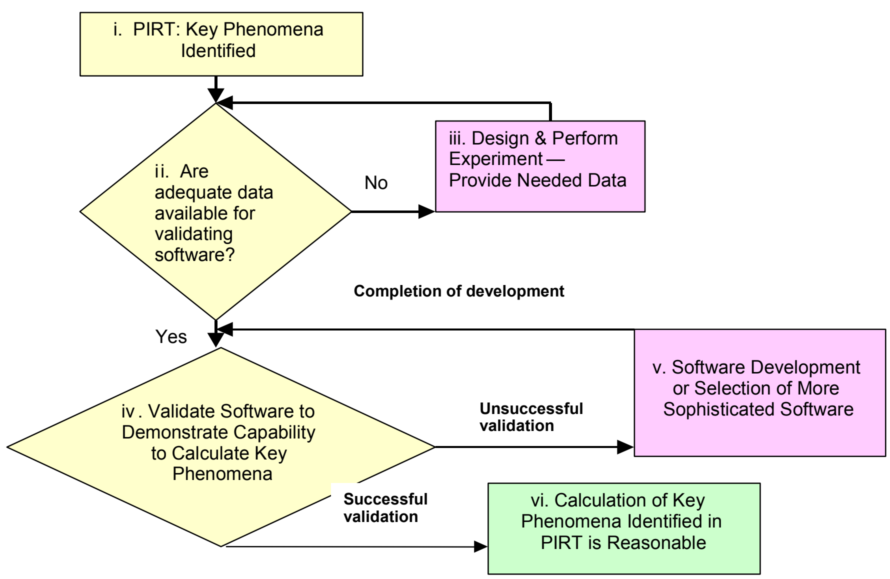


Figure 4-11. Thermal-hydraulic software validation methodology.

a. **The R&D is based on the latest PIRT.** Presently the only available PIRT is the "first-cut" PIRT given in Table 4-1. However, as the design of the NGNP matures, an increasingly more sophisticated PIRT will be required to identify the key scenarios and important phenomena (see Figure 4-11 Step i). Hence the R&D plan is based on the assumption that an ever-improving PIRT will be available. Thus it is clear that all phenomena that must be calculated have not yet been identified. A formal PIRT should be created in conjunction with the pre-conceptual design in approximately 2006 or 2007 and then updated as the conceptual design, the preliminary design, and finally the final designs are formulated. Additional discussions on the upcoming PIRT requirements are summarized below.

b. **The software used to analyze the NGNP behavior must be validated for the scenarios of importance.** The process thus begins using existing data. If either existing data are not available or the existing data are not adequate to cover the NGNP's operational envelope, then experiments must be defined and built, and data must be produced to provide the basis for software validation (see Figure 4-11 Steps ii and iii). Hence the first R&D categories discussed below (Sections 4.3.1, 4.3.2, and 4.3.3) are experimental.

c. **Software development.** If the validation studies show the software cannot adequately calculate the key phenomena in the important plant scenarios, then development must be done to improve the software or, alternatively, more sophisticated software must be used if available or developed if not available (Fig. 4-11 Steps iv and v).

d. **Analyses.** Once the software has been validated and shown to be capable of calculating the important phenomena to the accuracy required (Fig. 4-11 Step vi), then best-estimate analysis may begin.

Software validation, development and analysis (items c and d) are summarized in Section 4.3.4 for both the computational fluid dynamics and systems analysis codes.

Both the experimental research areas and the software-directed research areas are focused on the high-priority R&D areas identified in the "first-cut" PIRT, as outlined in Table 4-1, where key regions of concern are identified. In each case, the issues are whether the system will survive, particularly under the most challenging accident conditions, and whether the system will have an adequate operational lifetime for the conditions that are postulated (rated operational conditions, off-normal operational conditions, and accident conditions). The high-priority research areas include: (i) the core heat transfer, (ii) mixing in the upper plenum, as well as the lower plenum, hot duct, and turbine inlet, (iii) the heat transfer in the reactor cavity cooling system, (iv) air ingress following a system depressurization, and (v) the behavior of the integral system during the key scenarios, including the contributions of the balance-of-plant. These R&D areas are outlined in Table 4-3 together with a summary of the key needs.

The R&D areas, including the relevant R&D tasks and the specific needs, are discussed in more detail in Sections 4.3.1 through 4.3.4. For each of the R&D topics study areas have been assigned (see 3rd column of Table 4-3) that indicate whether the R&D is experimental, i.e., an activity designed to produce validation data; is computational fluid dynamics (CFD) code-related; or systems analysis code-related.

The "first-cut" PIRT focuses solely upon the phenomena that are expected to dominate within the reactor itself and does not include any phenomena that may occur in the balance of plant or in an intermediate heat exchanger (IHX) if one is present. While it is likely that the phenomena that present the most significant risk to the safety of the facility occur within the core, phenomena resulting from equipment failures in the balance of plant or flow instabilities in an intermediate heat exchanger might also have significant impacts on the safety of the facility and may be included in a future PIRT. In particular, the nuclear-chemical coupling IHX, which provides the heat transport to the hydrogen production plant, similar to the core, has many coolant channels that could produce flow instabilities. The IHX will have inlet and outlet plenums, with mixing and stratification phenomena, so the reactor vessel plenum experiments could also apply in this case.

Flow instabilities may also be phenomena with complicated transient effects. At some point, as the PIRT progresses, there may be a future need to separate out a task for the development of transient CFD analyses techniques. As this could be a major undertaking, the budget may need to be adjusted accordingly at that time. What is acknowledged at this stage, even though the PIRTs have not reached this level of detail, is that local features in many cases determine the generation of the flow field turbulence structure. Design features to mitigate thermal stresses, promote mixing, enhance heat transfer and reduce vibration are fluid-structure coupling mechanisms, which need to be treated in the CFD development.

4.3.1 CFD Code Validation Experiments - Introduction

The experiments that stem from the areas identified in Table 4-3 are described in the following six sections. These experiments are aimed at producing validation data for CFD codes. Some potential issues identified to date include "hot streaking" in the lower plenum evolving from "hot channels" in the core (Figure 4-12), the geometric transition from the lower plenum into the outlet duct and the resulting temperature distribution in the short outlet duct, "hot plumes" in the upper plenum during "pressurized cooldown" (loss of flow accident) and parallel flow instability in the core during pressurized cooldown [Bankston 1965; Reshotko 1967]. Several of these phenomena are pertinent to pebble-bed versions of the NGNP as well as the block versions. Although the geometry used as the basis for the first experiments is specific to the prismatic design shown in Figure 4-12, the strategy for designing the CFD code validation experiments is rooted in using scaling studies that will enable the resulting data to be directly related to other designs by using non-dimensional parameters. This approach maximizes the relevance of the specified experiments to the design that is eventually selected whether it is a pebble-bed or a prismatic design. The initial studies will concentrate on the coolant flow distribution through reactor core channels (hot channel issue) and mixing of hot jets in the reactor core lower plenum (hot streaking issue), phenomena that are important both in normal operation and in accident scenarios. In the future we can expect new thermal hydraulic issues to be identified as the NGNP development proceeds through the various design stages and then construction, licensing, and operation.

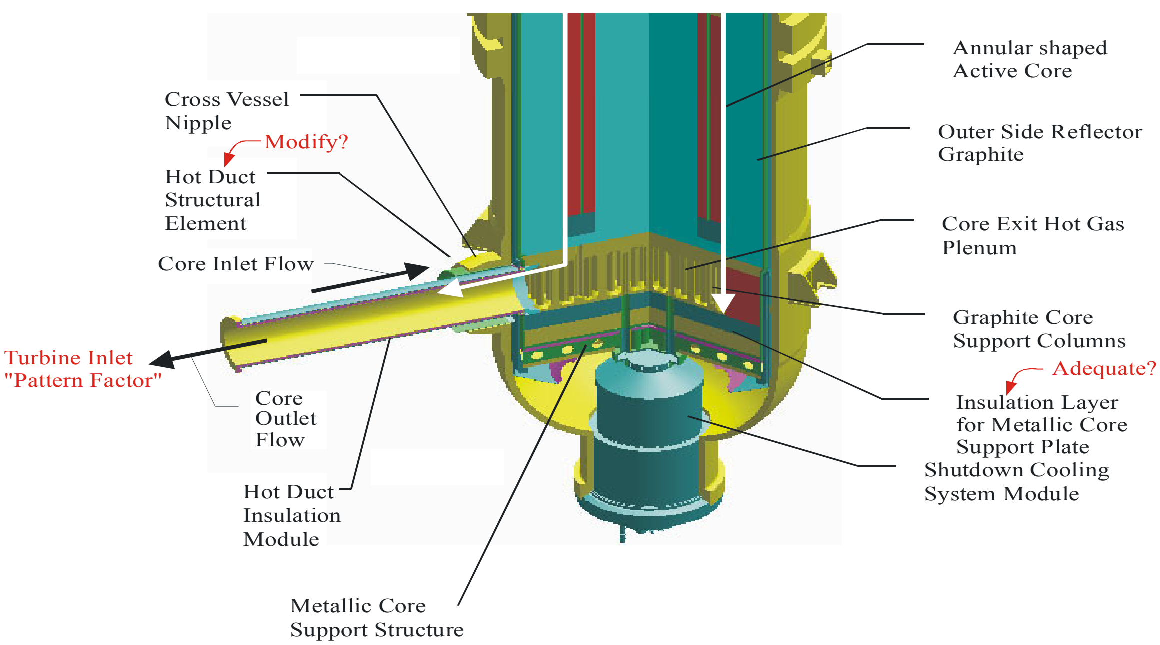


Figure 4-12. Core high power zones generate very hot exit gas and lead to high temperatures at turbine inlet.

Table 4-3. Methods thermal-hydraulics research areas.

R&D Area	Related R&D	Study Area	Need
1. Core Heat Transfer	Mixed convection experiment, heated experiments, core heat transfer modeling, bypass experiments, system performance enhancements, Sana experiments.	Experimental (E), CFD, & systems analysis codes (S)	The core heat transfer, both with cooling flow (operational conditions) and without cooling flow (DCC and PCC), are instrumental in setting the maximum temperature levels for fuel and material R&D (core graphite, structural materials, and heat load to RCCS). The core heat transfer will determine the material selection and configuration in the NGNP core, vessel, and RCCS designs.
2. Upper & Lower plenums (UP & LP)	HTTR UP & LP, HTR_10 UP & LP, MIR, heated experiments, scaled vessel, jets & cross-flow data, upper plenum experiments, system performance enhancements.	E & CFD	Circulation in the upper plenum is important during the PCC scenario since hot plumes rising from the hot core may impinge on the upper head structures and lead to a potential overheating of localized regions in the upper vessel. The degree of lower plenum mixing determines both the temperature variations and the maximum temperatures that are experienced by the turbine blades, the lower plenum, hot duct, and power generation vessel structural components. The lower plenum mixing will determine the material selection and configuration in the NGNP lower plenum, hot duct, power generation vessel, and turbine designs.
3. RCCS	ANL (air-cooled), Seoul National University (water-cooled), HTTR RCCS, fission product transport, system performance enhancements	E, CFD, & S	The heat transfer efficiency of the RCCS will determine the overall design concept (whether air-cooled is sufficient or water-cooled is required in accordance with either a confinement or containment RCCS design), plus material selection of outer vessel wall, coatings (e. g., selection of materials with emissivities that change with surface temperature), natural circulation characteristics, etc.
4. Air Ingress	Diffusion model development, NACOK experiment	E, CFD, & S	A Generation IV reactor system should be able to survive the most challenging accident scenarios with minimal damage and thus should be able to resume operation in a minimum time frame. The system must be shown to sustain minimal damage following potential air ingress into the core region.
5. Integral System Behavior	HTTR, HTR-10, AVR, fission product transport, CFD and systems analysis code coupled calculations, behavior of balance-of-plant components (intermediate heat exchanger, turbine, compressor, reheater), analyses of pre-conceptual design, conceptual design, preliminary design, and final design	E, CFD, & S	The ultimate system characterization, to show the final design is capable of meeting all operational expectations and of surviving the most challenging accident conditions, is performed using validated software tools. The tools consist of the neutronics and thermal-hydraulics software (coupled CFD and systems analysis software) used in concert. This step is the culmination of the comprehensive R&D effort outlined herein.

Meaningful feasibility studies for NGNP designs will require accurate, reliable predictions of material temperatures to evaluate the material capabilities. In NGNP concepts, these temperatures depend on the thermal convection in the core and in other important components. Unfortunately, correlations in one-dimensional system codes for gas-cooled reactors typically under-predict these temperatures, particularly in reduced power operations and hypothesized accident scenarios. Conceptually, CFD codes with turbulence models can yield predictions for improvement of correlations and preliminary design. However, most turbulence models in general-purpose CFD codes also provide optimistic predictions in that they under-predict resulting surface temperatures as shown in Figure 4-13 where the data are plotted as solid red squares [Mikielewicz et al. 2002; Richards, Spall and McEligot 2004]. These treatments are further complicated by the non-homogeneous power distributions with strong peaking that can occur and buoyancy, strong pressure gradients and gas property variations in the channels ("hot channel" issue).

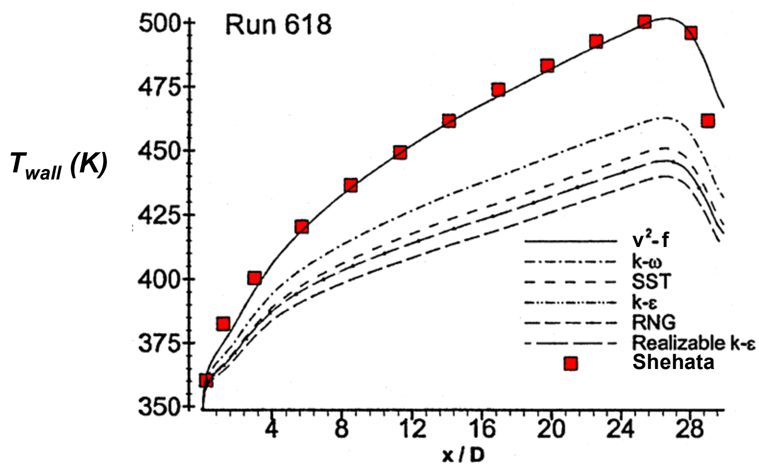


Fig. 4-13. Preliminary assessment of popular turbulence models for flow in a vertical circular tube such as a prismatic NGNP cooling channel.

Unfortunately, no universal turbulence model has yet been developed -- so CFD predictions must be re-assessed via experiments for each new complex situation encountered in the development of advanced reactors and their supporting systems. Further benchmark data are needed for complex situations - to avoid these problems and to improve predictive capabilities. These bases can be obtained from physical experiments or from numerical experiments such as direct numerical simulations (DNS) or large eddy simulations (LES), after validation with measurements. And, ultimately, prototypical integral experiments will be required for licensing confidence.

The general approach is to develop benchmark experiments needed for assessment in parallel with CFD and coupled CFD/systems code calculations for the same geometry. In each case, the benchmark experiments must be linked to the “potential” design by comprehensive scaling analyses that illustrate the relationships between the experiments and design—to ensure the experiments yield benchmark data that are within the design’s operational or postulated accident envelope.

