Appendix 1.0
Next Generation Nuclear Plant

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Acronyms

2-MGEM	2-modulator generalized ellipsometry microscope
AGR	Advanced Gas Reactor
ANL	Argonne National Laboratory
ANP	Aircraft Nuclear Propulsion
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATHENA	Advanced Thermal-Hydraulic Energy Network Analyzer
ATR	Advanced Test Reactor
AVR	Arbetisgemeinschaft Versuchsreaktor
BOP	balance-of-plant
B&PV	Boiler and Pressure Vessel
C_f/C	carbon/carbon
CD	critical decision
CEA	Centre d'Etude Atomique
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CO	oxycarbide
CV	cross vessel
D&EM	Design and Evaluation Methods
DCC	depressurized conduction cool down
DNS	direct numerical simulation
DOE	Department of Energy
EPACT	Energy Policy Act
FY	fiscal year
GIF	Generation IV International Forum
GRSAC	Graphite Reactor Severe Accident Code
GT-MHR	Gas Turbine-Modular Helium Reactor
HFIR	High-Flux Isotope Reactor
HL F&R	high-level functions and requirements
HTDM	High-Temperature Design Methodology
HTGR	High-Temperature Gas Reactor
HTR-10	10 MW High Temperature Gas-Cooled Reactor
HTTR	High Temperature Engineering Test Reactor
IHX	intermediate heat exchanger
I-NERI	International Nuclear Energy Research Initiatives
INL	Idaho National Laboratory
IPyC	inner pyrolytic carbon
IRPhEP	International Reactor Physics Evaluation Project
ITRG	Independent Technology Review Group
LES	large eddy simulation
LP	lower plenum
LWR	light water reactors
MCNP	Monte Carlo N-Particle Transport Code
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
MIR	Matched-Index-of-Refraction
MSBR	Molten Salt Breeder Reactor
NERI	Nuclear Energy Research Initiatives
NGNP	Next Generation Nuclear Plant

NHI	National Hydrogen Initiative
NRC	Nuclear Regulatory Commission
ORELA	Oak Ridge Electron Linear Accelerator
ORIGEN	Oak Ridge isotope generation
ORNL	Oak Ridge National Laboratory
PARCS	Purdue Advanced Reactor Core Simulator
PBMR	Pebble Bed Modular Reactor
PCC	pressurized conduction cooldown
PIE	postirradiation examination
PIRT	phenomena identification and ranking table
PMBs	project management boards
R&D	research and development
R/B	release to birth ratio
RCCS	reactor cavity cooling system
redox	oxidation-reduction
RELAP	Reactor Excursion and Leak Analysis Program
RPV	reactor pressure vessel
SCALE	Standardized Computer Analyses for Licensing Evaluation
Si_fC/SiC	silicon carbide-silicon carbide
THTR	Thorium High-Temperature Reactor
TRISO	tri-isotopic
TSUNAMI	Tools for Sensitivity and Uncertainty Analysis Methodology Implementation
UCO	uranium oxycarbide
UO_2	uranium dioxide
UP	upper plenum
V&V	verify and validate (verification and validation)
VHTR	Very-High-Temperature Reactor
VHTRC	Very-High-Temperature Reactor Critical

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A1.1 INTRODUCTION AND BACKGROUND

The Department of Energy (DOE) has selected Idaho National Laboratory (INL) as the lead national laboratory for nuclear energy research. Per the terms of the Energy Policy Act (EPACT) of 2005, Title VI, Subtitle C, Section 662, INL, under the direction of the DOE, will lead the development of the Next Generation Nuclear Plant (NGNP) by integrating, conducting, and coordinating all necessary research and development (R&D) activities and by organizing project participants. INL will be responsible for conducting site- and project-related procurements and coordinating project efforts with industry and the international community.

As presented in the National Energy Policy, there is a national strategic need to promote further reliance on safe, clean, economical nuclear energy. In the 2003 State of the Union Address, President Bush launched a new National Hydrogen Initiative (NHI) to provide domestically produced clean-burning hydrogen to the transportation sector as an alternative to imported oil. The combination of these two objectives, to promote nuclear energy and to produce clean-burning hydrogen, can be met simultaneously with the development of new advanced reactor and hydrogen generation technology. The DOE's mission need is to develop this combined technology that will enable the continued use of secure, domestic nuclear energy, and establish a greenhouse-gas-free technology for the production of hydrogen, thereby supporting both the President's agenda for a hydrogen economy and DOE's strategic goal to promote a diverse supply of energy.

In July of 2005, Congress passed the EPACT of 2005, which was signed into law by the President in August of 2005. Under Section 641, the Act states, "The Secretary shall establish a project to be known as the 'Next Generation Nuclear Plant Project.'" It continues, "The Project shall consist of the research, development, design, construction, and operation of a prototype plant, including a nuclear reactor that:

1. "Is based on research and development activities supported by the Generation IV Nuclear Energy Systems Initiative….
2. "Shall be used to
   - generate electricity
   - produce hydrogen
   - both generate electricity and to produce hydrogen."

This appendix of the *Generation IV Nuclear Energy Systems Ten-Year Program Plan Fiscal Year 2006* provides a preliminary project management strategy for the NGNP Project consistent with the authorization in the EPACT 2005. In preparing this appendix, certain assumptions were made concerning acquisition, alternatives, etc., which are noted further in this document. The acquisition strategy is developed during the definition phase; therefore, these assumptions will either be validated or changed as the proposed project progresses through the phases of planning, project definition, and preliminary design.

A1.1.1 System Description

The reference NGNP prototype concept is based on what is judged to be the lowest risk technology development that will achieve the needed commercial functional requirements to provide an economically competitive nuclear heat source and hydrogen production capability. This is the primary mission of the NGNP. The reference concept includes a helium-cooled, graphite moderated, thermal neutron spectrum reactor. The reactor outlet temperature will be in the range of 850 to 950°C, with future capabilities that could reach above 1,000°C. The reactor core technology will either be a prismatic block or pebble bed concept; the decision on the reference will be made during the definition phase. The NGNP will produce both electricity and hydrogen using an indirect cycle with an intermediate heat exchanger (IHX) to transfer the heat to either a hydrogen-production demonstration facility or a gas turbine. The IHX and primary gas circulator are located in an adjoining power conversion vessel. Figure A1-1 is a conceptual schematic of the NGNP.

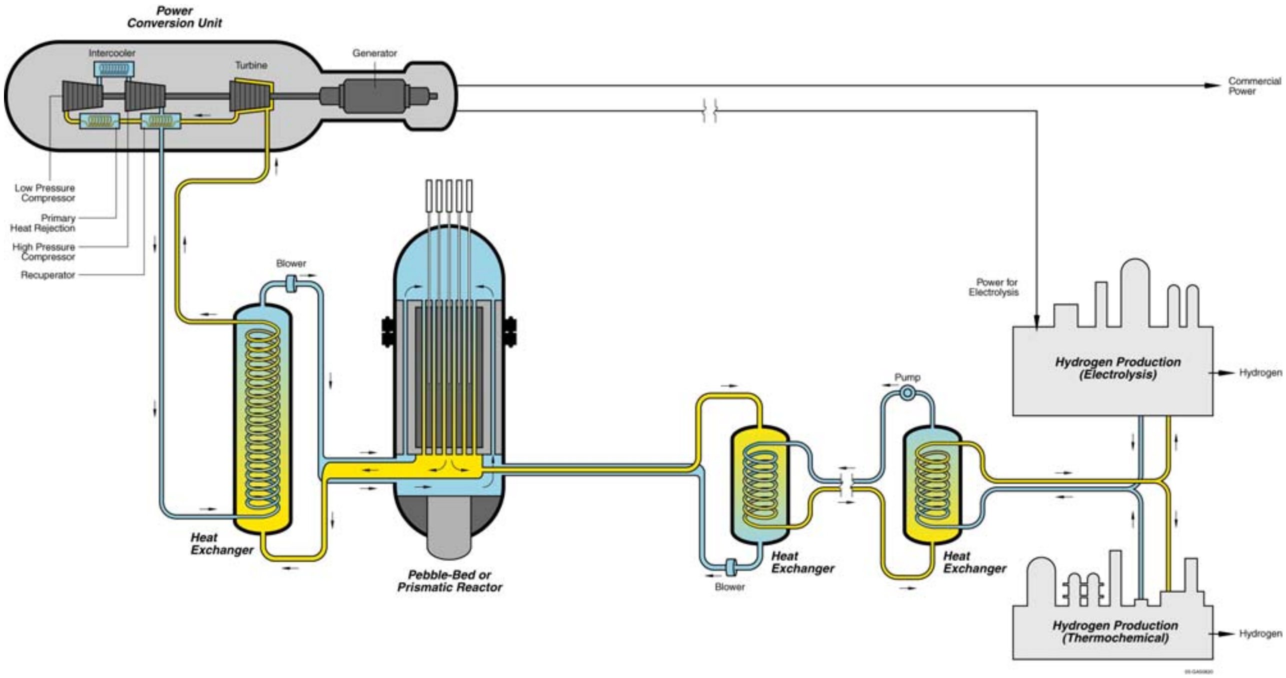


Figure A1.1. NGNP conceptual schematic showing power generation and hydrogen production.