Velocity and turbulence fields will be measured in the INL's unique Matched-Index-of-Refraction (MIR) flow system; these data will be used to assess the capabilities of the CFD codes and their turbulence models and to provide guidance in improving the models. The virtue of the MIR stems from the use of test sections that are clear and with the same index of refraction as the working fluid. Therefore, the experiments are not only quantitive—since scaled data measurements are recorded—but also qualitative since the various flow processes can be visually observed and filmed. Heat transfer experiments will be developed and accomplished for the same purposes. Existing databases from experiments, direct numerical simulations and large eddy simulations will also be utilized where appropriate. The experiments defined below provide essential validation data in the following R&D areas:

1. Core heat transfer experiments:
    a. Turbulence and stability data from vertical cooling channels (2005-2007)
    b. Bypass flow studies (2007-2008)
    c. Exit flows in pebble beds (2006-2008)

2. Upper & lower plenum fluid behavior experiments:
    a. Fluid dynamics of lower plenums (2005-2008)
    b. Heated flows in lower plenums (2005-2007)
    c. Interactions between hot plumes in an upper plenum and parallel flow instabilities (2007-2009)

3. Air ingress experiments: heat transfer and pressure drop of mixtures of air and helium (2006-2007)

4. Larger scale vessel experiments: to examine the behavior in the core, in the plenums, and the interactions between them (2010-2014).

5. Integral Experiments: HTTR and HTR-10

6. Reactor Cavity Cooling System Experiments

In general, two types of experiments are planned: fluid dynamics measurements and heated flow studies. The purpose of the fluid dynamics experiments is to develop benchmark databases for the validation of CFD solutions of the momentum equations, the scalar mixing, and the turbulence models for typical NGNP geometries in the limiting case of negligible buoyancy and constant fluid properties—that is, when the flow is turbulent and momentum-dominated. The intent of the heated flow experiments is to provide data on the modifications of the thermal hydraulic behavior (and proposed turbulence models) as additional effects, such as gas property variation and buoyancy, become important.

The subsequent discussion is divided into four general areas: (1) core heat transfer including experiments 1.a through 1.c above, (2) upper and lower plenum fluid behavior including experiments 2.a through 2.c above, (3) air ingress experiments, and (4) larger scale vessel experiments.

4.3.2 Core Heat Transfer Experiments

Vertical cooling channel turbulence and stability experiments: This experiment will provide documented temperature, velocity, and turbulence field data for forced and mixed convection (buoyancy effects) and gas property variation in NGNP cooling channels in order to validate the turbulence models at reactor conditions for which benchmark data are not available. The proper calculation of turbulence directly influences both the flow and temperature of the cooling. Instrumentation will include miniaturized multi-sensor hot-wire probes developed as a task in a recent NERI project for gas-cooled reactors [McEligot et al. 2002]. Both down-flow (normal operation) and up-flow ("pressurized cooldown") will be considered.

Turbulence modelers request measurements of the basic quantities (dependent variables) of their governing partial differential equations for validation (and guidance), quantities such as turbulence kinetic energy and Reynolds stresses, etc. These data generally have not been available for internal flows with gas property variation. Experimental databases are currently available to assess some aspects of the "hot channel" problem, particularly forced convection, but not with details of the turbulence structure [McEligot 1986]. Additional measurements will be obtained in ongoing university projects. These sources and the existing literature will be compared to the requirements identified above for qualified databases to determine which additional measurements are needed for heated gas flow in circular tubes.

These data will be sought and, if of benchmark quality and available, will be acquired to reduce our experimental needs and costs. Unfortunately, many of the extensive archival publications on mixed convection (buoyancy influences) are not adequate to serve as databases for CFD assessment -- and measurements to evaluate details for turbulence models are lacking for complex situations as in the NGNP concepts. Ultimately sensitivity studies will be required to quantify the relative importance of factors that influence the turbulence, including potential geometrical and hardware configuration variations.

For normal operation, the flow in the NGNP coolant channels can be considered to be dominant turbulent forced convection with slight transverse property variation. In a pressurized cooldown simulation, the flow quickly becomes laminar with some possible buoyancy influences [McEligot and McCreery 2004] and parallel flow instabilities may become important [Bankston 1965; Reshotko 1967]. Flow is predicted to be upwards in the inner and middle rings and to remain downward in the outer ring; heat transfer may be to or from the gas, depending on location and timing. During the initial transient, the turbulent criteria are all predicted to remain below their thresholds for significant effects as in normal steady operations [McEligot and Jackson 2004]. Some insight into the complicated buoyancy influences in heated laminar flows was provided by Scheele and Hanratty [1962, 1963] for developed flows. In contrast to turbulent flow, an "aiding" laminar flow (heated up-flow or cooled down-flow) enhances heat transfer parameters; an "opposing" flow reduces these parameters until it becomes unstable and undergoes transition to a turbulent-like flow. For opposing flow, Scheele and Hanratty suggest that above a threshold there is a transition to an asymmetrical flow with local separation at the wall and then transition to an unsteady and later intermittently turbulent flow; this situation cannot be predicted adequately with a systems code or a steady, axi-symmetric CFD code.

This experiment (Figure 4-14) will support the efforts of the current computational task concerning the hot channel issue by providing benchmark data for detailed assessment of its turbulence models for forced and mixed convection with helium property variation. The miniature multiple-sensor hot-wire probes from Profs. Wallace and Vukoslavcevic (Figure 4-15) will be inserted through the open exit to obtain point-wise temperature and velocity

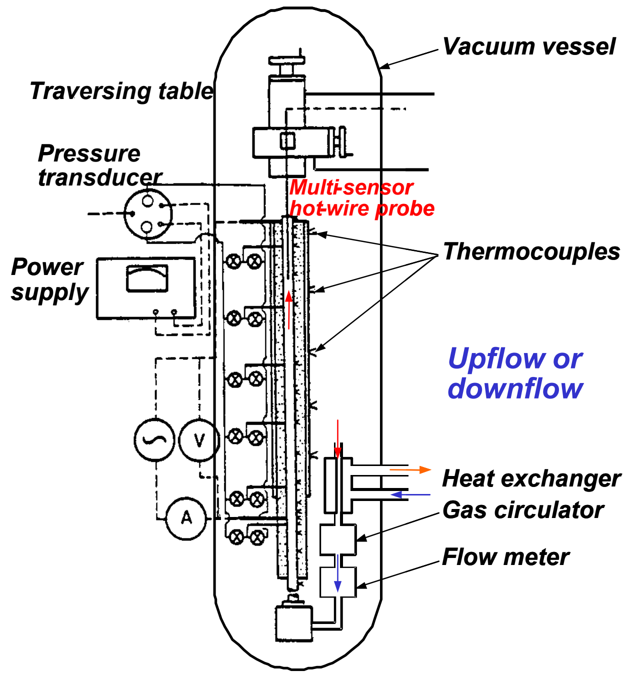


Fig. 4-14. Potential apparatus to obtain benchmark turbulence data in heated channel flow.

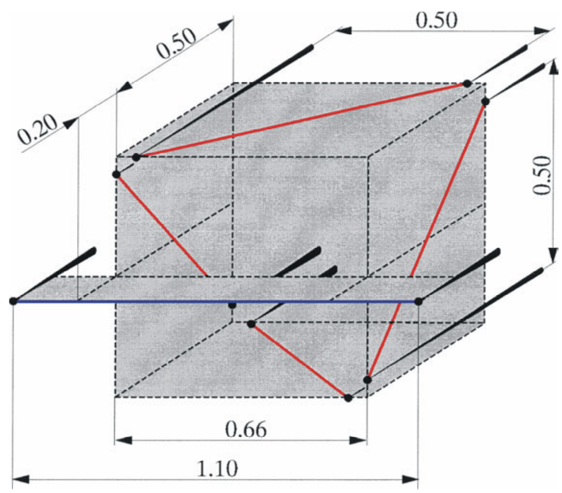


Fig. 4-15. Schematic diagram of miniature five-sensor probe by Vukoslavcevic and Wallace [2003], the dimensions are in millimeters.

measurements. Hence, our objectives are to measure the fundamental turbulence structure and to obtain benchmark data to assess CFD codes for high temperature gas flows that are in the forced and mixed convection regions, for a range of conditions important in VHTRs. To achieve these objectives the experiment will provide an approximately uniform wall heat flux boundary condition in a tube for helium, either ascending or descending and entering with a fully-developed turbulent velocity profile at a uniform temperature as in coolant channels after passing through an end reflector.

The studies for FY-05 include initiating preliminary design of the apparatus and estimating costs and schedules. Final design, re-evaluation of costs and schedules and initiation of fabrication are planned for FY-06 with equipment operation, measurements and documentation occurring in FY-07 and FY-08.

Bypass flow experiments. The bypass is the core flow that moves through the core via the interstitial passages and non-cooling passages in a prismatic reactor and through unanticipated zones of low resistance in a pebble-bed reactor and through the reflector regions in both designs. The bypass may vary from 10% to 25% or more of the total core flow and will vary during the lifetime of the reactor as a function of the local temperature and the changes in the dimensions of the graphite blocks due to irradiation damage. Because the bypass flow exerts considerable influence on the core temperatures and the peak exit cooling channel jet temperature and thus the temperature distribution in the lower plenum at operational conditions, identification of the NGNP core bypass characteristics and its influence on the reactor's peak temperatures is crucial.

The resolution for this phenomenological issue may well have to be a statistical approach similar to that used for the classical hot spot/hot channel factors. A high-level stochastic structure involving a combination of materials modeling, both first-principles and correlations, thermal-hydraulics R&D and manufacturing practice will need to be put in place early. This will guide the research of university participants. It is anticipated that university researchers will investigate the various factors that influence the bypass and perhaps develop preliminary models. Bypass flow experiments are envisioned, in FY-07 and -08, to test and confirm the various theories regarding factors that influence the quantity of flow bypassing the coolant passages and core (in either prismatic or pebble bed reactors). Functional dependences on factors such as manufacturing tolerances and core configuration changes due to irradiation and thermal expansion will be determined. In such experiments hardware may be built to represent core hardware that might result from prolonged irradiation and consequent non-uniform distortion.

Exit flows in pebble beds. An MIR experiment will be conducted to examine flows in pebble beds near their outlets. A key difficulty in analyzing the safety of pebble bed reactor systems is predicting the maximum fuel temperatures and chemical reaction rates locally in the coolant outlet region (e.g., "hot spots") where the temperature field is generally high. Typically, one-dimensional system codes are applied for transient safety analyses and parameter studies during preliminary design [Oh, Moore, and Ambrosek 2000]. A one-dimensional calculation predicts quantities that are averaged across the flow (e.g., the core diameter) and does not predict the highest temperatures or their locations. Further, since chemical reaction rates vary nonlinearly with temperature, the average reaction rate is not the reaction rate at an average temperature. While these systems codes are needed, it is desirable to supplement them with three-dimensional calculations for final designs and for estimating "hot spot factors" to improve their predictions. Potentially, 3-dimensional CFD codes can be applied - using a porous medium approximation - to find the coolant velocity and temperature in localized "macroscopic" regions. Then direct numerical simulations (DNS) can be used to identify the point-wise peak temperatures and their locations ("microscopic" treatment). The goal of this research is to develop accurate techniques for predicting maximum temperatures in NGNP concepts that use pebble bed technologies by coupling CFD calculations with experiments in the unique INEEL MIR flow system.

The flow through a pebble bed core is not uni-directional as in experiments to derive flow correlations. The general flow converges and diverges (in addition to the localized changes in direction at the pore-scale). However, it is well known to fluid physicists that a convergence stabilizes flows [Schlichting 1979]; in a turbulent flow, the turbulence levels can be reduced below expectations and the flow can even be "laminarized" [McEligot and Eckelmann 1993; Satake et al. 2000]. A consequence is a reduction in convective heat transfer coefficient and, hence, an increase in surface temperatures. While criteria for this occurrence have been hypothesized for turbulent boundary layer flows [Murphy, Chambers and McEligot 1983], none is known to us for converging flows in porous media. Appropriate measurements are needed to quantify this phenomenon and, hence, to determine its importance in pebble bed reactor technology.

Under accident conditions (no forced flow) heat is transported by radiation, heat conduction of the pebbles (through the contact points) and convection. An integral simulation can only be done using the porous medium approach. For this treatment several parameters are needed, such as pressure loss and volumetric heat transfer coefficient between gas and pebbles. A model that accounts for the "turbulent" mixing due to the complex path of the gas through the pebble array will be important. Such models have so far been developed on the basis of "intuition;" thus experiments are needed. An additional difficulty for predictive techniques near a converging outlet region is that, as the radius of the bed decreases, the effects of the surrounding wall increase relatively [Cheng and Hsu 1986]. In the interior an isotropic approach seems to be appropriate, but near the boundaries of the pebble bed the porosity becomes strongly nonisotropic. Here the average porosity increases and the flow resistance decreases, resulting in "channeling" along the boundaries. Little is known about the current macroscopic models that take account of this effect (e.g., what is their accuracy?). A similar problem becomes important in the context of flow with heat transfer, because the boundary between pebble bed and a plane wall may act as an insulation layer. The combined effect of the convergence and wall effect is another unknown that needs study. Measurements are needed to examine the validity of any models employed and their related constitutive theories. The INEEL MIR flow system is ideal to investigate these difficulties in detail.

The work planned for the period FY-06 through FY-08 will develop experiments to provide an understanding of converging porous flows, to assess codes predicting the flow in pebble bed reactors and to model physically the outflow from pebble beds. Experiments using refractive-index-matching have already been employed to examine flow and particulate transport in saturated, homogeneous porous media [Johnston, Dybbs, and Edwards 1975; Cenedese, Cushman, and Moroni 2002]. Usually, the experiments have been small. Of interest for the current application are the results of Dybbs and Edwards [1984] who used flow visualization and LDV for flow through hexagonal packed beds of spheres with refractive-index-matching -- and defined four successive, distinct flow regimes. Despite the wide range of previous work on flow and heat transfer in porous media and in pebble bed reactors, a number of important scientific and technological needs remain in order to design and operate NGNP reactors using pebble bed technology confidently and safely:

- Quantify the effect of side walls on the reduction of coolant flow resistance due to increased porosity and the regular arrangement of the spheres (for flow channeling),
- Evaluate the effects of coolant flow convergence (and divergence) on the stability of coolant flow in pebble beds,
- Evaluate effects of converging walls of pebble beds on flow, heat transfer and chemical reactions
- Physically model typical outflow regions from pebble beds and obtain benchmark measurements

The objectives of the INEEL laboratory study will be to answer the experimental aspects of the needs identified above for treating the coolant flow and transport through characteristic VHTR pebble bed geometries—including investigation of a larger range of Peclet and Reynolds numbers than has been previously obtained but are badly needed. The experimental model and apparatus will be designed and fabricated using a variety of new and existing equipment. The MIR test section would be filled with pebble beds of differing permeabilities consistent with scaling considerations. For refractive index matching, the current combination of quartz and light mineral oil will be employed.