The reactor thermal power (~600 MW_t) and reactor configuration will be designed to ensure passive decay heat removal without fuel damage during licensing basis accidents. The initial fuel cycle will be a once-through, high-burnup, low-enriched uranium fuel cycle. Other fuel cycle possibilities will be considered after the prototype has become operational.

The basic technology for the NGNP has been established in former high-temperature gas-cooled reactor plants (Dragon, England; Peach Bottom Unit 1, U.S.; Arbetisgemeinschaft Versuchsreaktor (AVR), Germany; Thorium Hochtemperatur Reaktor, Germany; and Fort St. Vrain, U.S.). In addition, the technologies for the NGNP are being advanced in the Gas Turbine-Modular Helium Reactor (GT-MHR) project for a prismatic reactor, and in the Pebble Bed Modular Reactor (PBMR) project for a pebble bed reactor. Furthermore, the Japanese High Temperature Engineering Test Reactor (HTTR) project and Chinese High Temperature Reactor (HTR)-10 project are scaled reactors demonstrating the feasibility of some of the planned NGNP components and materials.

The nuclear fuel is tri-isotopic (TRISO)-coated fuel particles embedded in graphite, either as compacts to be placed in prismatic blocks or as pebbles. The center of the core is a nonfueled graphite reflector. Normal operating maximum fuel temperatures do not exceed 1,250°C.

Passive safety is achieved by designing for a core cooldown during a postulated long-term depressurized loss-of-forced-convection accident that limits the peak fuel temperatures to 1,600°C. This is accomplished by conducting the decay heat radially through the core and pressure vessel, by radiation to the reactor building structure, and, finally, by conduction to the ground. A cross vessel (CV) connects the reactor vessel to the IHX, or a power conversion vessel, that is deliberately made as short as possible to minimize thermal expansion differences between the two large vessels. Within the CV, the reactor inlet gas flows in an annular duct along the inside surface of the CV to the reactor inlet. The core exit hot gas flows in a central duct along the centerline of the CV to the IHX. Other design configurations will be considered during the conceptual design process.

One or more processes will use the heat from the high-temperature helium coolant to produce hydrogen. One possible option is the thermo-chemical splitting of water into hydrogen and oxygen. The primary candidate thermo-chemical processes are the sulfur-based processes. A second option 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. The waste heat from the pre-cooler and inter-cooler of the Brayton cycle can be used to further improve the efficiency of hydrogen production. Additional options, including a hybrid of the first two options, are being evaluated.

The IHX also transfers heat to the Brayton cycle power conversion systems and equipment. This equipment will be operated at conditions representative of conventional commercial industrial gas turbines for electrical power generation, using conventional electrical generator technology.

The result of this project is the demonstration of a Nuclear Regulatory Commission (NRC)-licensed, full-scale prototype (~600 MW_t), helium-cooled reactor for electricity production and/or hydrogen production.

A1.1.2 Overall System Timeline

The NGNP Project prototype will require over 12 years from initiation of conceptual design through completion of acceptance testing, assuming there is no overall funding constraint and irrespective of technology development. As shown in Figure A1-2 below, this span of time is primarily determined by the controlling path through major portions of the conceptual and preliminary designs, preparation of the licensing application for a construction permit (including preparation of the preliminary safety analysis report and the environmental report), final design, construction, and acceptance testing.

Currently, to support the anticipated project schedule, Critical Decision (CD)-1 must occur in 2009 so the prototype can be operational no later than 2021. Within this timeframe, many of the tasks identified in Phase 1 of the EPACT of 2005 will have been completed.

A1.2 PROJECT STRATEGY

A1.2.1 Objectives

High-level NGNP project objectives support both the NGNP mission and the DOE vision, as follows:

- Developing and implementing the technologies important to achieving the functional performance and design requirements determined through close collaboration with commercial industry end-users.
- Demonstrating the basis for commercialization of the nuclear system, the hydrogen production facility, and the power conversion concept. An essential part of the prototype operations will be demonstrating that the requisite reliability and capacity factor can be achieved over an extended period of operation.

- Establishing the basis to license the commercial version of NGNP by the NRC. This will be achieved in major part through licensing the prototype by NRC and initiating the process for certification of the nuclear system design.
- Fostering rebuilding of the U.S. nuclear industrial infrastructure and contributing to making the U.S. industry self-sufficient for our nuclear energy production needs.

A1.2.2 Scope

The scope of the NGNP project is to:

- Execute and complete all project deliverables, including conceptual design, preliminary and final design, construction, and startup and acceptance testing for the NGNP facility
- Complete and integrate specifically assigned technology development and system confirmatory and verification tasks
- Obtain NRC licensing as required for a commercial demonstration reactor prototype
- Provide project management and integration that will coordinate and combine the efforts of the many and varied project partners, subcontractors, and stakeholders.

The scope is further defined by the requirements of the EPACT of 2005.

A1.2.3 High-Level Requirements

The preliminary planning for the NGNP Project is based on managing implementation of three sets of requirements. The first and highest requirements are from the EPACT of 2005, which was signed into law by the President in August of 2005. The EPACT of 2005 also directs that the project be reviewed in light of the Independent Technology Review Group (ITRG) review of, *The Next Generation Nuclear Plant – High-Level Functions and Requirements,* INEEL/EXT-03-01163, and that the ITRG recommendations are addressed in the NGNP planning. The high-level functions and requirements (HL F&Rs), as modified based on the recommendations of the ITRG, will therefore be the second set of requirements. The third are the INL contractual requirements, which specify that capital projects be conducted under DOE Order 413.3, *Program and Project Management for the Acquisition of Capital Assets,* and DOE Manual 413.3-1, *Project Management for the Acquisition of Capital Assets*. The DOE Order and Manual will be followed to the extent possible, as they provide an excellent systems approach to managing projects.

The HL F&R document established highest-level functions and performance requirements to be performed and met by the NGNP. The HL F&Rs provide a foundation to define technical and functional requirements (T&FRs) and additional performance requirements at each successive lower level of the design. In addition, the HL F&Rs provide a framework for agreement by partnering organizations in achieving a roadmap to meet a goal for operation of the NGNP. The high-level goals and objectives are represented in Figure A1-2.

Since the NGNP is to provide a prototype for the commercial power industry, commercially applicable federal regulations such as 10 Code of Federal Regulations (CFR) and 40 CFR, industry consensus standards such as American Society of Mechanical Engineers (ASME) standards, and Nuclear Regulatory Commission requirements will be used to develop the requirements and procedures for the project. Federal environmental and safety regulations will also be included.

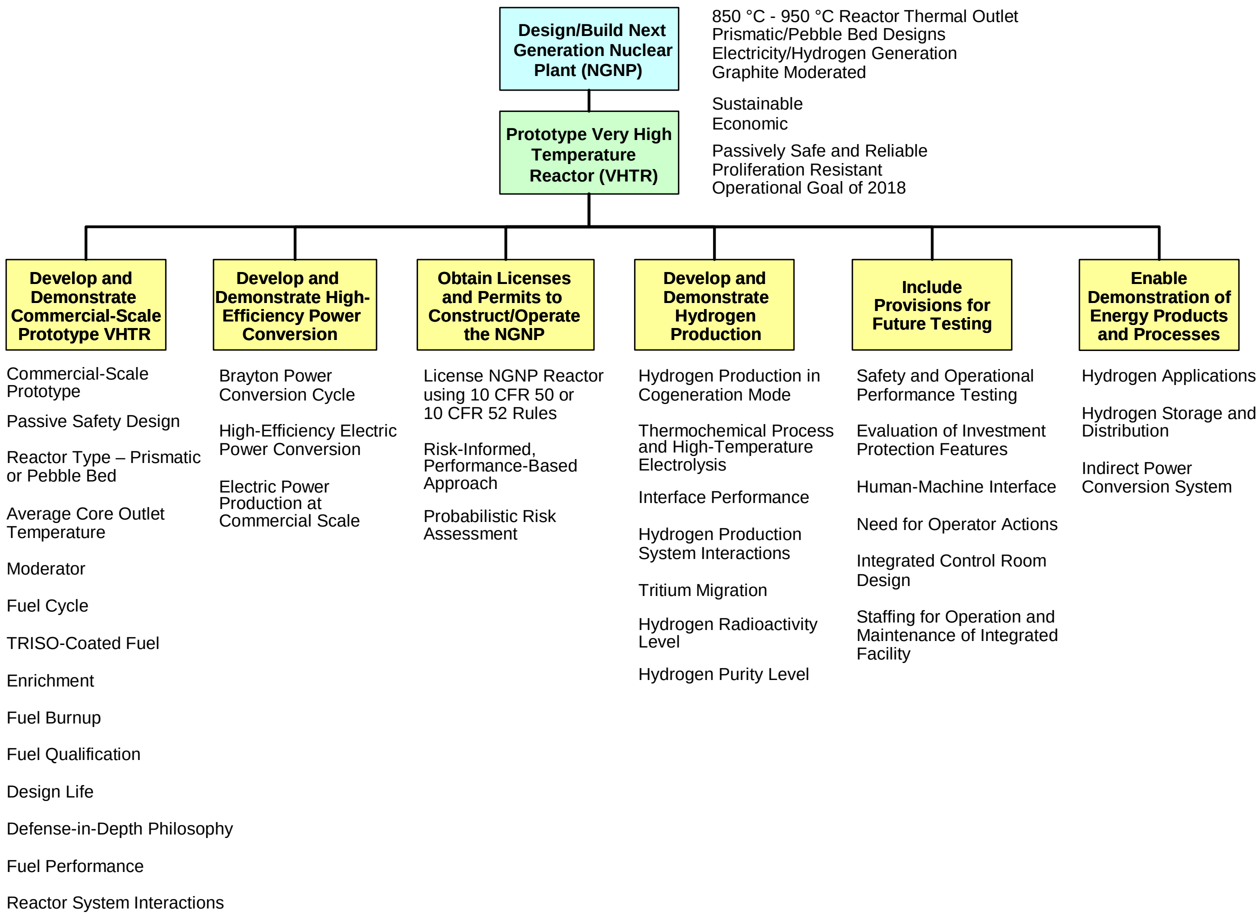


Figure A1.2. NGNP high-level functions and requirements.

A1.2.4 High-Level Project Logic and Integration

Preliminary planning for the NGNP Project has been built on a platform incorporating and following the Phase I requirements of EPACT 2005, DOE Order 413.3, and DOE Manual 413.3-1, as outlined above. A NGNP high-level logic diagram (shown in Figure A1-3) has been prepared to graphically display the initial planning-level project logic. The project logic has been broken into phases I and II. The specific approach to Phase II depends on the final, approved acquisition strategy. Selected acquisition strategy assumptions have been made to support determination of schedule durations and licensing strategy discussions, the most important of which include the following:

- INL will perform overall project management
- INL will act as the agent of DOE in day-to-day activities associated with management of interfaces with collaborative partners
- The project management requirements of DOE Order 413.3 are satisfied; however, no substantive schedule interruption is included for completion of critical decisions
- Funding is available as required to accomplish project activities, including technology development, design, construction, licensing, and acceptance testing of the prototype
- Engineering, procurement, and construction activities will be performed by others subcontracted through INL

- An organizational component of INL will be the license applicant and license holder for the NGNP prototype, and will operate the prototype either directly or through a subcontracted arrangement.

Preliminary planning indicates that the NGNP will require over 12 years, assuming there are no funding constraints. This time span was determined by using the controlling path through major portions of the conceptual and preliminary designs; preparation of the licensing application for a construction permit, including the Preliminary Safety Analysis Report (PSAR) and the Environmental Report; and final design, construction, and acceptance testing.

This planning also assumes a period of demonstration (about two years) following completion of acceptance testing to demonstrate achievable reliability and capacity factors before considering the plant operational, based on experience in light water reactors (LWRs). This operation demonstration period will be the equivalent of a cycle of operation in a commercial facility, and will be followed by a period of inspections and evaluations to determine the condition of the nuclear system, hydrogen production facility, and power conversion (e.g., defueling of the reactor core to inspect reactor internals; inspection of an IHX module; selected disassembly of the power conversion equipment to inspect rotating elements). This shakedown period becomes particularly important for new technology implementation in the NGNP Project for the nuclear system and in the hydrogen production facility. The preoperational evaluation provides proof of principle to establish the basis for commercialization decisions by the end users.

Accordingly, to consider the NGNP Project prototype to be operational by 2021 requires that conceptual design be initiated no later than 2006. Precursors to initiating conceptual design include preparation of a high-level functional specification for the commercial application(s), preparation of the prototype functional and design specification in support of requests for proposal for trade-off studies, and preconceptual scoping. Currently, to support the anticipated project schedule, CD-1 must occur in 2008 to support the prototype being operational no later than 2021. Within this timeframe, tasks in Phase 1 of the EPACT of 2005 will have been completed. Specific milestones in the project logic require appropriate progress in selected aspects of the technology development. Current technology plans to support a project schedule have not been developed. Furthermore, mitigation approaches for technical or schedule risks have not been considered/identified to ensure success. Based on the summary assessment reported in this preliminary project management plan, changes to these technology development plans are required to support a project schedule that fulfills the authorization in EPACT 2005.

While not explicitly shown in the high-level project logic, the cost estimate includes operator training and procurement of a control room simulator. The high-level project logic considers the practicalities of technology development, licensing by the NRC, and the design, construction, testing, startup, and demonstration operations of a first-of-a-kind nuclear system, hydrogen production facility, and power conversion capability. The high-level project logic provides the important design, construction, and licensing milestones, and logic ties required for the reference NGNP Project prototype for comparison with the schedule and sequence in the technology development plans.

A1.2.4.1 Acquisition Strategy

The acquisition strategy assumes a project structure that fulfills the requirements of DOE Order 413.3 and draws on efficiencies of commercial large-project management practices. The overall strategy assumes that an organizational component of INL will be the NRC license holder for the prototype and will act as the agent of DOE in day-to-day decisions and interfaces with technology development, design, construction, testing, and operations of the prototype.