Flow through the model will be provided by an existing auxiliary flow system with its own temperature control system for adjusting the refractive index of the oil passing through the porous media. For optical access, oil flow is maintained in the main loop and its temperature control system is employed to ensure that the refractive indices are matched from the test section walls to the model wall. Measurements will be obtained by particle image velocimetry (PIV), laser Doppler velocimetry (LDV), by video and/or camera recording and by Particle Tracking Velocimetry (PTV) with a Moving Particle Tracking system (MPT). Instantaneous velocity components will be measured by LDV at fixed positions. Three experiments will be constructed, two using the MIR flow system and one with airflow for initial evaluations:

1. MIR measurements of converging flow in pebble beds
2. Pressure distribution data in air flow corresponding to Experiment (1)
3. Physical modeling of typical outlet region by MIR benchmark measurements

Typical results will include time-resolved, point-wise distributions of the mean velocities, U, V, W, and their Reynolds stress components for assessment of proposed CFD codes. The applicability of using oil (or any Newtonian fluid to represent the chosen reactor working fluid), instead of helium, is confirmed by using equivalent ranges of non-dimensional parameters in non-dimensional forms of the field equations. Because all of the fluid behavior occurs at low Mach numbers, the flow is treated as incompressible.

4.3.3 Upper and Lower Plenum Fluid Behavior Experiments

A number of experiments focused on analyzing typical behavior in the lower and upper plenums are summarized in the following paragraphs. Although the specific lower and upper plenum geometries have not been specified yet, it is known that the reactor will most probably have both a lower and upper plenum. Also the final lower and upper plenum geometry designs, whether the reactor is a pebble-bed or a block-type configuration, will probably have features similar to the baseline design that is being used to define the preliminary experiments. That is, the upper plenum will probably have accommodations for inserting control rods and also a number of flow channels will be available for the working fluid to proceed through the core. Although the aspect ratio (height to diameter) will probably be different than that chosen, it is probably safe to assume the flow making the transition in the plenum to the core will not be developed flow. For the lower plenum, there will probably be various flow obstructions in place whose function is to provide structural support for the core hardware and the flow will likely exit through a duct such that the core flow will be required to shift direction from downward to a horizontal direction. Finally, the flow characteristics will likely be quite different on one side of the lower plenum versus the other side due to the siting of an exit duct on the side of the reactor vessel. Thus, the validation data produced in the experiments described below are envisioned as scalable, to a degree, to the final design geometry.

Lower plenum fluid dynamics experiments. Accurate predictions of the thermal mixing in the lower plenum are needed to predict the temperature distribution of the core outlet duct and its material behavior. Due to the variation of power and the heat generation across and along the core, the jets from the cooling channels into the plenum may vary substantially in temperature. If the turbulent mixing of these flows is incomplete, high temperature gas may impinge on lower plenum surfaces and/or the entrance of the outlet duct causing potential structural problems. Non-uniformity of the temperature distribution in the gas may also adversely affect the intermediate heat exchanger or high-pressure turbine. The geometric transition from the circular cooling channels in the core to the lower plenum is complex, as is the configuration of the lower plenum itself with its array of posts supporting the core. Plenum studies are pertinent to pebble bed reactors as well as prismatic block versions although the details of the lower plenum designs can be expected to differ. Hence, reliable, accurate data are needed to validate predictive techniques for the flow and mixing in the plenum.

To partially address the above concerns, an experiment to study fluid dynamics and mixing in a lower plenum is being designed and will be used to generate data by the end of FY-05. The experiment design is being undertaken by considering the flow in the lower plenum to consist of multiple jets into a confined, density-stratified, cross-flow with obstructions. Since the flow converges ultimately to a single outlet, the hot jets encounter different cross-flow velocities depending on their locations relative to the outlet. Possible flow routes in the lower plenum of a typical block reactor design concept are shown in plan view in Figure 4-16. As described in the previous section, the choice of experimental working fluids may be postponed since whatever the fluid chosen its behavior is related to the NGNP reactor working fluid using similar operational ranges as determined using non-dimensional parameters in the applicable non-dimensional field equations.

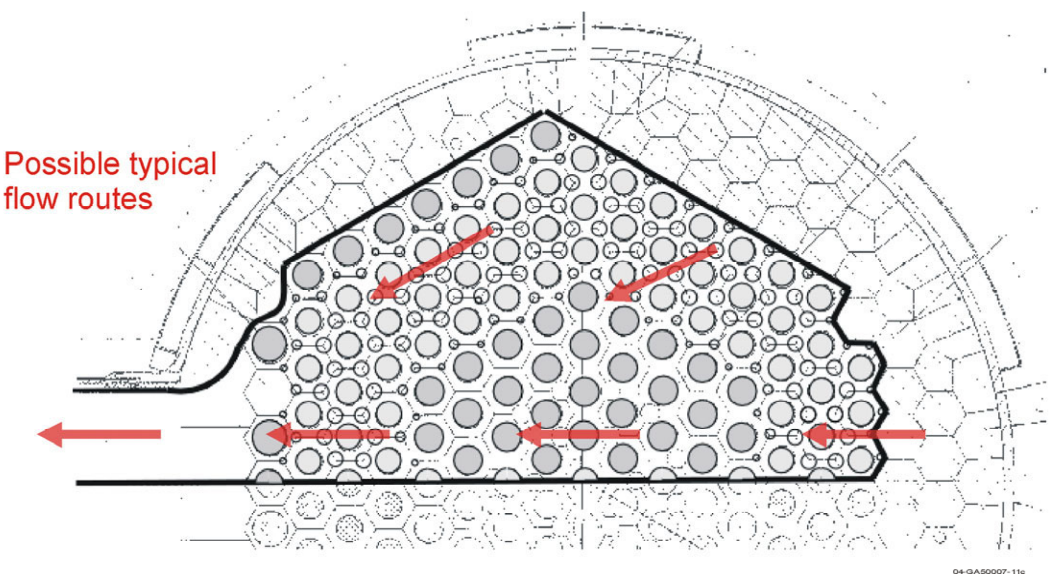


Figure 4-16. Examples of some possible flow paths in the lower plenum of a typical VHTR block reactor concept.

The large circles represent support posts while the smaller ones identify locations of the inlet ducts from the cooling channels in the active core. Some bypass flow can also be expected to enter the lower plenum after passing vertically between the hexagonal graphite blocks both in the core and the reflectors. The arrows give intuitive examples of some paths the flow could be expected to take through the lower plenum from the far side to the outlet duct. In some regions the flow pattern would be comparable to cross-flow over a triangular array of rods as in a shell-and-tube heat exchanger; in other locations the flow may tend along passageways formed by parallel rows of posts. The flow rate (or Reynolds number) increases from the right side of the figure to the left as more incoming jets participate.

The jets furthest from the outlet essentially exhaust into stagnant surroundings between the adjacent posts with the exception of the flow that they induce. If a "hot channel" region is exhausted via one of the furthest jets, there is concern that its impingement on the floor of the lower plenum may be too hot for the insulation layer protecting the metallic core support plate below. On the other hand, the last row of jets before the outlet encounters cross-flow from all the other jets. The "hot streaking" issue pertains primarily to the entrance of the hot outlet duct to the turbomachinery. If a "hot channel" region exhausts through one of the last jets before the outlet duct, there is concern that it may not mix (and thereby cool) sufficiently before flowing along the metallic outlet duct.

Data from a number of separate effects experiments appear to be available for initial assessment of the capabilities of CFD codes to handle some individual phenomena in a lower plenum [Schultz, Ball, and King 2004]. The proposed INEEL MIR studies are aimed at taking the next step, i.e., providing databases for key coupled phenomena, such as jet interactions with nearby circular posts and with vertical posts in the vicinity of vertical walls - with near stagnant surroundings at one extreme and significant cross-flow at the other.

Current prismatic NGNP concepts have been examined to identify proposed flow conditions and geometries over the range from normal operation to decay heat removal in a pressurized cooldown [McEligot and McCreery 2004]. Approximate analyses have been applied to determine key non-dimensional parameters and their magnitudes over this operating range. Flow is expected to be turbulent with momentum-dominated turbulent jets entering. An approximate analysis was conducted to estimate when a temperature gradient will stabilize a horizontal turbulent channel flow, thereby leading to reduced thermal transport near the upper surface. Initial conclusions are that:

- Buoyancy influences are probably not important at full power,
- Buoyancy is important at reduced power (ten per cent), at the side of the lower plenum away from the outlet but not near the outlet.

Thus, experiments without buoyancy effects will provide useful benchmark data for assessing CFD codes for some lower plenum flow conditions.

The purpose of the fluid dynamics experiments is to develop benchmark databases for the assessment of CFD solutions of the momentum equations, the scalar mixing, and the turbulence models for typical VHTR plenum geometries in the limiting case of negligible buoyancy and constant fluid properties. The MIR facility and a sketch of the lower plenum model to be used in MIR are shown in Figure 4-17. Using optical techniques in the MIR facility such as laser Doppler velocimetry (LDV), measurements can be obtained in the complex passages anticipated in the VHTR designs without disturbing the flow. The refractive indices of the fluid and the model are matched so that there is no optical distortion. The large size provides good spatial and temporal resolution. This facility has already

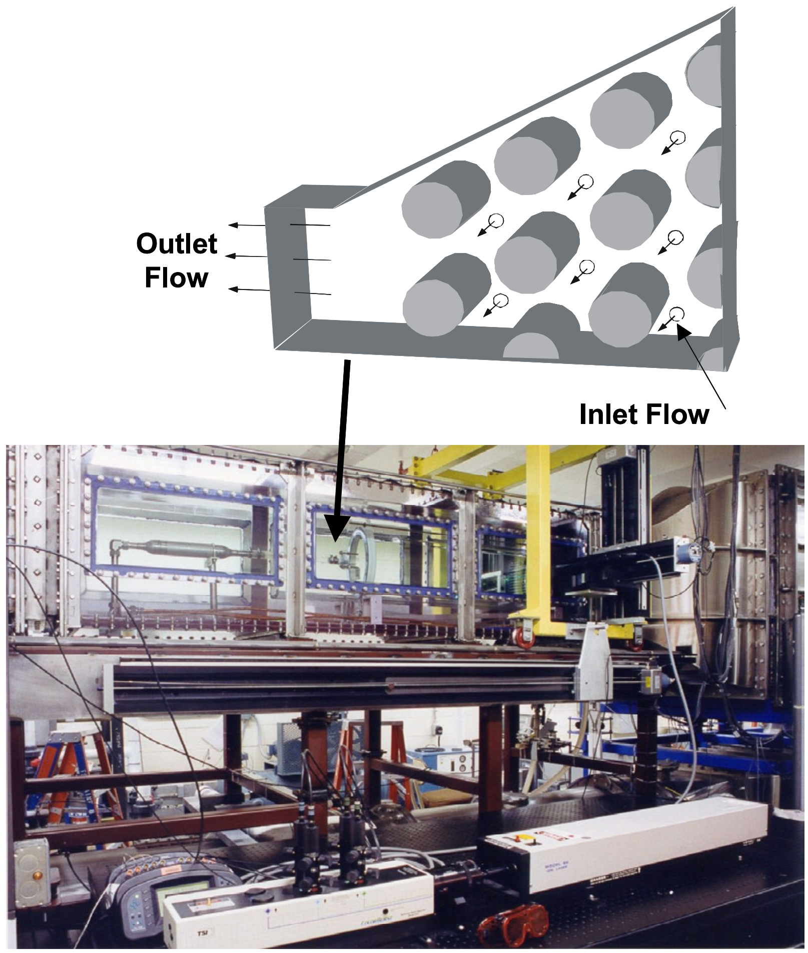


Figure 4-17. Matched-Index-of-Refraction flow system and a conceptual model design to study important flow features in a VHTR lower plenum.

been used to obtain velocity / turbulence data in scaled fuel channels for a VHTR concept [McCreery et al. 2003] and an SCWR concept [McEligot et al. 2003].

McEligot and McCreery [2004] described the general characteristics of a typical VHTR lower plenum and, consequently, the features desired in idealized experiments aimed at assessment of CFD codes for VHTR applications. The first MIR lower plenum experiment will model flow across an array of posts—as near the outlet duct or in line with the duct centerline at the opposite side of the reactor -- as indicated by near horizontal arrows in Figure 4-16. A plan view is shown in Figure 4-18 with open circles representing the locations of jet inlets and cross hatching indicating the support posts, approximately to scale. With cross-flow from the right, flow is simulated from across the central region of the lower plenum (below a central reflector) into the last rows of jets from the active core. For the region furthest from the outlet, a solid wall can be inserted at the position indicated by the dashed line. In this case, flow would be completely provided by the jets as at the simulated location. Simulated plenum dimensions will be based on geometrical scaling of a current prismatic VHTR concept. The *second* MIR experiment will simulate flow aligned between a row of posts as suggested by the inclined arrow in Figure 4-16.

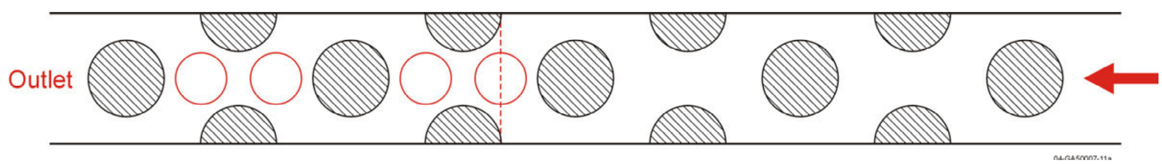


Figure 4-18. Conceptual design of model for the anticipated first MIR lower plenum experiment (plan view).

The experiments will be performed using a state-of-the-art particle imaging velocimetry (PIV) system, to be installed during FY-05. Measurements will be obtained with this PIV system, by laser Doppler velocimetry (LDV) and possibly by three-dimensional particle tracking velocimetry (3D-PTV) with a moving particle tracking system. If feasible, preliminary flow visualization of mixing patterns will be obtained by micro-bubble injection as part of the shakedown tests for the model. Both LDV and PIV (and PTV) have advantages. They may be considered to be complementary approaches. Visualization of mean flows and instantaneous measurement of two velocity components should be available via the PIV system. For flow visualization and for measuring mixing, the PTV should be useful. LDV gives time-resolved measurements. With PIV (and PTV) one needs many realizations to deduce means and the higher-order moments. The LDV technique is applied a point at a time; 3-dimensional measurements are obtained by traversing (taking profiles) in three directions in turbulent flows that are steady in the mean. Provided the application can be considered to be at least quasi-steady (residence times quicker than other response times), useful data for assessment can be obtained from such steady flows. Typical results will include time-resolved, point-wise distributions of the mean velocities and their Reynolds stress components. The LDV time series will also be available for spectral and wavelet analysis to investigate potential shedding of eddies from the posts. Further background and information on this experiment are presented in the report by McEligot and McCreery [2004].

It is planned that the first model (Figure 4-18) will be designed, fabricated and installed in the MIR test section and that measurements will commence during FY-05. Although this test section is narrow, wall effects will be minimized by injecting jets from the side and by having a sufficiently large test section width to pitch ratio. Further measurements and documentation will be conducted during FY-06 and the second model will be designed and fabricated as well. The key deliverables will be documented databases for CFD code assessment as described above. The approach of this experiment will also be employed for the later experiments on exit flows in pebble beds, on geometric transitions of plenum inlets / outlets and on prototypical geometries identified in the NGNP design and licensing phases.

Heated flows in lower plenums: A follow-on experiment will treat heated (buoyant) jets entering a model which simulates key geometric aspects of a lower plenum; an apparatus with gas flow will likely be employed for this case. Preliminary analyses indicate that injection of buoyant jets could lead to thermal stratification of hot helium near the upper surface of the plenum in the region away from the outlet, under reduced power operations and possibly under full power. The objective of this experiment is to provide benchmark data to assess CFD codes that include treatment of buoyancy forces and gas property variation in predicting thermal mixing.

Two types of conceptual model designs are being considered, some using gas flow and some using water. Density differences between hot and cool jets in designs using gas flow are introduced by varying the gas temperature. Density differences in a water flow apparatus are simulated by varying the density of the injected water with the addition of soluble material, such as salt to water stored in a reservoir, before injection. A scaling analysis indicates that the two methods are equivalent for modeling buoyancy influences in lower plenum flow. Further details of various proposed concepts are discussed in McEligot and McCreery [2004].

A model concept for heated gas flow is shown as Figure 4-19. The posts and lower plenum walls are idealized as cylinders positioned between flat walls, somewhat like the plan view in Figure 4-18. The inlet flow channels, from which the buoyant jets are formed, are represented by tubes attached to the top wall. Hot and cold gases are injected through these tubes at flow rates and temperatures determined from scaling analyses. In this case, all flow in the simulated plenum is induced by the incoming jets similar to the region opposite the plenum outlet in a prismatic VHTR concept.

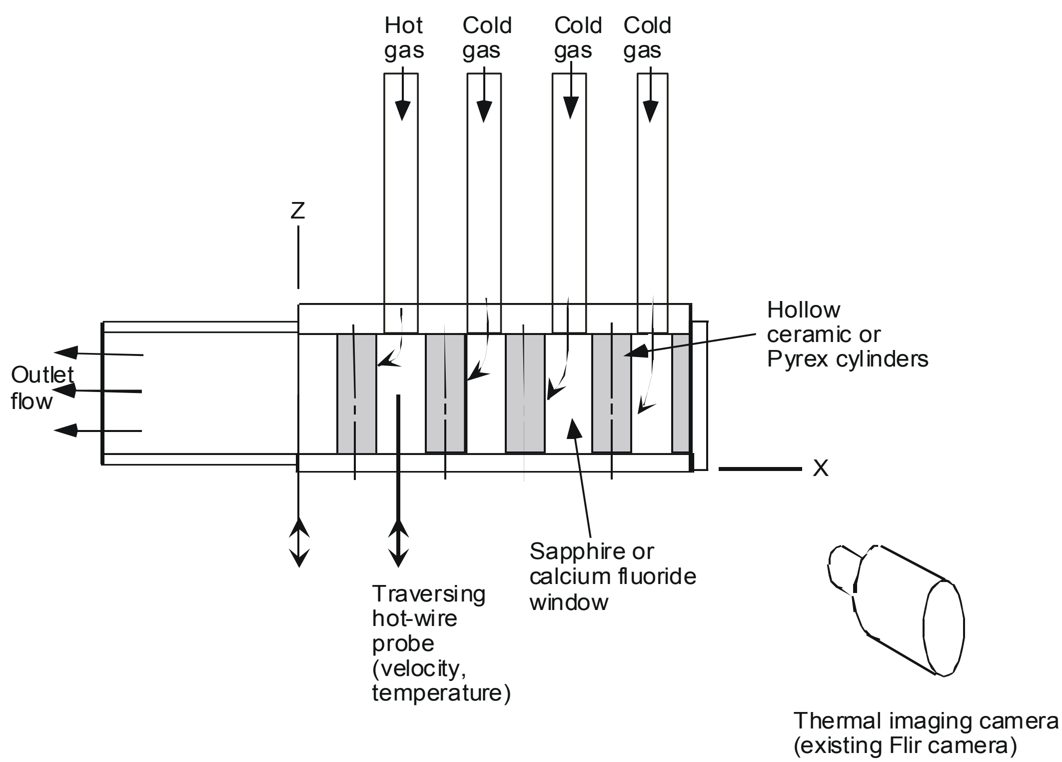


Figure 4-19. Schematic diagram of benchmark experiment to simulate effect of buoyancy on thermal mixing in a lower plenum.

The gas temperature field will be mapped using both a traversing thermal-sensor probe and a thermal imaging camera. The bottom wall of the apparatus will contain multiple access holes for installing probes in various positions. Local mean temperatures and temperature fluctuations will be measured using a traversing "cold"-wire probe (effectively a fast-response resistance thermometer). The front wall of the apparatus will consist of a sapphire window that will transmit visible and infrared light for reception in the thermal imaging camera. The camera will record the thermal images of the cylinder surfaces. The cylinder material will be a thin-wall ceramic or Pyrex. The thin walls minimize axial conduction heat transfer, so that the surface temperature distribution will be approximately equal to the local mean gas temperature. Although maximum gas temperatures will be limited to approximately 500 °C due to material temperature limits, this limitation poses no restriction of the validity of the validation data since the data will still be within the proper range of applicable non-dimensional numbers. Additional measuring techniques under consideration include employing heat pipes to represent the support posts, embedding heat flux gages on the surfaces and using naphthalene to apply the heat-mass transfer analogy to infer heat transfer coefficients [Rhee, Yoon and Cho 2003; Angioletti et al. 2003]. With the latter approach, assessment of CFD codes could be accomplished directly by comparing mass transfer predictions (solutions of species conservation equations) to the data.

The PIV system to be employed in the MIR experiments will also be ideal for mapping velocity fields in buoyant flow experiments with air (or water) flow. The PIV system may be extended to measure velocity and temperature fields (for use in the gas flow experiments) or velocity and concentration fields (for use in the water flow experiments) simultaneously. Velocity plus temperature (or concentration) may be measured by using fluorescent seeding particles and two appropriate cameras with narrow-band filters. One camera, with a filter centered on the laser wavelength, records the velocity information and the other acquires the fluorescence signal. The latter method, termed planar-laser-induced-fluorescence, relates signal strength to either temperature or concentration, depending on the seeding material. The capability of extending the PIV system to include planar-laser-induced-fluorescence for the buoyant mixing experiments will be evaluated during the acquisition of the PIV system.

The studies for FY-05 include finalizing the conceptual design of the competing experiments for examining buoyancy effects in a lower plenum, estimating costs and schedules for each and selecting a path forward. Final design, fabrication and initiation of measurements are planned for FY-06 with most measurements and documentation occurring in FY-07.

Interactions between hot plumes in an upper plenum and parallel flow instabilities. The mixing of hot plumes in the upper plenum of a gas-cooled reactor is of concern during a pressurized cooldown [McEligot et al. 2002]. These plumes come from up-flow in the not coolant channels during natural circulation in the core and may impinge on the reactor vessel upper plenum structure and control rod apparatus causing localized hot spots that may be prone to failure. The flow rates and temperatures of the plumes may be affected by laminar flow instability caused by variations in the viscosity with temperature [Reshotko 1967] at the low Reynolds numbers in these channels and thus may possibly cause flow “choking.” A planned experiment on interactions between hot plumes and parallel flow instabilities will examine this problem.

The envisioned experiment will produce a scaled fluid behavior simulation of plumes moving upwards from the hot core cooling channels, of the natural circulation development in the upper plenum, and of the downward movement of upper plenum inventory into the cooler channels in route to the lower plenum. Sufficient instrumentation will be used to characterize the flow behavior for CFD validation data sets.

4.3.4 Air Ingress-Related Experiments

To provide data required to accurately treat heat transfer and wall friction of air-helium mixtures during air ingress events, experiments are planned during FY-06 and FY-07. For accident scenarios involving air ingress, the heavier air will displace the helium used as the normal working fluid. However, a consequence of this displacement process will be a mixture of the two gases in some components. As the concentration of a gas mixture changes, so do its properties. It is known that the variation of thermal conductivity and specific heat versus concentration will lead to a minimum in the Prandtl number at an intermediate concentration, where the Prandtl number realizes a value between that for liquid metals and common gases. Thus the applicability of typical correlations and turbulence models in these cases are open to question. Taylor, Bauer, and McEligot [1988] have shown that some of the most popular correlations over-predict convective heat transfer for other binary gas mixtures at high Reynolds numbers. Still needed are data for forced and mixed convection in low-Reynolds-number turbulent flows. Graphite oxidation data are available from other sources [Schultz, Ball, & King 2004].

The objective of this simple experiment will be to obtain benchmark data for the preliminary assessment of CFD turbulence models and systems codes for forced and mixed convection in low-Reynolds-number turbulent flows occurring during air ingress or resulting natural circulation. Wall temperatures and pressure drops will be measured for specified wall heat flux and inlet conditions with apparatus comparable to that employed by Taylor, Bauer and McEligot but modified to achieve lower relative heat losses at the lower Reynolds numbers. Experimental procedures employed will also correspond to those of Taylor, Bauer and McEligot. System pressure and test section diameter will be adjusted to provide the necessary range of non-dimensional parameters. With these data, turbulence models may be evaluated by comparison of wall temperature and static pressure distributions. Figure 4-13 shows an example of this type of comparison.

4.3.5 Larger Scale Vessel Experiments

Code development and assessment activities for previous reactor designs have required integral experiments at various scales to verify that small-scale laboratory experiments, experiments using simulated fluids, and experiments at non-rated conditions have been properly scaled for the full-scale plant. This premise holds true for any NGNP design. While some integral data may become available from the HTTR and the HTR-10 research reactors, there will undoubtedly be a data gap when considering measurements needed to validate calculations from coupled CFD/systems codes for the final NGNP reactor geometry. Therefore, a larger scaled vessel experiment will be performed to provide scaled data directly applicable to the final NGNP design and quantify potential distortions from data of small-scale facilities not apparent in the scaling studies.

A highly instrumented, geometrically correct, larger-scale simulator will be constructed consisting of an NGNP upper plenum, core simulator, lower plenum, hot outlet duct and turbine inlet channel. Geometry will be defined by the best available information on the actual design. The scale required will be determined from previous experimental results and available literature and from phenomena expected to occur. Size of the order of 1/4 to 1/3-scale is envisioned with lower pressures and temperatures than design targets. The core simulator may or may not be electrically heated. Overall instrumentation will be sufficient to provide detailed local data for CFD code assessment, as well as global data for systems code assessment. The facility will be capable of simulating both operational conditions and accident scenarios. Issues that can be studied for operational conditions include the influence of various bypass conditions on the system operational envelope, the progression of mixing and turbulence of the helium as it passes from the lower plenum through the hot outlet duct to the turbine inlet and the influence of various lower plenum configurations on the system performance. Accident conditions that can be examined include the influence of natural circulation on the thermal behavior of the system during depressurized cooldown.

4.3.6 Integral Reactor Experiments: HTTR and HTR-10

Presently there are two operational gas-cooled test reactors: the High Temperature Gas-Cooled Reactor—10 (HTR-10) and the High Temperature Test Reactor (HTTR). These experiments are located in Beijing, China at the Institute of Nuclear Energy Technology (INET) and in Oarai, Japan at the Japan Atomic Energy Research Institute (JAERI) respectively. Integral experiments are the only experimental sources that may be able to produce the complex interactions between dominant phenomena identified in the NGNP system specific PIRT. Therefore, the integral experiments are essential for systems analysis and CFD code validation studies. Undoubtedly data from both the HTTR and the HTR-10 will be included in the calculational matrix required for plant licensing by the U.S. Nuclear Regulatory Commission.

Sketches of the two facilities are shown in Figures 4-20 and 4-21 below. Validation studies are needed using the data generated at these facilities to date. In addition, arrangements will be made to enable the NGNP Program to collaborate with INET and JAERI such that specific experiments may be specified that can be linked directly to the NGNP preliminary and final design PIRTs.

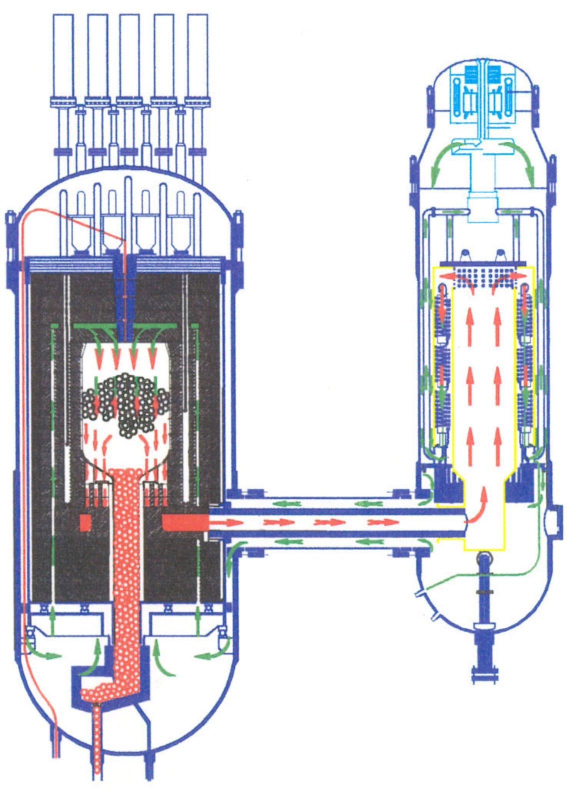


Figure 4-20. Schematic of HTR-10 (from HTGR Proceedings, Beijing, China, March, 2001).

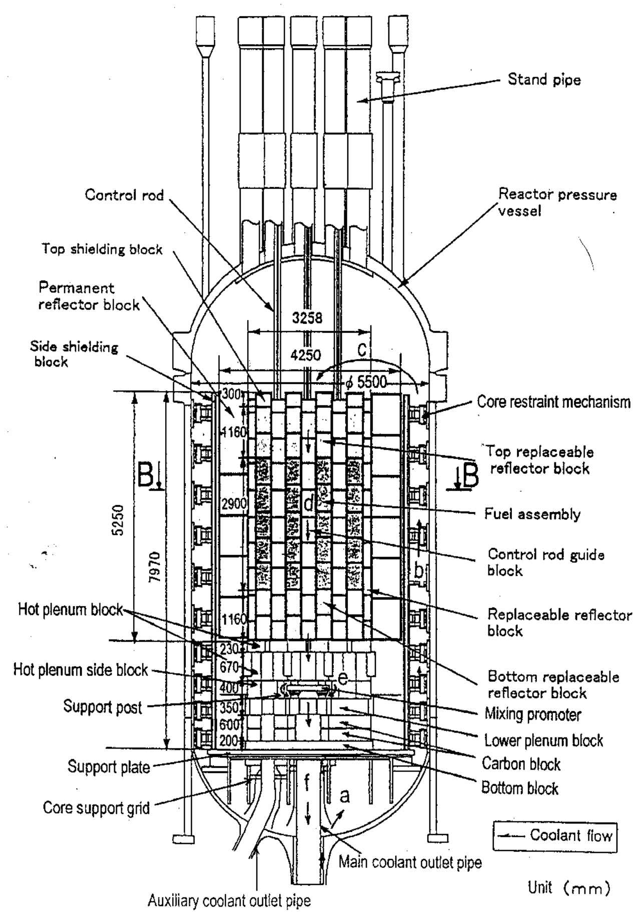


Figure 4-21. Schematic of HTTR.