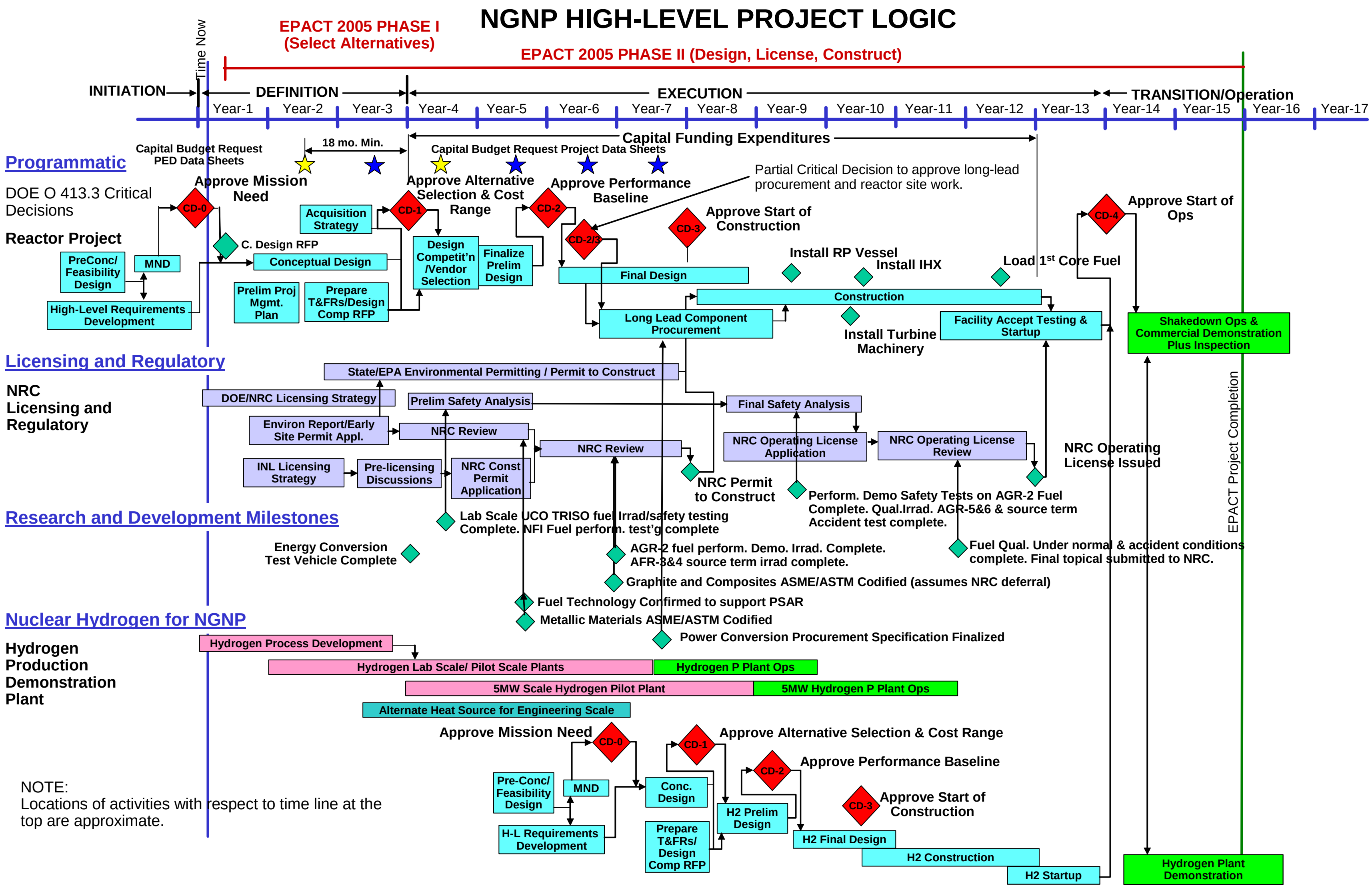


Figure A1.3. NGNP high-level project logic.

A1.2.4.2 Licensing Strategy

The overall licensing structure is anticipated to be under the provisions and requirements of 10 CFR 50. The licensing basis is expected to be risk-informed and use a mechanistic source term for accident consequence evaluation consistent with the concepts used in the proposed licensing approaches for the Modular High Temperature Gas Reactor (MHTGR) and the PBMR. Pre-application discussions with NRC will address these and other topics (e.g., approach to defining control interactions between the nuclear system and hydrogen production facility) to reach formal agreement on the overall licensing approach and NRC review criteria. These discussions are to begin in fiscal year (FY) 2006 to support the project logic. These pre-application discussions will also be required to ensure NRC and public familiarity with the NGNP prototype licensing methodologies before application for the construction permit and the subsequent operating license.

A1.3 RESEARCH AND DEVELOPMENT STRATEGY

Section 643 (a)(1-5) of the EPACT 2005 outlines five specific areas of research, called "Major Project Elements," that would support the NGNP project. These major project elements are:

1. High-temperature hydrogen production technology development and validation
2. Power conversion technology development and validation
3. Nuclear fuel development, characterization, and qualification
4. Materials selection, development, testing, and qualification
5. Reactor and balance-of-plant (BOP) design, engineering, safety analysis, and qualification.

EPACT Section 643 (b)(1)(A-D) states that Phase I of the research is to "…select and validate the appropriate technology under subsection (a)(1) (i.e., hydrogen production technology); carry out enabling research, development, and demonstration activities on technologies and components under paragraphs (2) through (4) of subsection (a) (i.e., power conversion, fuel, and materials); determine whether it is appropriate to combine electricity generation and hydrogen production in a single prototype nuclear reactor plant; and carry out initial design activities for a prototype nuclear reactor and plant, including development of design methods and safety analytical methods and studies under subsection (a)(5) (i.e., design, engineering, and safety analysis)."

The five areas described above have current research programs and R&D plans associated with them, but they have remained somewhat generic since conceptual design work that would focus the research efforts has not yet been initiated. Also, note that items 1 and 2 are covered under separate R&D plans, and will not be discussed in this appendix.

A1.3.1 Viability Issues

There are no viability issues associated with the Very-High-Temperature Reactor (VHTR). The basic technology for the NGNP was established in the former High-Temperature Gas Reactor (HTGR) test and demonstration plants (Dragon, Peach Bottom, AVR, Fort St. Vrain, and Thorium High-Temperature Reactor [THTR]). In addition, the technologies for the NGNP are being advanced in the GT-MHR Project, and the South African state utility Eskom sponsored project to develop the PBMR. Furthermore, the Japanese HTTR and Chinese 10 MW High Temperature Gas-Cooled Reactor (HTR-10) test reactors are demonstrating the feasibility of some of the planned NGNP components and materials.

A1.3.2 Research Interfaces

A key to cost effective long-term development, testing, and demonstration of the NGNP will be strategic partnerships that will provide the needed talents, infrastructures, and resources for much of the R&D. Of special importance are DOE and INL partnerships with industry, researchers at universities and national laboratories, and other Generation IV International Forum (GIF) countries.

A1.3.2.1 Relationship to Generation IV International Forum Research and Development Projects

The GIF has established a VHTR System Steering Committee for coordination of VHTR R&D. The GIF VHTR member countries are Canada, Euratom, France, Japan, the Republic of Korea, the Republic of South Africa, Switzerland, the United Kingdom and the U.S. Project Management Boards (PMBs), reporting to the Steering Committee, have been established to define R&D collaborations in specific areas. The four PMBs that are now active for the VHTR include Materials and Components, Fuel and Fuel Cycle, Hydrogen Production, and Design and Safety Methods.

Each PMB will develop multiple collaboration agreements within their area. For example, the Materials and Components Board is developing collaboration agreements for:

1. Graphite development and qualification
2. Composites (carbon/carbon [C_f/C], silicon carbide/silicon carbide [Si_fC/SiC])
3. Vessel steel qualification.

The member countries of the GIF have signed a government-to-government framework agreement, which provides the legal agreements allowing productive, yet protected, sharing of R&D results.

A1.3.2.2 University Collaborations

Partnerships with universities will be conducted through the Nuclear Energy Research Initiatives (NERI) process and formal subcontracts. The nature of these partnerships will focus on peer-reviewed, investigator-led projects in academia, as well as program R&D tasks jointly undertaken by universities and the DOE laboratories.

A1.3.2.3 Industry Interactions

The NGNP strategy is designed to engage private sector and international (e.g., agents of countries that are members of the GIF) cost-shared participation in this effort, while also integrating the research work to be conducted at the INL and other DOE laboratories.

A1.3.2.4 International Nuclear Energy Research Initiatives

A number of International-Nuclear Energy Research Initiatives (I-NERIs) are planned throughout the life of the NGNP R&D Program, some of which are currently in negotiations and others that are only envisioned. Table A1.1 provides a summary of I-NERIs currently starting, in progress, or being completed.

Table A1.1. NGNP-related I-NERIs.