HTR-10. The HTR-10 is a 10 MW pebble bed high temperature gas-cooled reactor that became operational in 2000. INET plans to perform a spectrum of experiments of essential to the NGNP Project. Among the experiments may be a LOCA, a pressurized conduction cooldown experiment, a rod ejection experiment, and an anticipated transient without scram.

The HTR-10 reactor vessel (see Figure 4-20) is approximately 11.2 m in height and contains a 1.8 m diameter core that is 1.97 m high with ~27000 pebbles. The reactor was designed to operate at 10 MWt. The average power density is 2 MW/m^3 and the core inlet temperature is 250 to 300 °C and the core outlet temperature will range from 700 to 900 °C. Benchmark experiments performed in the HTR-10 are available via the IAEA [Sun & Gao 2003].

HTTR. The HTTR Project is centered on the 30 MWt prismatic engineering test reactor (see Figure 4-21). However, the HTTR Project also has a number of support projects that provide useful data (e.g., the Vessel Cooling System test series based on cooling panels inside a vessel containing heating elements and the heat transfer studies based on the hemispheres heated from below and cooled using natural convection). The Japan Atomic Energy Research Institute's (JAERI) plans, to perform a spectrum of HTTR experiments that may include a LOCA, a pressurized conduction cooldown experiment, a rod ejection experiment, and an anticipated transient without scram.

The HTTR became operational in 1998. The reactor vessel is 13.2 m tall (inner dimension) and has a 5.5 m inner diameter. The core has 30 fuel columns and 7 control rod guide columns. There are 12 replaceable reflector columns and 9 control rod guide columns. The HTTR is fitted with a reactor cavity cooling system (RCCS). The HTTR operates at 4 MPa with a core inlet temperature of 395 °C and outlet temperature of 850 °C [Sikusa 2000]. However, it is known that the HTTR does not have a full set of instrumentation. Thus, additional instrumentation is required to obtain the needed data.

Supporting experiments include a series of 6 tests performed to simulate the heat transfer to the RCCS cooling panels [see IAEA, 2000]. The experiments are summarized in Table 4-4.

Table 4-4. RCCS Experiments: HTTR Project

Experiment	**I**	**II**	**III**	**IV**	**V**	**VIa**	**VIb**
Gas	Vacuum	helium	nitrogen	helium	helium	helium	helium
Presssure (MPa)	1.3×10^{-6}	0.7	1.1	0.47	0.64	0.96	0.98
Power (kW)	13.1	28.8	93.9	77.5	29.7	2.6	8.0
Cooling panel	water	water	water	water	air	air	air

Cooling panels were placed inside a pressure vessel and experiments were performed by varying the gas in the pressure vessel to change the natural convection characteristics; thus Experiment I was performed with a vacuum so no natural convection would occur and the only heat transfer from the heaters to the cooling panels would be radiation. Experiment III was performed with nitrogen and the remainder of the experiments were performed using helium. Also the cooling medium in the cooling panels was run with water for 4 experiments and air with 3 experiments. The power level was changed as shown.

A further example of the types of experiments performed in the HTTR project include a series of experiments making use of a 0.3 m hemisphere that was heated from below while the natural circulation characteristics were measured. This experiment was designed provide validation data relevant to calculating natural circulation in passive systems for CFD and systems analysis software.

4.3.7 Reactor Cavity Cooling System Experiments

Reactor Cavity Cooling System (RCCS) research is essential since the heat transfer from the reactor pressure vessel to the RCCS is a key ingredient in defining the peak core and vessel wall temperatures during postulated accident conditions. Two RCCS experimental efforts are presently underway. The first, at ANL, aims to characterize the heat removal capabilities of an air-cooled RCCS. The second, at the Seoul National University, aims to characterize the heat removal capabilities of a water-cooled RCCS.

ANL air-cooled RCCS. The objective of this task is to acquire the model/code validation data for natural convection and radiation heat transfer in the reactor cavity and the reactor cavity cooling system by performing experiments in the ANL Natural Convection Shutdown Heat Removal Test Facility (NSTF) shown in Figure 4-22. The first task will be to determine the scalability of existing data from the ANL RCCS simulator to a "typical" air-cooled VHTR RCCS design. The scaling studies will identify the important non-dimensional parameters for each separate-effects study. Based on the results of the scaling study, the range of experiment conditions will be determined as well as the appropriate experiment scale and appropriate fluids to be used that most effectively simulate full-scale system behavior.

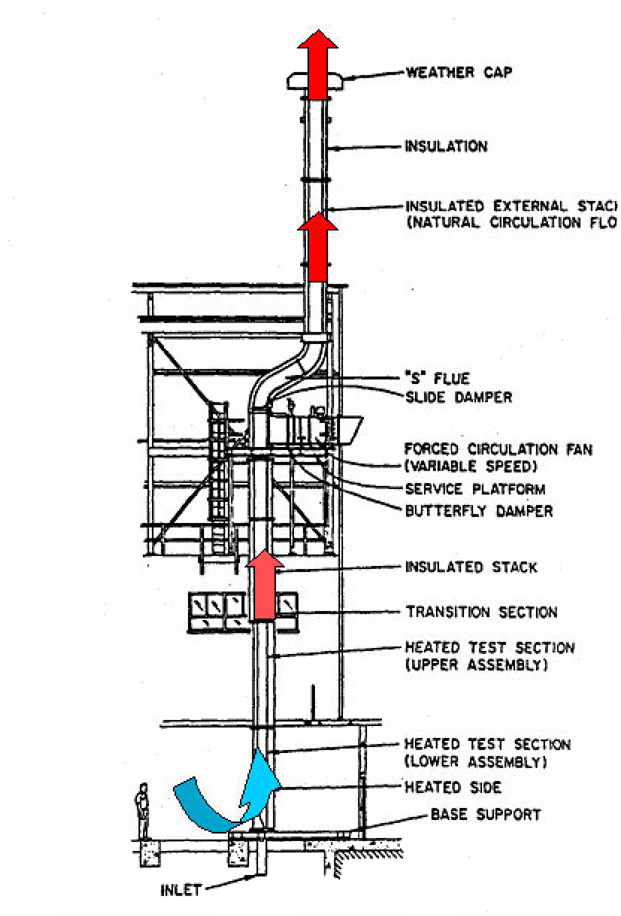
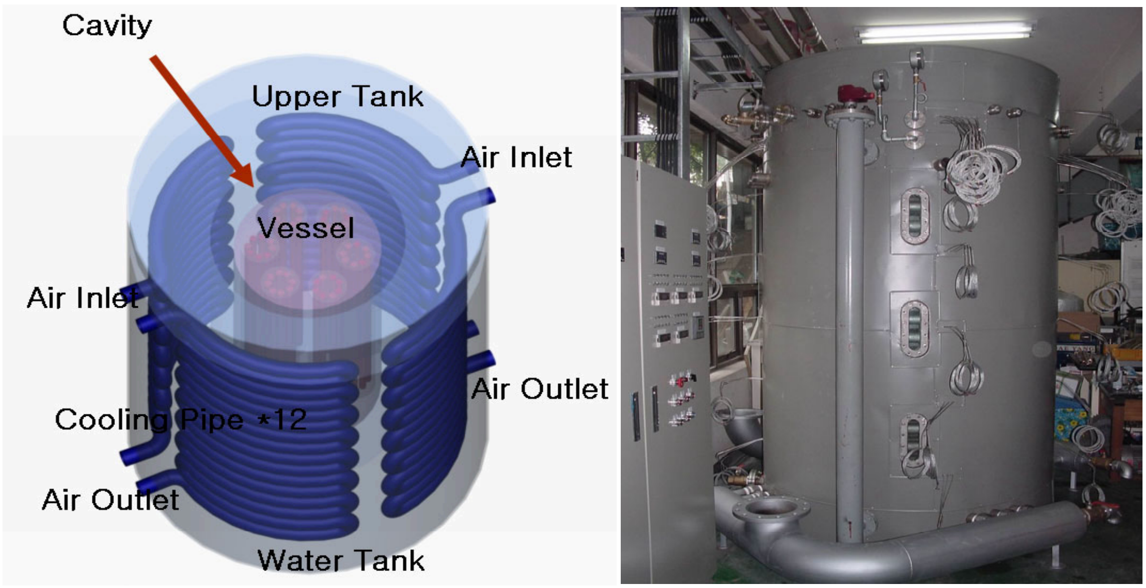


Figure 4-22. Schematic of ANL Natural Convection Shutdown Heat Removal Test Facility.

The R&D will include the identification of RCCS design candidates from both the pebble-bed and prismatic options and the range of thermal-hydraulic conditions for normal operating and accident events. An instrumentation strategy will be developed to assure that adequately detailed velocity and turbulence profiles are obtained, as well as surface pressure and/or temperature distributions for the validation of multidimensional simulation tools. Based on the results of these feasibility studies, a detailed engineering modification plan for the ANL RCCS facility will be developed. Next, a test matrix will be developed, and the indicated test program will be performed. The ANL RCCS experimental results will capture key phenomena expected to be present in the RCCS and provide data of sufficient resolution for development and assessment of applicable CFD and system codes.

Seoul National University water-cooled RCCS. A water-cooled RCCS design may be preferred since its heat removal capability is larger per unit heat transfer area than a comparable air-cooled design. Hence a water-cooled design would be more desirable if a high-pressure containment is required for the NGNP instead of a low-pressure confinement system.

The SNU RCCS facility consists of three parts: the reactor vessel, an air cavity, and a water pool (see Figure 4-23). The SNU experiments are being performed using various gas mixtures in the gap and with various water pool elevations. The temperatures on the various surfaces are measured together with the surface emissivities and water pool characteristics (temperature as a function of position, elevation, etc). Heat from the reactor vessel is transferred to the RCCS by radiation, natural convection, and conduction. The data provided by these experiments are the basis for validation CFD calculations specific to the behavior of water-cooled RCCS.



a. Schematic of test facility b. Photograph of test facility

Figure 4-23. SNU water-cooled RCCS experiment.

4.3.8 Thermal-Hydraulic Design Methods Development, Validation, and Analysis - Introduction

The thermal-hydraulics design, performance analysis, and ultimately the licensing of the NGNP will require the use of validated computer codes for modeling the reactor's behavior during normal operation, anticipated transients, and accident conditions. The modeling strategy chosen for this effort is to make use of both thermal-hydraulic systems analysis codes and computational fluid dynamics (CFD) codes. The reference codes chosen are the RELAP5-3D systems code and the Fluent and STAR-CD CFD codes[e]. However, other codes, such as GRSAC, Abaqus, and NPHASE will be used to supplement the reference codes.

A systems analysis code is needed to model the integrated behavior of the entire NGNP system, including the interactive coupling of the reactor with the hydrogen and power producing components, including the intermediate heat exchanger, turbine, compressor, reheaters, etc.

CFD software is needed to analyze or qualify simulated fluid behavior wherever two- or three-dimensional fluid behavior is expected, particularly in plenums and cavities. Regions of applicability for the NGNP include the upper and lower plenums, the hot duct and the intermediate heat exchanger or

[e] RELAP5-3D includes all the working fluids presently being considered for the NGNP together with their associated constitutive models. Fluent is a commercial CFD code that was selected on the basis of its extensive usage and validation history and the commitment of its vendor to the NGNP program. During FY-02 through FY-04 a coupling capability was implemented to link RELAP5-3D models to a Fluent model. Fluent is being used by INL and STAR-CD is being used by ANL.

turbine inlet region as well as the reactor cavity cooling system. Other NGNP regions will be identified during the upcoming PIRT studies.

Although two commercial CFD reference codes (Fluent and STAR-CD) are presently being used and a university-developed[f] or a national laboratory-developed[g] CFD code may also be used, it is suspected that none of them will meet all of the NGNP analysis requirements and thus some modifications will be required. Consequently, a three-track approach will be used to meet the CFD analysis needs for the NGNP:

Track 1: validation of currently available CFD software,

Track 2: modification of existing tools as necessary, and

Track 3: pursuit of R&D to obtain more efficient and effective simulation tools that may take several years to mature.

The near-term thermal-hydraulics tasks follow the first track: validating existing tools. As the CFD tools are validated, it may become necessary to add new turbulence models or pursue other modeling strategies, such as Lattice-Boltzmann, Large Eddy Simulation (LES) or Direct Numerical Simulation (DNS), thus following Track 2. Track 3 is designed to ensure that needed simulation tools will be available in the future that are more efficient and capable than existing tools. This approach is outlined in Figure 4-24.

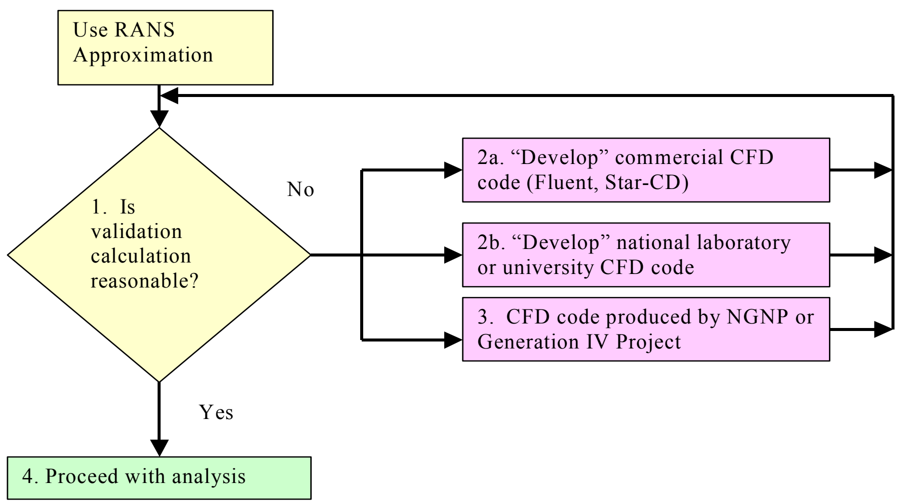


Figure 4-24. Approach for achieving validation objective for CFD.

Following the strategy outlined in the above two paragraphs, the commercial CFD codes Fluent (at INL) and STAR-CD (at ANL) will initially be validated and developed for the near-term thermal-hydraulics tasks (Tracks 1 and 2). As deficiencies are isolated that cannot be addressed using the commercial CFD codes, experimental CFD software such as NPHASE and CFDLib will be tested and used to

[f] An advanced next-generation CFD solver for both single-phase and multiphase flows developed at Rensselaer Polytechnic Institute. A new upgraded version of this code is currently under development via the sponsorship of the U.S. Nuclear Regulatory Commission for future reactor safety analyses of next-generation reactors [Antal, et al 2000].