I-NERI Title	Collaborators	Scope	Period
Fuels			
Development of Improved Models and Designs for Coated-Particle Gas Reactor Fuels	INL, French Centre d'Etude Atomique (CEA), Massachusetts Institute of Technology (MIT)	Develop improved fuel behavior models for gas reactor coated particle fuels. Develop improved coated-particle fuel designs that can be used reliably at very high burnups.	2003-2006
Materials			
Composites	INL, Oak Ridge National Laboratory (ORNL), French CEA	Develop, irradiate, and qualification test tubular Si_fC/SiC composite material for control rods.	2005-2010
Design Methods and Evaluation			
Development of Safety Analysis Codes and Experimental Validation for a Very-High-Temperature Gas-Cooled Reactor	INL ,University of Michigan, Korea Advanced Institute of Science & Technology (KAIST), and Seoul National University	Develop new Advanced Computational Methods for safety analysis codes and numerical and experimental validation of these computer codes. Includes improving two well-respected light water reactor transient response codes (Reactor Excursion and Leak Analysis Program (RELAP5)/Advanced Thermal-Hydraulic Energy Network Analyzer [ATHENA] and MELCOR).	2005-2006
Thermal-Hydraulic Analyses and Experiments for VHTR Safety	INL, Argonne National Laboratory (ANL), Iowa State University, Stanford University, and French Investigating Organization: CEA	Provide benchmark data for the assessment and improvement of thermal-hydraulic codes proposed for evaluating normal and reduced power operations plus decay heat removal concepts and designs in VHTRs.	2005-2007
Screening of Gas-cooled Reactor Thermal-hydraulic and Safety Analysis Tools and Experiment Database	ANL, INL, and Korean Investigating Organization: Korea Atomic Energy Research Institute (KAERI)	Develop a formal qualification framework, filter existing databases (initial), screen tools for use in thermal-hydraulics and safety analyses (preliminary).	2005-2006

A1.4 HIGHLIGHTS OF RESEARCH AND DEVELOPMENT

A1.4.1 Fuel Development

The DOE Advanced Gas Reactor (AGR) Fuel Development and Qualification Program is designed to provide a fuel qualification baseline with the following overall goals:

- Provide a baseline fuel qualification data set in support of the licensing and operation of the NGNP. Gas-reactor fuel performance demonstration and qualification comprise the longest duration R&D task for NGNP feasibility. The baseline fuel form is to be demonstrated and qualified for a peak fuel centerline temperature of 1,250°C.

- Support near-term deployment of VHTRs by reducing market entry risks posed by technical uncertainties associated with fuel production and qualification.
- Utilize international collaboration mechanisms to extend the value of DOE resources.

A1.4.1.1 Fuel Form

The fuel for the NGNP is based on the TRISO-coated particle fuel design (see Figure A1.4) demonstrated in HTGRs in Germany, the United Kingdom, the U.S., and elsewhere. The baseline fuel kernel for the NGNP is low-enriched uranium oxycarbide (UCO)—about 15% ^{235}U—for the prismatic block reactor version of the NGNP and low-enriched uranium dioxide (UO_2)—about 8% ^{235}U—for the pebble bed version of NGNP. Historically, for prismatic VHTRs, the high power densities (>6 W/cm^3) and the associated large thermal gradients 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 can result in damage to the SiC layer. Furthermore, and more importantly, at the high burnups proposed for a prismatic VHTR (15 to 20% fissions per initial metal atom), the oxycarbide (CO) and fission product gas pressure in a UO_2 fuel particle could be substantial, resulting in particle failure, especially under accident conditions. The high VHTR fuel temperatures (maximum time-averaged temperature ~1,250°C) increase the effect of both of these mechanisms. As a result, UCO has historically been the fuel kernel of choice for the prismatic VHTR to mitigate these performance risks, 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 CO fuel still ties up the lanthanide fission products as immobile oxides in the kernel, which gives the fuel added stability under accident conditions.

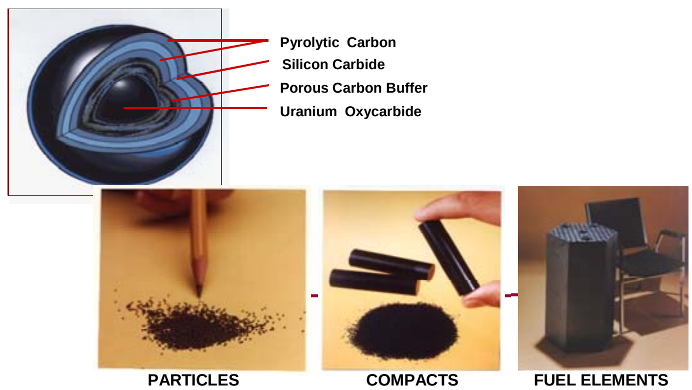


Figure A1.4. Cutaway of a TRISO-coated fuel particle and pictures of prismatic-fueled high-temperature gas reactor fuel particles, compacts, and fuel elements.

For the pebble bed version of an 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. Similarly, the prismatic version of the NGNP uses over-coated particles 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 A1.4 shows a sketch of a 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 HTGR at Fort St. Vrain.

Without a design for the NGNP, the AGR Fuel Development and Qualification Program is currently focusing on the more bounding fuel form for development and qualification (UCO TRISO). Furthermore, the AGR program determined that the lowest risk path to successful fuel qualification for UCO TRISO is to produce coatings on UCO kernels using German technology applied for AVR and THTR fuel development and qualification. The AGR program coating development activities have successfully reproduced the coatings based on the German technology at laboratory scale, and the program currently plans to irradiate a number of fuel variants (each with slightly different coatings yet still produced within the acceptable process phase space) in experiment AGR-1. This will increase confidence in establishing an acceptable fuel, provide important irradiation performance feedback to the fabrication process, and decrease the technical risk associated with coating early in the program before fabrication of qualification fuel using production scale equipment. The program then calls for a scale up of coating activities from laboratory scale to pilot scale, followed by a performance demonstration irradiation (AGR-2) and associated safety testing and postirradiation examination (PIE). The third step involves production scale, followed by formal qualification irradiations (AGR-5 and AGR-6) and associated qualification safety testing and PIE to demonstrate that high quality fuel can be manufactured from production-scale equipment and to demonstrate acceptable in-reactor and accident performance. (This three-step approach is very similar to that used by the Germans and planned by South Africa for the PBMR).

A1.4.1.2 History

The AGR program began in late 2002. Since that time, it has taken lessons-learned from past poor U.S. fuel experience (with New Production Reactor and DOE HTGR programs), recommendations from international coated particle fuel experts, and a historical review of the successful coated particle fuel development program based on German technology to establish the initial scope/direction for the AGR program. Laboratory-scale chemical vapor deposition (CVD) coating capability has been developed for TRISO-coated fuel and has demonstrated the ability to deposit coatings on UCO kernels similar to that achieved in the German-developed technology. Coating process development has established quantitative relationships among coating process parameters and key properties affecting irradiation performance of inner pyrocarbon, a primary source of past U.S. poor irradiation performance. The AGR program has also developed and qualified the requisite characterization techniques (over 35) and associated statistical sampling methodology needed to demonstrate the high quality of the TRISO-coated fuel (kernels, coating layers, and compacts). A new particle over-coating/warm pressing thermosetting resin process for making compacts, similar to that used by the Germans to make pebbles for AVR and THTR, has been developed with much higher dimensional stability and fewer defects induced in the particles during pressing of the compact than in the prior U.S. process. A detailed design of a multicapsule test train to enable more efficient, better-controlled, and better-monitored irradiation of large quantities of coated particle fuel in compacts, is complete. As of this writing, fabrication of the test train and ancillary control and monitoring equipment is underway, and the program is utilizing laboratory-scale equipment to fabricate TRISO-coated particle fuel for irradiation in the first experiment (AGR-1) which is scheduled for insertion in October 2006.

A1.4.1.3 Fuel Fabrication

The fuel-fabrication portion of the AGR program will produce coated-particle fuel that meets fuel performance specifications and includes process development for kernels, coatings, and compacting; quality control methods development; scale-up analyses; and process documentation needed for technology transfer. Fuel and material samples are produced for characterization, irradiation, and accident testing as necessary to meet the overall goals. Automated fuel fabrication technology suitable for mass production of coated-particle fuel at an acceptable cost will eventually be developed in later stages of the program and in conjunction with industrial partners.

The near-term activities focus on production of UCO kernels and coating of particles in a continuous process using a small (2-in.) laboratory-scale coater. The goal of these initial coating studies is to provide coatings produced under a range of coating conditions, like those produced by the German program in the late 1980s. However, coating variants are planned that will confirm our understanding of the historical coating fabrication database, and some of the variants will be irradiated in AGR-1. The second phase of the coating development involves scale-up of the continuous coating process to production size (e.g., 6-in.) coaters. The goal is to produce high quality coatings for performance demonstration and, ultimately, qualification.

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, is a substantial departure from the thermoplastic matrix injection process used previously in U.S. development work, and is required to adapt the process to the U.S. fuel compact specifications.

In parallel with the fuel fabrication, fuel characterization will develop more advanced and robust techniques to measure key attributes of the fuel that can be integrated into a continuous production-scale coating process. Initial activities focus on developing improved anisotropy, coating layer thickness, and particle sphericity measurement techniques. Computer-controlled sample positioning and digital imaging, plus an Oak Ridge National Laboratory (ORNL)-developed image analysis software, is used to quickly and easily analyze thousands of particles, or particle cross-sections, for size and shape with a 1- to 2-µm resolution. An example of information from that system is shown in Figure A1.5.

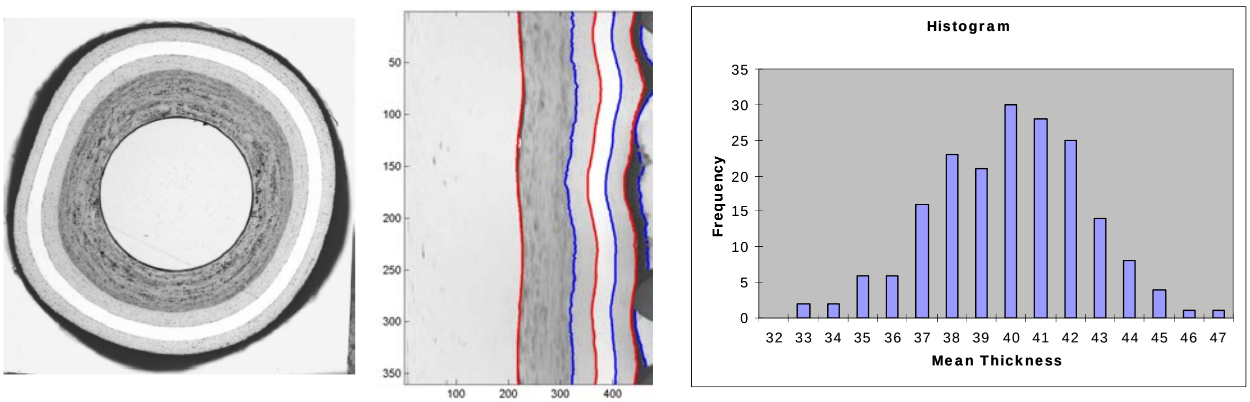


Figure A1.5. Example information from the ORNL computer-automated optical characterization system. An inner pyrolytic carbon (IPyC) layer histogram is shown on the right.

The crystallographic orientation in the pyrocarbon layers is measured by a scanning ellipsometry technique called the 2-modulator generalized ellipsometry microscope (2-MGEM). Figure A1.6 shows typical results from that equipment.

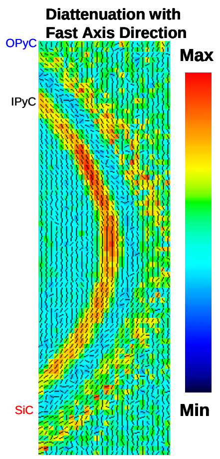


Figure A1.6. Typical results from the ORNL 2-MGEM equipment for measuring pyrocarbon anisotropy.

A1.4.1.4 Fuel Irradiation

The fuel irradiation activities will produce fuel performance data 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 for PIE and ex-core high-temperature furnace safety testing.

Eight irradiation capsules will be used to obtain the necessary data and sample materials. Details on each irradiation are listed in Table A1.2. The purpose of AGR-1 is to test a number of variants of fuel produced under different processing conditions from laboratory-scale coating equipment. 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 A1.2. 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 verification and validation
AGR-8	Fission product transport – 3

Given the very large number of fuel particles in a VHTR core, a large number of fuel specimens are needed to fully qualify 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 (V&V) fuel performance and fission product transport models.

The capsules will be irradiated in one of the large “B” irradiation positions in the Advanced Test Reactor (ATR) at INL. 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 two years to 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. Each cell containing fuel specimens will be “sniffed” for fission gas. The sniffing gas will also be used to transport 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.

Data from the PIE will supplement the in-reactor measurements (primarily fission gas release-to-birth ratio [R/B] measurements) as necessary to demonstrate compliance with the fuel performance requirements and support development and validation of the computer codes. This work will also support the fuel manufacture with feedback on the performance of kernels, coatings, and compacts. The various analyses and measurements that are proposed and their purpose are shown in Table A1.3.

Table A1.3. AGR PIE.

PIE Analysis or Test	Purpose
Gamma scan of the entire test train	Provides information to determine whether any fuel elements have broken or if a significant number of fission products have been released.
Post capsule disassembly specimen weights and dimensional measurements	Provides shrinkage or swelling characterization from irradiation.
Optical metallography on cross sections of fuel pebbles or compacts	Provides physical characteristics of irradiated fuel particle coatings.
Capsule components Gamma-scanning or leaching and gamma counting	Identifies migration and distribution of fission products following irradiation.
Electron microscopy of deconsolidated individual fuel particles	Examines x-ray characteristics of specific fission products providing evidence of fission product accumulation at the IPyC/SiC interface, fission product attack of the SiC, and fission products outside the fuel particles.
Measurement of fuel particle failure fraction independently of the on-line R/B measurements using the leach-burn-leach method	Provides a measurement of free uranium, which is converted to a SiC defect fraction.
Irradiated microsphere gamma analysis (IMGA)	Provides a histogram of the ratio of $^{137}Cs/^{152}Eu$ based on all the particles in individual spheres compared to a normal distribution.
Metallography following IMGA	Ties the microstructure of the anomalous particles to the fission product release.
Traditional burnup analysis	Determines the concentration of transuranics and minor actinides, to assess burnup.

A1.4.1.5 Safety Testing

An important goal of this program is to evaluate the integrity and performance of the coated particle fuel under high-temperature accident conditions, which is essential to 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 is 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.

The data needed from safety testing are fission product release, TRISO coating layer integrity, and fission product distribution within fuel particles (likelihood of corrosion) and fuel elements.

Post heating 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 IPyC layers. Detailed test matrices will be developed as the program evolves.

A1.4.1.6 Fuel Performance Modeling

Computer codes and models of the HTGR TRISO-coated fuel performance 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 also simulate 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. The new models will be first-principles-based, mechanistic, integrated, thermal-mechanical, physio-chemical irradiation performance models for particle fuel, which have the proper dimensionality yet capture the statistical nature and loading of the fuel.

These fuel performance 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. Models are very useful for both pre-test and post-test predictions for any experiment performed in this program. Sensitivity studies with the model can also 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, piggyback cells 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 the pyrocarbon, fission product release behavior in a purposely defective or initially failed particle). Moreover, some of the PIE techniques can provide maps of fission products through the particle, which can be compared with model predictions of fission product transport through the coatings.

A1.4.1.7 Fission Product Transport and Source Term

Transport of fission products produced within the coated particles will be modeled to obtain a technical basis for source terms for AGRs 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 phenomena to be modeled include:

- Fission product release from the kernel
- Transport through failed coatings
- Deposition fraction of:
    - the released fission products in the compact or sphere matrix
    - what gets through the compact on fuel element graphite (prismatic variant only)
    - 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.

A1.4.2 Materials

A1.4.2.1 Introduction

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, power conversion unit, IHX, and associated BOP. Materials for hydrogen production will be addressed by DOE’s NHI.