[g] CFDLib is the Los Alamos Computational Fluid Dynamics Library. This is a collection of codes. The CFDLib collection is a repository for all the numerical methodologies developed in the Fluid Dynamics Group (T3) of LANL's Theoretical Division. For example, the MAC method (due to Harlow & Welch), the ALE method (Hirt et al.), the multifluid ICE method (Harlow & Amsden), and the FLIP method (Brackbill & Ruppel) are all schemes that reside in the CFDLib collection. In recent years the CFDLib collection has been made into a sort of 'open-source' project, with contributors from all over the academic world as well as many other divisions of LANL and other US National Laboratories. For multiphase flow, the original capabilities of K-FIX (Rivard & Torrey) are contained in CFDLib [Kashiwa et al. 1993, 1994].

analyze the more difficult tasks.

Because the spectrum of turbulent mixing behaviors that will be present in the plenums and flow passages are key phenomena that require evaluation, a significant research effort is aimed at the identification of the proper model and approach. One of the more common approaches is based on taking an average of the incompressible Navier-Stokes equation to obtain the Reynolds-Averaged Navier-Stokes equations (RANS) as described in Speziale & So [1998]. A newer approach toward predicting the ensemble average of the fluid velocity is called large-eddy simulation (LES) "…in which the Navier-Stokes equations are 'filtered' instead of averaged. This generates equations for the large flow scales yet uses a 'subgrid' model to capture the effect of the smaller scales" [Bernard et al. 1998, p. 13-3]. Finally, the direct numerical simulation (DNS) approach of representing the Navier-Stokes equations enables all turbulence scales to be resolved.

The key technical issues identified by the first-cut PIRT, as summarized in Table 4-3, are the basis for defining the code development, validation, and analysis R&D program. The following sections are organized on the basis of the five key technical issues listed below:

1. Core heat transfer
2. Lower and upper plenum coolant flow
3. Reactor Cavity Cooling System
4. Air ingress and Fission Product Release
5. Integral system behavior

4.3.9 Core Heat Transfer Model Validation, Development, and Analyses

Both systems analysis and CFD software will be used to analyze the core heat transfer. CFD is presently being used to calculate the maximum coolant jet temperature into the lower plenum for a baseline prismatic reactor. Systems analysis is used to analyze the distribution of the helium in the core and also the influence of the bypass on the overall core pressure distribution.

Convective heat transfer during Normal Operation. During normal operation a non-uniform power and flow distribution in the core will give rise to "hot channels". Helium, as a working fluid, experiences an increase in viscosity that is proportional to the absolute temperature raised to the 0.7 power. At the same time, the heat transfer coefficient decreases. Hence, helium flow through hot channels in the core region (whether a prismatic or pebble-bed core) will tend to decrease, while the flow in cooler channels will tend to increase due to this effect. Analyses are required to determine the maximum fuel temperature during normal operation, the maximum variation in channel outlet temperatures exiting the core into the lower plenum, and the effect of redistributing the flow between the channels. Also, analyses are required to study the behavior of variations in the specified design—such as inlet orificing.

CFD analyses, already underway in FY-05, are aimed at investigating the influence of the turbulence in the hottest cooling channel on the heat transfer and hence the subsequent exit temperature. The maximum exit temperature is instrumental in determining the potential for local hot spots in the lower plenum and various structural components as the gas moves to the intermediate heat exchanger or turbine inlets. Also, the potential for "hot-streaking" at the intermediate heat exchanger or turbine inlets is linked to the peak-temperature coolant jets that enter the lower plenum. Although analyses have been performed in FY-04 and -05, additional analyses will be performed in subsequent years to evaluate additional geometry configurations, geometry/manufacturing dimensional uncertainties, and conditions once the CFD software are better validated. Experimental data to be used for validation include heated experiments (section 4.3.2) to evaluate the channel behavior under conditions where asymmetrical heat transfer loads are imposed on cooling channels (FY-06 & -07).

Systems analysis calculations to consider the redistribution of coolant channel flow as a function of the local peaking factors (i.e., hot channel flow) and variations in bypass flow as a function of core life will be performed for the various NGNP reactor design stages. These calculations are necessary to establish the temperature characteristics of the system and the environmental losses during operational conditions. The core heat transfer will be considered in conjunction with the system calculations described in Section 4.3.13—including the effects of fuel depletion on power distribution and control rod insertion.

Convective heat transfer during PCC. Core flow behavior following a PCC event will be accompanied by internal vessel recirculation with flow upward through the core in hotter coolant channels and downward through the core in cooler coolant channels and bypass paths. Natural convective heat transfer within the core will include several regimes not currently available in RELAP5-3D. Accurately calculating core fuel temperatures will depend on this convective heat transfer as well as the conduction and thermal radiation that will transfer heat out of the vessel to the RCCS.

CFD validation calculations will be performed based on available mixed convection data [McEligot, Magee, and Leppert 1965; Perkins and McEligot 1975; Reynolds 1968; Shumway 1969; Vilemas and Poskas 1999; Bae et al. 2004] to demonstrate the capability of the CFD tools to adequately calculate the appropriate fluid and heat transfer behavior. Following validation, a CFD code may be used during FY-07 to evaluate the core heat transfer in conjunction with the flow distributions in the lower and upper plenums to calculate the potential for localized hot spots in the vessel upper head and control rod apparatus (see Section 4.3.10).

Also, the correlations developed on the basis of available mixed convection data [McEligot, Magee, and Leppert 1965; Perkins and McEligot 1975; Reynolds 1968; Shumway 1969; Vilemas and Poskas 1999; Bae et al. 2004] will be evaluated for applicability and installed in the systems analysis software. The validation calculations will demonstrate the capability of RELAP5-3D to adequately represent these mixed convection regimes (FY-05). Following confirmation of the code's capability to perform such calculations RELAP5-3D will be coupled with a CFD code to enable the modeling of the core flow into the upper plenum and then down through the cooler channels to the lower plenum.

Axial and radial conduction during PCC and DCC. Decay heat removal following a PCC or DCC event will be accompanied in part by axial as well as radial heat transfer within the core. The decay heat removal rate is a key ingredient in determining the peak fuel temperature and the peak structural temperatures in the reactor vessel. Hence confirmation of the software's capability to calculate this behavior is crucial. Therefore, this R&D consists of four parts: (i) revision of the heat structure modeling capability of RELAP5, (ii) validation calculations based on the expected behavior of a prismatic reactor during a conduction cooldown event, (iii) validation calculations based on the AVR experimental data, and (iv) validation calculation based on the Sana I experimental data.

Revision of existing RELAP5-3D heat structures. RELAP5-3D only calculates heat transfer between heat structures in the radial direction (via contact conductance as well as thermal radiation). In either the prismatic or pebble bed core design, the core structure will be modeled using a series of RELAP5-3D heat structures. Therefore, this task will implement the same heat transfer mechanisms in the axial direction (FY-06).

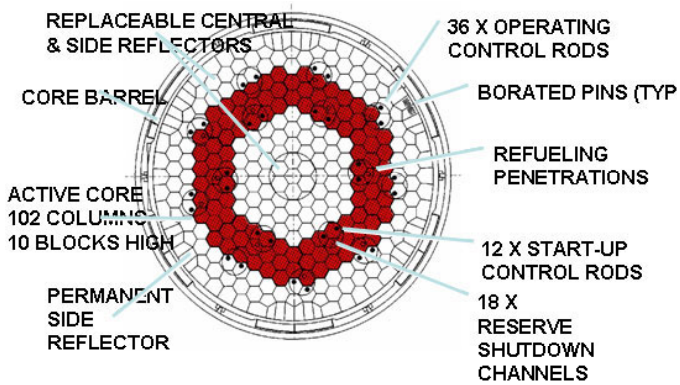


Figure 4-25. Prismatic core layout.

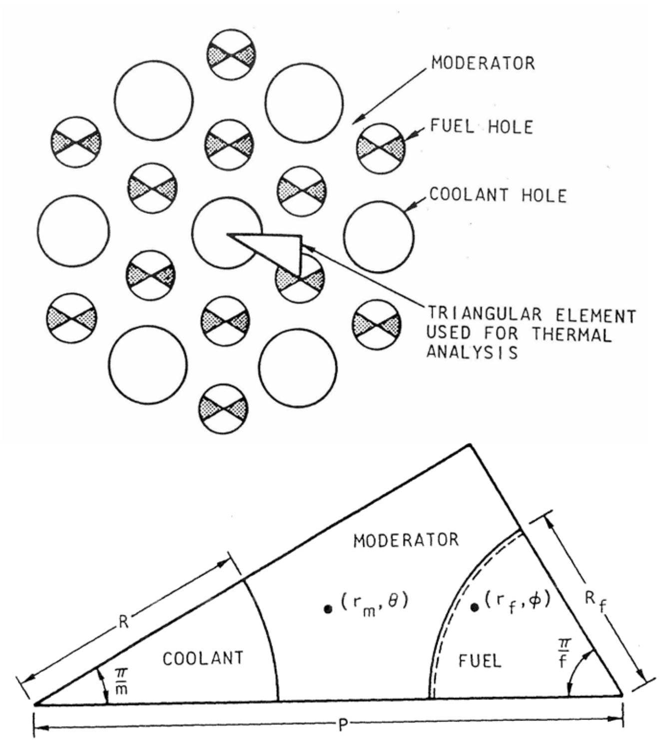


Figure 4-26. Prismatic block reactor core primitive.

For the prismatic block core designs (an example of which is shown Figure 4-25) this modeling approach assumes each graphite block has a uniform temperature, which should prove adequate for modeling core behavior during PCC and DCC events. However, a more detailed model is needed to capture the elements of the core geometry and predict the temperature response at the smallest scale, called the "primitive." Figure 4-26 shows the primitive generally used for calculating heat transfer in prismatic block reactors. The primitive forms the link between a RELAP5-3D model that represents each block in the prismatic reactor and studies that will be done using a code such as Fluent or Abaqus to subdivide each block into a large number of mesh cells to obtain a detailed temperature distribution within the blocks (FY-06).

It is planned to use data from the HTTR facility to perform the needed validation calculations for the prismatic block type VHTR. The HTTR, described in Section 4.3.6, is an operating prismatic test reactor. Both PCC and DCC scenarios are planned for the HTTR in the future. Data from these experiments (if the facility is appropriately instrumented) will enable a thorough validation of the systems analysis code's calculational capability (FY-07).

Data from the AVR German pebble bed reactor will be used to validate core heat transfer modeling in pebble bed VHTRs. The reactor operated from 1967 to 1988 and was the only nuclear power plant that was intentionally subjected to a loss-of-coolant event without emergency cooling [Krüger et al. 1991]. The data includes core power and temperatures measured at various locations in the reactor vessel including the reflectors, reactor shroud, inner vessel, and within instrumented pebbles. The data are considered valuable for qualitative validation of pebble bed modeling. The validation effort and will provide conclusions on the capability of a coupled RELAP5-3D/Fluent model to properly predict pebble bed system behavior (FY-08).

Data from the Sana-I facility shown in Figure 4-27 [IAEA 2000] will also provide for validation of the pebble bed modeling. The test rig was designed to study the heat transfer mechanisms in a pebble bed core filled with 9500 graphite pebbles (diameter = 6 cm) and to provide the basis for validating the models required to determine whether sufficient energy can be transferred to the environment to prevent the fuel from becoming damaged following failure of all heat sinks with a simultaneous depressurization. These data will be used to validate RELAP5-3D through simulations of the experiment (FY-08).

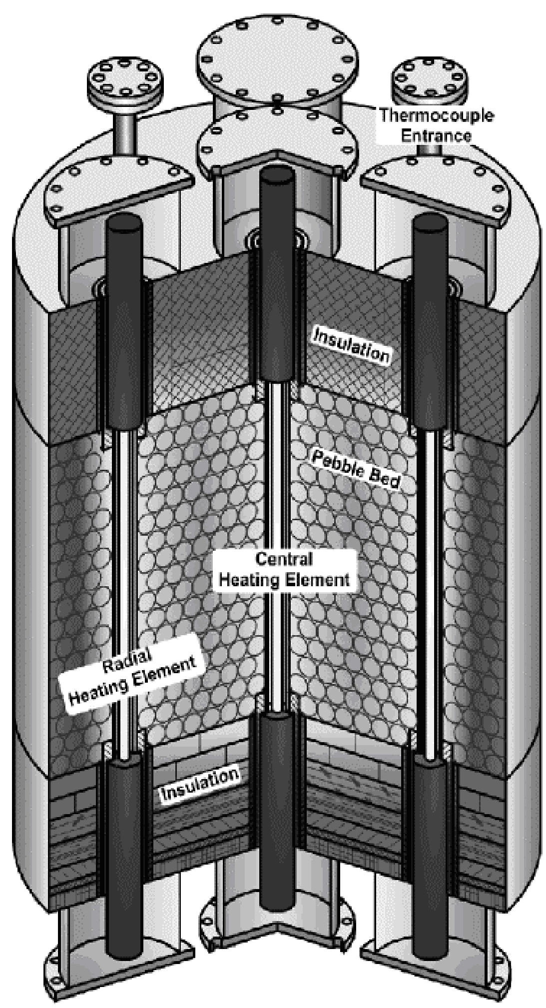


Figure 4-27. SANA-I facility.

Core bypass. Preliminary studies have shown that the amount of core bypass flow in the prismatic core design (due to gaps between graphite blocks in both the core region and the reflectors) can influence peak fuel temperatures during a DCC event [MacDonald et al. 2003]. Follow-on analyses with more refined models need to be performed to understand the tolerance of the design to changes in bypass flow that might occur over time due to structural changes in the graphite blocks. Reports, based on RELAP5-3D analyses, will be issued at each of the design stages documenting this sensitivity.

4.3.10 Upper & Lower Plenum Flow Mixing Validations, Development, and Analyses

Lower plenum. Introduction of the hot jets from the core into the lower plenum creates the potential for the hottest jet streams to impinge on the lower plenum structural surfaces both in regions of low cross-flow and also to move via flow streams passing through low mixing regions into the hot duct and then to the intermediate heat exchanger or turbine inlets. Given that the hottest jets will have exit temperatures well in excess of the average exit temperature, a rigorous, accurate evaluation of the peak temperature jet behavior and the interactions of these jets with adjacent jets and especially their interactions with the lower plenum structural materials are crucial. Figure 4-28 shows a preliminary calculation of the mixing behavior in the lower plenum of a representative advanced gas-cooled reactor design (produced in the collaborative effort between Fluent, General Atomics, and INL). To calculate mixing in the lower plenum, the validation effort will be focused on CFD software.