The primary program tasks include the following:

1. Graphite development
2. High-temperature metallic materials and design methods
3. American Society for Testing and Materials (ASTM) standards and ASME code support related to NGNP development
4. Environmental testing
5. Thermal aging
6. Metallic materials irradiation qualification
7. Composites development
8. Data management handbook
9. Reactor pressure vessel (RPV) fabrication and transportation
10. RPV emissivity evaluation and testing
11. IHX and recuperator prototype evaluation and testing
12. Metallic core internals evaluation and testing
13. Piping fabrication and evaluation
14. Hot duct liner evaluation and testing.

The *NGNP Materials R&D Program Plan* provides additional detail (INEEL 2004a).

A1.4.2.2 Component Candidate Materials

A variety of materials options have been identified for potential use in the NGNP reactor and BOP components. This section summarizes the options currently identified by function.

Graphite will be the major structural component and nuclear moderator in the NGNP reactor core. The graphite used previously in the HGTR programs in the U.S. (designated H-451) is no longer in production; thus, replacement graphites must be found and qualified. Fortunately, likely potential candidates currently exist, including fine-grained isotropic, molded or isostatically pressed, high-strength graphite suitable for 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 materials are expected to meet the requirements of the draft ASTM materials specification for the nuclear grade graphite.

The reactor internals may 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. The upper limit of these materials, however, is judged to be 1,000°C. Any component that could experience excursions above 1,000°C would need greater high-temperature strength and corrosion resistance capabilities. Carbon fiber reinforced carbon matrix (C_f/C) or carbon fiber reinforced silicon carbide (Si_fC/SiC) composites are the leading choices for materials available in the near future for service that might experience temperature excursions up to 1,200°C. Compatibility of the metals with the helium coolant and irradiation resistance of the potential candidate materials needs to be addressed.

An IHX 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. The pressure difference between the primary to secondary loop may be small (0.1 MPa) if helium is used for the intermediate heat transfer loop, or it may be large if a liquid such as molten salt is used. The leading IHX 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 and will be addressed. 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-term thermal exposure, since embrittlement could affect the performance of the IHX during thermal transients. Welding/brazing and fabrication issues exist. The leading potential candidates for service at temperatures of 900 to 1,000°C are Alloy 617, Alloy X, and Alloy XR. Other nickel-base alloys such as CCA617, Alloy 740, and Alloy 230 may be considered.

Several possible primary coolant pressure boundary systems are envisioned for the NGNP. These comprise (1) a large RPV containing the core and internals, (2) a second vessel containing an IHX and circulator (or a power conversion unit), and (3) a pressure vessel containing a CV joining the two vessels. Because these three vessels will be exposed to air on the outside and helium on the inside, emissivity of the material is an important factor regarding radiation of heat to the surrounding air to ensure adequate cooling. If the temperature can be maintained less than 375ºC by cooling or other means, conventional materials can be used. However, if the pressure boundary temperature is in the range of 375 to 500°C, advanced materials will be required.

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 for the needed materials information. The recuperator may be a modular counter-flow helium-to-helium heat exchanger, and current technology for the expected temperatures and pressures of operation is relatively mature.

Once appropriate materials have been designated for NGNP use, it will be necessary to gain ASME Boiler and Pressure Vessel (B&PV) Code acceptance of those materials at the desired operating conditions. To achieve B&PV Code acceptance, specific material information must be submitted to the appropriate subcommittees and will require significant justification. Once the material is accepted in Section II, it must also be submitted for construction approval in Section III, Subsection NH. 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.

A1.4.2.2.1 Graphite Testing and Qualification. 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.

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. A strategy for the selection and acquisition processes, and material receipt and storage requirements for the purchased graphite is being developed.

Engineers at INL, in consultation with graphite experts at ORNL, have started an ATR creep capsule design. Prior Oak Ridge Research Reactor and Idaho Engineering Test Reactor graphite creep test capsule designs are being used as the basis for the new design. The graphite samples will be loaded under compressive stress and irradiated at representative temperatures. In addition to creep rate data, PIE of the control samples will yield valuable irradiation-effects data.

Mathematical models that describe and predict the behavior of nuclear graphite under neutron irradiation must be developed. Such models must be based on physically sound principles and must 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.

Significant dimension and material property changes can occur in graphite subjected to neutron irradiation, as illustrated in Figure A1.7 for the old H451 material. 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 2004. Each of the 18 rabbit capsules contained a SiC temperature monitor. PIE of the samples began at ORNL in FY 2005.

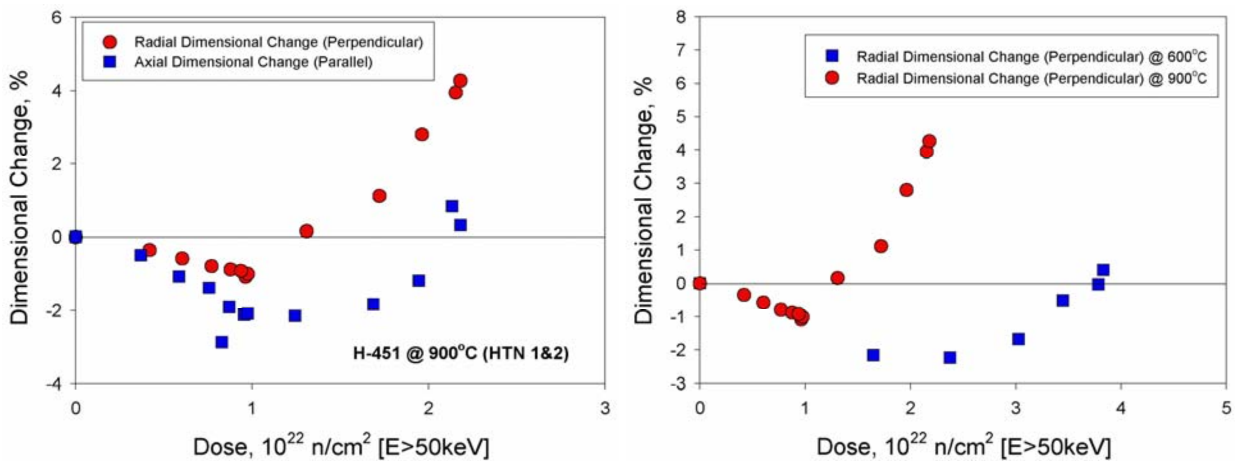


Figure A1.7. H-451 graphite dimensional changes as a function of orientation and temperature.

In addition, there is little data for the irradiation behavior of graphite at temperatures greater than 1,000°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 1,200°C.

A1.4.2.2.2 High-Temperature Design Methodology. 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 Subgroup on Elevated Temperature Design).

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. 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. Figure A1.8 shows some of this work in progress.

A1.4.2.2.3 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. Therefore, the NGNP Materials R&D Program must initiate a presence at ASTM and ASME B&PV Code meetings at the relevant committee and subcommittee levels in order to incorporate new materials, and/or extend the application of materials presently in the ASME B&PV Code, and/or further develop test standards.

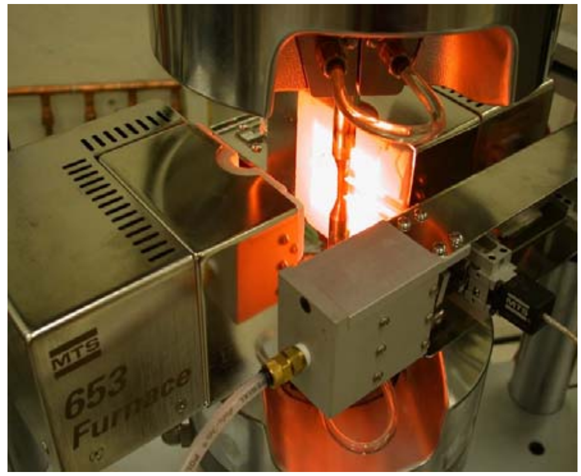


Figure A1.8. Creep-fatigue test in progress in air at 1,000°C at the INL (furnace opened to show specimen).

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. 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. Standard test methods for graphites and composites 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 it is also developing several standard test methods for graphites (for example, crystallinity by x-ray diffraction [XRD], surface area, thermal expansion, fracture toughness, and graphite oxidation). INL and ORNL will also support the formation of an ASTM working group on Si_fC/SiC composite testing development, and will ensure that setting guidelines for testing tubular Si_fC/SiC structures proceeds.