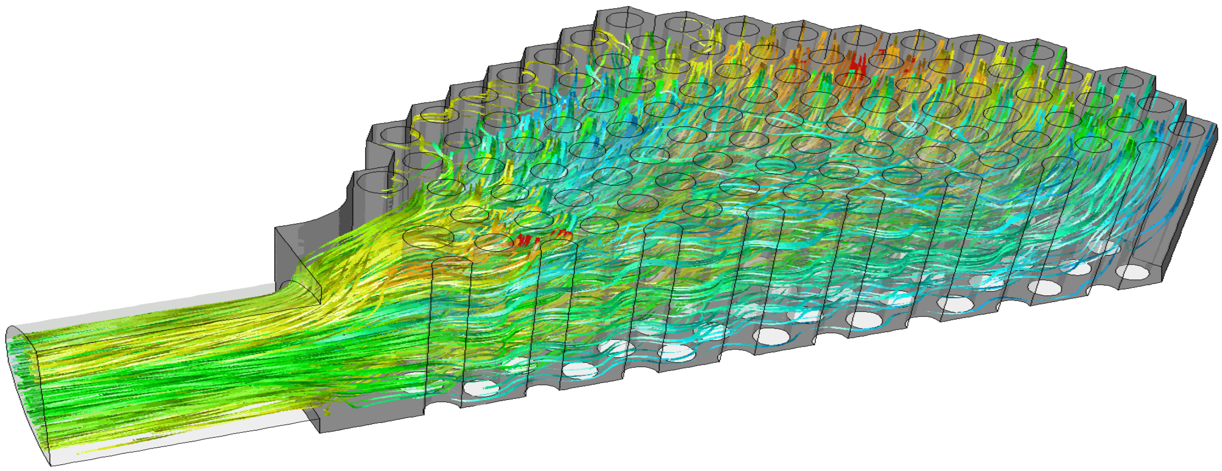


Figure 4-28. Preliminary calculation of mixing in prismatic reactor lower plenum. Courtesy of Fluent, Inc and General Atomics Corp.

To validate the CFD model that will be used to calculate mixing in the lower plenum, a number of experimental validation sets will be used:

- *Benchmark calculations of mixing behavior in the lower plenum:* The turbulent intensity and turbulence as a function of location in the NGNP design lower plenum will be calculated and evaluated. The benchmark calculation will be used as a frame-of-reference for the scaled lower plenum experiments and jet/cross-flow data, e.g., MIR, jet and cross-flow interactions data (FY-05). In addition to serving as an experiment design tool, this calculation serves as a baseline that will be used as a basis of comparison or starting point for the validation. As the CFD code is validated and a more detailed understanding of the mixing behavior is obtained, the baseline calculation will be updated and studied in more detail.
- *Matched-index-of-refraction (MIR) experiments:* CFD models of the MIR experiments described in Section 4.3.3 will be constructed and validation calculations will be performed. The emphasis will be on evaluating the turbulence models and relating the scaled experiments to the fluid behavior postulated to occur in the NGNP design. (FY-05 to FY-07)
- *Heated experiments:* Since the MIR experiments are isothermal they will not reveal the buoyancy contribution due to temperature variations in both the inlet jets and cross-flow will be evaluated. The evaluation and validation calculations will be performed by constructing a CFD model of the experiment (see Figure 4-19). The turbulence intensity and turbulence behavior as a function of location will be compared with data. (FY-05 to FY-08)
- *Benchmark validation studies based on jet and cross-flow data [Schultz, Ball, & King 2004]:* Data describing the general jet and cross-flow phenomena, including the interactions between jets and cross-flow, will be used to perform separate-effects validation studies of the Fluent code (FY-05).

Upper plenum flow. Following a PCC event internal vessel recirculation flow will occur in which helium coolant will flow upward through the core in the hotter coolant channels and downward through the core in the cooler coolant channels and bypass paths. Mixing of these flows will take place in the lower and upper plenums. The recirculation will cause heating of the upper plenum structure and the local temperatures may approach the limiting values for the structural materials or the control rod apparatus. Validation calculations are required to ensure NGNP reactor analyses of the vendor's design give reasonable results. The coupled RELAP5-3D/Fluent model will be used to analyze the PCC event. Validation of the modeling approach will be accomplished by using the data from the planned upper plenum plume experiments (Section 4.3.3) (FY-09).

4.3.11 Reactor Cavity Cooling System Validations, Development, and Analyses

The reactor cavity cooling system (RCCS) is the primary system for transferring the core residual and decay heat to the environment following a PCC or DCC event. Consequently, the RCCS plays an important role in determining the core temperature distribution, the peak fuel temperatures, and the peak structural member temperatures for a given design and power rating. Also, since the RCCS operates in the "null" mode during rated operational conditions with a noticeable fraction of the generated power transferred to the environment, the RCCS has some influence on the plant efficiency and operational conditions (core temperature distributions, etc). Thus, the software tools must be capable of accurately predicting the system behavior under all conditions.

The RCCS design will be dependent in large measure on whether a low-pressure filtered and vented system confinement is allowable or a high-pressure containment is required by the regulators. If a confinement is allowable the RCCS may likely be an air-cooled RCCS whereas if a containment is required the RCCS will likely be a water-cooled RCCS. In either case the largest fraction of the energy transferred from the reactor vessel to the reactor cavity walls occurs through radiation heat transfer. Heat transfer from the walls to the environment may be either through a natural circulation-driven air-cooled duct system or through a water-cooled sleeve. A smaller fraction of the energy transferred to the reactor cavity walls occurs through convective heat transfer via natural circulation of the gases enclosed in the reactor cavity. A validated model is required to analyze the heat transfer via radiation and convection from the reactor vessel to the environment. A conceptual plan view of such a system is shown in Figure 4-29.

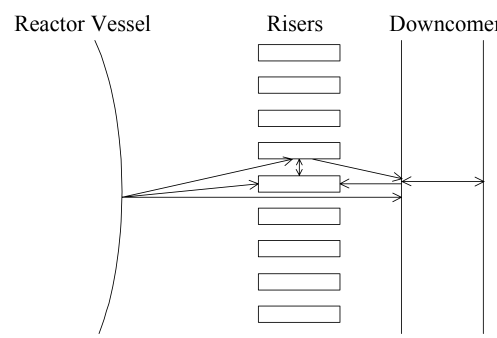


Figure 4-29. Concept for an RCCS design.

To validate a model's capability to calculate the RCCS behavior, the following experimental validations are planned:

ANL air-cooled RCCS (see Section 4.3.3). The key objective of this effort is to validate the capability of the CFD software to reasonably calculate the dominant heat transfer modes in a scaled air-cooled RCCS. The computer code validation effort will consist of development of a model of the experimental matrix, performance of pre-experimental design calculations, performance of blind calculations, and post-calculational analysis. The experimental matrix will cover the NGNP normal operational conditions as well as the expected depressurized conduction cooldown and pressurized conduction cooldown accident conditions. A set of either STAR-CD or FLUENT models for the selected candidate RCCS systems will be defined and the corresponding CFD calculations for a selected set of driving boundary conditions. The STAR-CD or Fluent CFD codes will be validated against the experimental database for the prediction of the RCCS performance under operational conditions and depressurized conduction cooldown and pressurized conduction cooldown accident conditions. In addition, if feasible, the data will be used to formulate a heat transfer correlation for use in RELAP5-3D to enable a more simplified approach to modeling the RCCS (FY-08 to -10).

Seoul National University water-cooled RCCS (see Figure 4-23). A similar validation effort, to that described above for the ANL air-cooled RCCS experiments, will be performed. However, because some two-phase behavior will likely occur in the water-filled thermal sleeve, RELAP5-3D or the CFD code NPHASE will be used (FY-09).

HTTR RCCS separate-effects. Validation will be performed using existing experimental data from the HTTR mockup experiments [IAEA, 2000]. Data were obtained in a series of six tests performed to simulate the heat transfer to the RCCS cooling panels.

For these experiments, cooling panels were placed inside a pressure vessel (Figure 4-30) and the gas in the pressure vessel was varied to change the natural convection characteristics. Vacuum (heat transfer from the heaters to the cooling panels by radiation alone), nitrogen; and helium were used. The cooling medium in the cooling panels was water for four experiments and air for three experiments. The power level was also varied. These data will be used for validation of a coupled RELAP5-3D/Fluent model to correctly calculate and predict RCCS behavior. A report will be issued documenting the validation effort and will provide conclusions on the capability of the codes to properly predict RCCS behavior (FY-08).

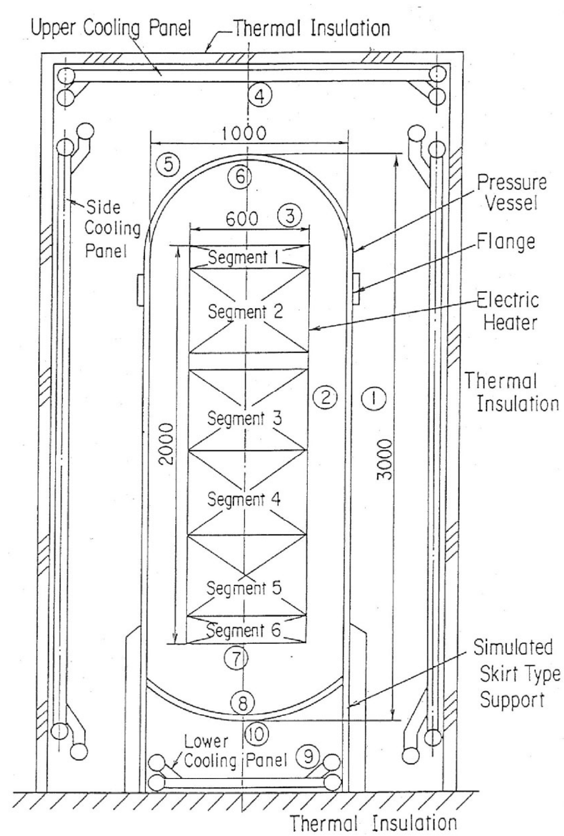


Figure 4-30. HTTR mockup facility.

4.3.12 Air-Ingress Validations, Development, and Analyses

Following a loss-of-coolant accident (DCC), air ingress into the vessel may result in exothermic graphite oxidation, aggravating the core cooldown and resulting in potential core damage. The process driving the exchange of vessel and containment/confinement gases will be molecular diffusion. This diffusion process will involve several gases, including helium, nitrogen, oxygen, carbon monoxide and carbon dioxide.

Air ingress scenarios will be calculated using two approaches: (i) using RELAP5-3D alone and (ii) using Fluent coupled to RELAP5-3D. Because the air that diffuses into the reactor lower plenum will probably do so non-uniformly, a CFD tool will be required to produce a three-dimensional air distribution in the lower plenum using first-principles. Hence it is essential to validate the capability of the CFD software to calculate air diffusing into a plenum occupied by helium. In the event that the air distribution into the lower plenum can be shown to be calculable using a one-dimensional systems analysis, the use of RELAP5-3D alone may be adequate. Thus it is also essential to validate the capability of RELAP5-3D to perform the air ingress calculation. If a CFD analysis of air diffusion into the lower plenum is required, then the CFD software will be used coupled to RELAP5-3D since RELAP5-3D will have the capability to calculate exothermic graphite oxidation.

RELAP5-3D currently has a two-species diffusion model (helium and nitrogen). The model will be extended to a multi-species model. In addition, a model will be implemented for the graphite oxidation, which will represent both the accompanying heat generation and mass consumption of oxygen and generation of CO and CO_2. A report documenting the verification of these models will be issued (FY-06). Validation studies, for both the CFD software and RELAP5-3D, that are planned include:

- *Takeda & Hishida Experiments:* The experiments performed by Takeda and Hishida centered on a reverse U-shaped tube configuration and a simple model of the HTTR. The studies will focus on the flow behavior of multi-component gas mixtures due to molecular diffusion and the natural circulation of the multi-component gas mixture (FY-07).
- *NACOK Experiment:* The NACOK experiments [Schaaf et al. 1997] were designed to model a representative section of a VHTR core undergoing the effects of air ingression following a LOCA. Therefore, the data from these experiments are suitable for validating diffusion modeling capability. A RELAP5-3D and Fluent model will be constructed and validation calculations will be performed. A report documenting the validation will be issued effort and will provide conclusions on the capability of the codes to properly predict system behavior (FY-08).
- *Helium/Air Heat Transfer Experiments:* Data from experiments planned for FY-06 to -07 will provide the basis for validation of heat transfer and pressure drop in the core for mixtures of air and helium (FY-08).

The DCC event also creates the potential for the release of fission products into the confinement or containment. The release of fission products from the fuel (including radioactive dust in the case of the pebble bed reactor), transport within the coolant system and confinement, and the deposition of these products must be calculable. To provide these capabilities in an overall systems code approach, the inherent models in PARFUME (fission product release) will be augmented by the capabilities of a code such as VICTORIA (fission product transport and deposition). These codes will be linked to RELAP5-3D in FY-10 using the existing PVMEXEC protocol. A report will be issued demonstrating the verification of the coupled models (FY-10).

4.3.13 Integral System Behavior Validations, Development, and Analyses

The ultimate objective of the NGNP Program software validation and qualification effort is to demonstrate the capability of the required software to produce calculations that describe the NGNP integral system behavior with acceptable accuracy for operational conditions and off-normal or accident conditions. The focus of these essential calculations is usually the location and magnitude of the peak fuel temperatures and peak structural temperatures, although other variables will be identified that are important, e.g., peak structural loads, peak power under adverse conditions and operational conditions, conditions that lead to adverse operating conditions for the intermediate heat exchanger or the turbine, potentially damaging oscillatory conditions, etc. Because the NGNP reactor has many components, the net system behavior is described properly when all of the system component interactions are accurately calculated. The NGNP system model will require both development and validation. Development tasks consist of:

- *Concurrent Fluent Models:* A coupled RELAP5-3D/Fluent model will be used to model the reactor vessel for the PCC events, in which Fluent will model the inlet and outlet plenums and RELAP5-3D will model the core region. Fluent is needed for the upper and lower plenum modeling to capture the three-dimensional flow patterns that will occur in these regions as coolant circulates within the vessel. To accomplish this, the present capability to link RELAP5-3D models to Fluent models using the coupling protocol will be extended to enable two separate Fluent models communicating with the RELAP5-3D model. A report will be issued documenting the dual coupling capability in FY-07.
- *Coupled Neutronics:* It appears likely that it may be necessary to model part of the core region (prismatic or pebble bed) using Fluent, and the remainder with RELAP5-3D so as to enable a direct comparison of coolant channel flow behavior between the codes under the same conditions, i.e., a code-to-code validation. This is presently possible except for the exchange of neutronic data

between the codes—although the mechanism to exchange neutronic data has been developed. This task will extend the coupling capability to allow neutronic data to be exchanged between the codes during each time step. A report will be issued demonstrating this capability in FY-09.

- *RELAP5-3D/PEBBED Linkage:* PEBBED will be the neutronic module for the pebble bed reactor. This task will link PEBBED with RELAP5-3D providing a complete thermal hydraulic and neutronic systems analysis package, assuming a pebble bed design has been chosen by then. The linkage strategy will be the same as that currently employed with the NESTLE code imbedded in RELAP5-3D. A report will be issued documenting the verification of the linkage in FY-09.

- *Balance-of-Plant Components:* Data will be acquired that are representative for turbines, compressors, and reheaters planned for use in the NGNP. Models of these components will be developed to perform the necessary validation. RELAP5-3D already contains the modeling elements required for these components. The design data are needed to build the system-wide NGNP model and should be available by FY-09. A report documenting the basis for these models will be issued in FY-09.

- *Intermediate Heat Exchanger:* The NGNP system model will require a model for the intermediate heat exchanger that couples the coolant system to the hydrogen production system. By FY-09, sufficient information should be available to enable incorporating a mathematical model of the intermediate heat exchanger into RELAP5-3D. A sufficient representation of the hydrogen production system interface will be implemented through the use of RELAP5-3D control blocks. A report will be issued demonstrating the functionality of the intermediate heat exchanger model as compared to design specifications in FY-09.

To ensure the coupled CFD/systems code software can calculate integral system behavior properly, a series of validation calculations will be performed using data from the two operational integral facilities: HTTR in Japan and HTR-10 in China. Finally, validation of the system model will utilize data from the scaled vessel experiments.

HTTR and HTR-10 reactors. The High Temperature Reactor-10 is a Chinese 10 MW pebble bed gas cooled reactor that became operational in 2000 and presents an ideal source of data for validation of pebble bed system modeling. A spectrum of experiments is planned, including perhaps a LOCA and pressurized conduction cooldown. Portions of these data will be available through the International Atomic Energy Agency and the Institute of Nuclear Energy Technology in Beijing. This task will develop a system model of HTR-10 and perform validation calculations.

The High Temperature Engineering Test Reactor (HTTR) is a 30 MW prismatic gas cooled reactor in Japan. The potential for HTTR experiments to provide high quality data for code validation is great, but may be limited by available instrumentation and restrictions on the range of transients that may be permitted.

System behavior calculations. NGNP system behavior during normal operation as well accidents will be will be performed using RELAP5-3D and Fluent models of the plant at each stage of the design process. The first complete model of the plant will be built after completion of the pre-conceptual design. Calculations will focus on peak temperatures during the most challenging scenarios. Reports will be issued at each design stage.

4.4 Liquid Salt-Cooled NGNP Methods Development and Design Assessments

The reference NGNP concept utilizes helium gas as the primary coolant based on wide-spread experience with helium for high-temperature reactors systems as discussed in Section 1.2 and helium's demonstrated compatibility with high-temperature graphite fuels. Although gas coolants suffer from low thermal capacity and thermal conductivity, most liquid coolants such as water and liquid metals are limited in their application to very high temperature systems because of low boiling temperatures and the associated difficulties of controlling two-phase systems. Liquid salts are an alternative to helium gas for NGNP because they have very high boiling temperatures (up to 1400 °C) and have thermal capacities that are orders of magnitude greater than helium gas. Table 4.5 shows several relevant thermal-physical parameters for various reactor coolants including two candidate liquid salts. Although lead also has a high boiling temperature, it is incompatible with high-temperature graphite fuels and dissolves most metals at very high temperatures; hence gas and liquid salts are the only viable coolants for very high-temperature reactors.

Table 4.5. Thermo-physical properties* of common reactor coolants.

Coolant	T_{melt} (°C)	T_{boil} (°C)	ρ (kg/m^3)	C_p (kJ/kg°C)	ρC_p (kJ/m^3°C)	K (W/m°C)	$v \cdot 10^6$ (m^2/s)
Li_2BeF_4 (FLiBe)	459	1,430	1,940	2.34	4,540	1.0	2.9
0.58NaF-0.42ZrF_4	500	1,290	3,140	1.17	3,670	~1	0.53
Sodium	97.8	883	790	1.27	1,000	62	0.25
Lead	328	1,750	10,540	0.16	1,700	16	0.13
Helium (7.5 MPa)			3.8	5.2	20	0.29	11.0
Water (7.5 MPa)	0	100	732	5.5	4,040	0.56	0.13

*ρ is density; C_p is specific heat; K is thermal conductivity; v is viscosity.

The excellent heat transfer properties of liquid salts, compared with those of helium gas, result in lower fuel temperatures by: (1) reducing the temperature differential between the fuel channel and the coolant channel within the reactor core, and (2) reducing the temperature differential between the reactor outlet and the power conversion system or hydrogen production facility. These combined effects from the use of liquid salts as the primary coolant may result in more than a 200 °C drop in fuel temperature relative to helium coolant. The better heat transfer capabilities of liquid salts compared with those of helium provide for several additional benefits:

- *Design margins*. The thermal design margins can be increased compared with those for gas-cooled reactors.
- *Higher core power densities*. The power densities can be increased to reduce the reactor core size or increase the total power output. Gas-cooled reactors traditionally have very low power densities because of poor heat transfer. With a liquid-salt coolant, the power density can be increased significantly.
- *Improved decay heat removal*. Improved heat transfer by natural circulation of the liquid salt allows the design of larger reactors with passive safety.

There are several molten-fluoride salts that have been used in test reactors or other applications that are applicable to the AHTR. The 2.5 MW(t) Aircraft Reactor Experiment (ARE) operated in the 1950s with an NaF-ZrF_4 molten salt, while the 8 MW(t) MSRE [Weinberg et al. 1970] operated in the 1960s with Li_2BeF_4 (FLiBe) molten salt in both the primary and secondary loops. Although it was never used in a reactor, extensive investigations were performed for a ternary alkali-fluoride (LiF-NaF-KF—"FLiNaK") for high-temperature nuclear service. In both the ARE and the MSRE reactors, the fuel was dissolved in the salt, whereas for the NGNP, a fuel-free salt is being considered as the primary coolant. The term *molten salt reactors* discussed in the literature typically refers to reactors in which the fuel and fission products are dissolved in the coolant and, unfortunately, leads to frequent confusion regarding the nature of the liquid-salt-cooled reactors. Despite the confusion, there exists a large technological base of experience gained from both of the earlier molten salt reactor programs. These programs operated major test facilities for studying corrosion, pumps, valves, heat exchangers, and other components in liquid salt environments up to ~850 °C. This experience is captured in a repository of more than 1000 technical reports.

Like liquid-metal-cooled reactors, the use of liquid salts allows the system to be operated at nearly ambient pressure, thus greatly reducing pipe and vessel thickness. This is especially important at the elevated temperatures required for the NGNP. Therefore, the consequence of using a liquid salt versus helium coolant is that the balance of plant may be more similar to a liquid-sodium-cooled reactor, for which considerable technology development was performed during the period 1960-1980. Regarding the primary reactor system, technology development needs are dominated by high-temperature fuels and materials, regardless of the choice of coolant. Hence, although liquid salt is considered a non-traditional reactor coolant, it offers NGNP several potential advantages with relatively few additional technology development requirements.

4.4.1 Development of Thermal-Hydraulics Methods for Liquid-Salt-Cooled NGNP Design

In order to design a primary reactor system that uses liquid-salt coolant, existing thermal-hydraulic codes will need to be modified to include the thermo-physical properties of liquid salts. Two molten salt coolants will be implemented initially into the RELAP5-3D code to support required NGNP analyses. The salts currently being considered are Li_2BeF_4 (FLiBe) and LiFNaFKF (FLiNaK). FLiBe is currently the leading candidate for the primary coolant in the liquid-salt cooled version of the NGNP, while FLiNaK is a leading candidate for the intermediate heat transport loop. The implementation of FLiBe will be based on an existing equation of state developed from a soft-sphere model. Equations of state are not available for FLiNaK and a simplified implementation will be based on the available property data from ORNL. After the revised RELAP5-3D is working, analyses will be completed to support the pre-conceptual design of the liquid- salt-cooled NGNP. This work will be done in collaboration with related work at ORNL.

Three deliverables will be completed in FY-05: (1) the latest version of RELAP5-3D with the inclusion of properties of FLiBe and FLiNaK salts will be completed, (2) a report describing this development work will be prepared, and (3) a contribution to a year end report with ORNL presenting the results of the liquid-salt-cooled NGNP safety analyses.

If the liquid-salt-cooled variant of NGNP is carried forward, additional development and validation of thermal-hydraulic data and methods will be needed. Depending on the outcome of the salt selection task, property data for additional salt compositions may need to be added to RELAP. Because earlier molten salt reactor programs considered operating temperatures below about 750 °C, it will be necessary to conduct experimental measurements of salt thermo-physical parameters above 750 °C. This is especially true for thermal conductivity and heat capacity, which are difficult to measure at very high temperatures but are parameters that are critically important for safety analyses. Also, the use of liquid salts as the primary coolant will enable the system to operate at temperatures where radiative (infrared) heat transfer is important. This will require an evaluation, and possible extension, of existing thermal-hydraulic codes to accurately model heat transfer within a coolant channel, across the gap between the pressure vessel and guard vessel, and within the reactor cavity. Also, scaled or full-sized test loops for integral validation of thermal-hydraulic behavior of the salt at temperatures >750 °C will be required. This will include both normal forced circulation modes and natural circulation operation resulting from a loss of forced circulation.

4.4.2 Liquid-Salt-Cooled NGNP Neutronics and Thermal-Hydrualics Assessments

The significantly better thermo-physical properties of liquid fluoride salts relative to helium permit a wider range of core design options. Preliminary analyses [Ingersoll, 2004] suggest that using a core design very similar to the helium-cooled NGNP can yield a total power output of 2400 MW(t). However, a core design optimized for liquid-salt coolant will likely be quite different, e.g. will likely be smaller with a higher power density and may not be annular. Also, the nuclear absorption characteristics of the coolant salt constituents can result in an increase in the core reactivity in the unlikely event of core voiding; hence it is highly desirable to develop a core design that precludes a positive void coefficient. During FY-05, ORNL will lead an effort of ORNL, INL, and ANL to perform neutronics and thermal hydraulics analyses to determine pre-conceptual core design parameters such as fuel pin and coolant channel diameters, pitch-to-diameter ratio, fuel packing fraction, etc. Reactivity coefficients (temperature, coolant voiding, etc.) will be characterized and the preferred core volume, shape, power density, etc. will be assessed. Specifically, INL will develop a Monte Carlo core model for the study of key reactor physics parameters and perform safety analyses to assess passive decay heat removal characteristics. ANL will also contribute to the neutronic analyses efforts using diffusion theory codes.

The FY-05 deliverable for this task will be a comprehensive report describing the neutronics, thermal-hydraulics, and safety analyses performed for the liquid-salt-cooled NGNP. The report will provide a new baseline pre-conceptual core design for the liquid-salt-cooled NGNP variant.

The FY-06 development program will involve the completion of the pre-conceptual design of the liquid-salt-cooled NGNP, including the design of a control rod system and a balance of plant. This will be followed in subsequent years by conceptual and final plant design. As stated previously, the liquid-salted-cooled variant shares many technology development needs with the helium-cooled variant. This is true for neutronics methods, which are largely driven by the ability to accurately model the double heterogeneity effect of the coated particle fuel. However, some different or additional development needs to the support the conceptual and final design effort are: (a) demonstration and validation of passive decay heat removal at elevated temperatures and for higher power output, (b) experimental validation of reactivity coefficients and feedback effects, and (c) possible measurement, evaluation, and processing of improved nuclear data for salt constituents. Regarding the latter requirement, a sensitivity and uncertainty analysis will need to be performed in FY-06 to assess the impact of the nuclear cross sections for the constituents of the selected salt on the neutronics performance of the reactor core, including transients.

4.4.3 Salt Selection

There exists a wide range of liquid salts that can be considered as a primary reactor coolant. Fluoride salts are generally agreed to be the most promising and can be combined with many single or multiple constituents to form complex salts with widely varying properties. The choice of liquid salt is critically important because it drives many of the other design choices, including core design, structural materials, salt cleanup systems, and operations and maintenance considerations. ORNL will evaluate the implications of candidate salts such as impact on core neutronics, activation and transmutation, toxicity, material compatibilities, freeze and boil temperatures, viscosity, etc. Also, ORNL will survey previous data and experience on use of molten salts including those used in the ARE, MPRE and MSBR programs to assess the current knowledge base of thermo-physical and chemical properties and performance, and assess the status of molten salt phase diagram modeling for candidate salts.

The deliverables for FY-05 will be: (1) a letter report on the review, assessment, and recommendations regarding the thermo-physical properties of candidate salts, and (2) a letter report on the review, assessment, and recommendations of material compatibilities for candidate salts.

Although a considerable base of liquid salt properties and performance data exist, longer-term salt research and development activities will be needed to advance the liquid-salt-cooled variant of the NGNP. Material test loops will be needed to study and validate the compatibility of liquid salt with metal alloys and carbon-composite materials to be used in NGNP. Because of the long test times needed for corrosion tests, these need to be started as early as possible. Also, basic thermo-physical properties need to be measured, especially above 750 °C. The extent of the required measurements will depend on the outcome of the FY-05 review. Chemical considerations of liquid salts will also need to be studied to support the selection of plant materials and the development of salt cleanup systems. Finally, limited work has been done to date on phase diagram modeling for liquid salts. This work is needed to support final salt selection and to more accurately predict salt performance, material compatibilities, fission product retention, and cleanup.

4.4.4 Liquid-Salt-Cooled Test Reactor

A conceptual design of a relatively small scale (50 to 100 MWth) liquid-salt-cooled VHTR will be developed. This facility will be used to test the passive safety of a liquid-salt-cooled VHTR, to test liquid salt handling equipment, materials corrosion issues, and related topics. It will be located in the LOFT containment building at the INL.

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Research and Development Program Plan

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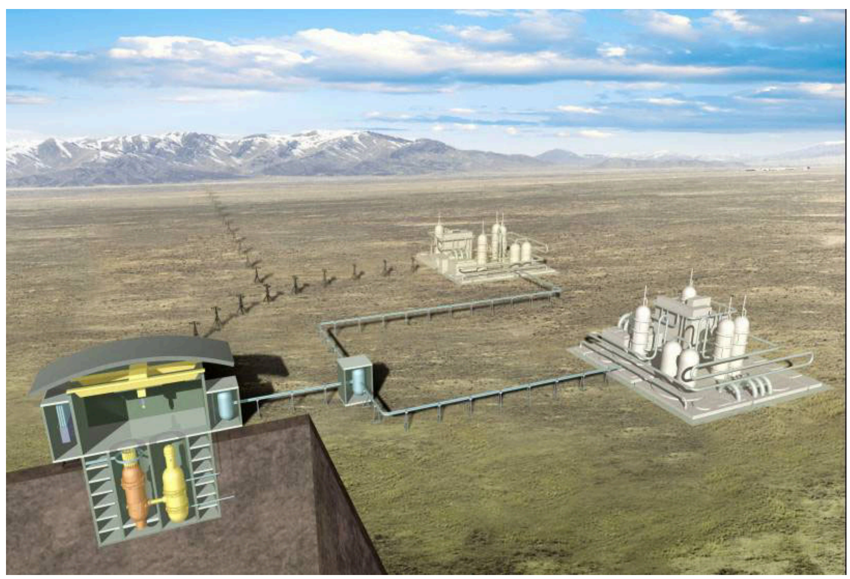

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