A1.4.2.2.4 Environmental Testing and Thermal Aging Project. The three primary factors that will most affect the properties of the metallic structural materials from which the NGNP components will be fabricated are (1) the effects of irradiation, (2) high-temperature exposure, and (3) interactions with the gaseous environment to which they are exposed. An extensive environmental testing and thermal aging evaluation program is needed to assess the effects of these factors on the properties of the potential materials to qualify them for the service conditions required.

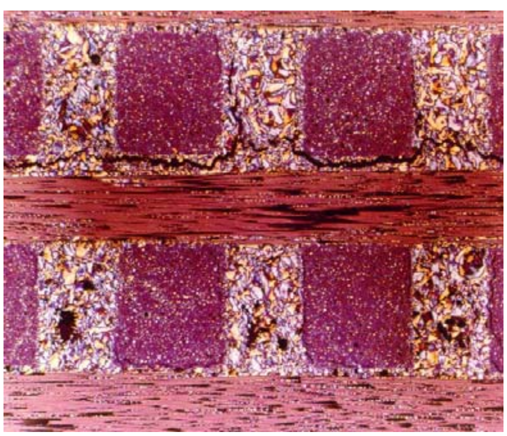
Procedures for the evaluation of aged and “service-exposed” specimens will be developed. Properties evaluation will be performed on a limited number of materials, including Alloy 617, Alloy 800H, and Alloy X, that have been aged at temperatures as high as 870°C for long durations in helium. It is expected that aging exposures of more materials will be performed to at least 25,000 hours. 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.

The out-gassing of 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 that the testing proposed in various parts of the program is performed in environments having consistent chemical potentials relative to the expected environment. 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.

A1.4.2.2.5 Test and Qualify Reactor Pressure Vessel and Core 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 that was shut down last year at the Ford Test Reactor at the University of Michigan.

A1.4.2.2.6 Composites Research and Development. The composites R&D program is directed at the development of C_f/C and Si_fC/SiC composites for use in selected very-high-temperature/ very-high-neutron fluence applications such as control rod cladding and guide tubes (30 displacements per atom [dpa] projected lifetime dose) where metallic alloys are not feasible. It is believed that Si_fC/SiC composites have the potential to achieve a sixty-year lifetime under these conditions. The usable life of the C_f/C composites will be less, but their costs are also significantly less. This program will eventually include a cost comparison between periodic replacement of C_f/C materials and use of Si_fC/SiC composites.

Unlike monolithic materials, composites are engineered from two distinct materials using complicated vapor infiltration techniques. Therefore, 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. In addition, 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). A typical C_f/C composite cross-section and weave pattern is shown in Figure A1.9.



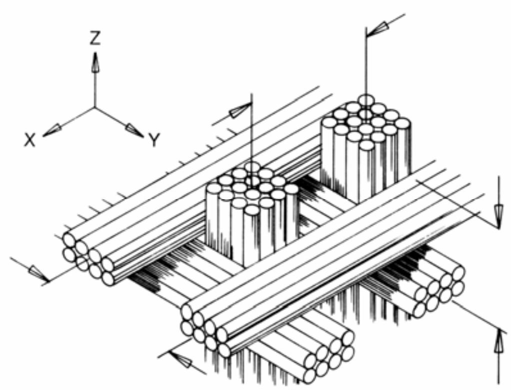


Figure A1.9. Typical C_f/C composite cross-section and weave pattern.

A1.4.2.2.7 Data Management and Handbook. The NGNP data will be managed by incorporating final materials data into the Generation IV Materials Handbook. The Materials Handbook is being developed in the Materials Crosscutting Program. Existing Materials Handbooks will be examined to determine what information might be extracted and incorporated into the Generation IV Materials Handbook. Once fully implemented, the Generation IV Materials 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. Near-term activities in this area will include assembling and inputting existing data on materials of interest to NGNP.

A1.4.2.2.8 Additional Materials Research and Development. Additional materials R&D will also include the power conversion unit and generator; reactor pressure vessel emissivity; metallic reactor internals; IHX and piping fabrication; hot duct liner and insulation; and valves, bearings and seals.

A1.4.2.2.9 Energy Transfer. Molten salt is a leading candidate for efficient transfer of heat from the NGNP reactor to the hydrogen production plant. Earlier developments at ORNL for the Molten Salt Breeder Reactor (MSBR) program established materials for heat transfer components (pumps, heat exchangers, pipes) for nuclear service up to 750°C and a thirty-year lifetime, and led to the development of an optimized alloy for this application: Hastelloy-N. Work done even earlier for the Aircraft Nuclear Propulsion (ANP) program focused on temperatures up to 870°C for shorter periods of time (~ 1 month) using Inconel. For a number of reasons, neither Hastelloy-N nor Inconel will be suitable for the NGNP heat transfer loop. The heat transfer loop application does not impose radiation damage constraints, but the NGNP loop application does impose very high temperatures. Heat-transfer loop lifetimes can be shorter than those associated with the MSBR application (~30 years), but much longer than those associated with the ANP application (~1 month). For t his reason, a new optimization of materials for the heat transfer loop is required. Specific lifetime and temperature requirements have not been established for the NGNP heat transfer loop, but preliminary documents indicate that the peak temperature will be between 850°C and 1,000°C. Because salt has a much higher heat capacity than helium, it is expected that an NGNP heat transport with salt (rather than helium) will establish the lowest peak temperature requirement for this loop.

Recent studies highlight the advantages of a molten salt coolant (INL 2005) for the intermediate heat transport loop and indicate the primary importance of materials compatibility at the higher temperatures envisioned for the NGNP loop. An assessment of molten salts as a primary reactor coolant established the factors to be used in selection and ranking of molten salt coolants (ORNL 2006). A report that extends this analysis to the additional salt options possible for a secondary coolant application is being prepared. The scope of work described in Section A1.4.2.10 is an integrated study of salt chemistry and materials compatibility (as recommended in previous reports).

A1.4.2.2.10 Materials Evaluation and Recommendations. It is recommended that the first year of work focus on batch-exposure tests under highly controlled conditions to evaluate different salt-material combinations. The materials focus will be on monolithic alloys and selected ceramics. Recommendations for evaluation of clad-materials will be formulated for future tests. First priority is given to the mature high-temperature nickel alloys identified as candidates for use in the NGNP reactor: Alloy 617, and Hastelloy-X/XR. Less-mature alloys will be tested on a limited basis after a more thorough review of alloy options is conducted early in the year.

Three eutectic salts are recommended for evaluation: (1) an alkali fluoride (FLiNaK = LiF-NaF-KF, 46.5–11.5–42 mol%), (2) an alkali fluoroborate (KF-KBF_4 25-75 mol%), and (3) an alkali-halide (LiCl-KCl, 59-41 mol%). These salts represent the basic classes of salts that exhibit the necessary stability and properties for extreme temperature service. They are also relatively inexpensive salts, and each one exhibits a unique corrosion behavior. Differences in corrosion within each class of salt (fluorides, fluoroborates, and chlorides) are much less distinct.

A1.4.2.2.11 Materials Testing. Only salts that have been thoroughly purified will be used, and these salts will be tested under rigorous cover-gas purity that is continuously monitored. The purification requirements for FLiNaK have been establish at ORNL, but the requirements for LiCl-KCl and KF-KBF_4 will be defined as part of this study.

The oxidation-reduction (redox) state of the salt will also be controlled and varied as a parameter of the test matrix. Redox sensitive rare-earth species will be used to set and monitor the redox state of the salt. Maintenance of the salt in a moderately reducing condition may permit the use of the most economical monolithic material for high temperature service.

The level of corrosion products in the salt will be measured by in-line electrochemical methods, and will be confirmed by periodic samples taken from the system. The intensity of corrosion will be indicated by the levels of dissolved alloy constituents and by analysis of specimens at the end of the test.

Some exposure of ceramic materials will also be conducted. Carbon materials are known to exhibit good stability with respect to the salts under consideration. However SiC materials are reported to exhibit a wide variety of responses. A review of this literature will be conducted to recommend the samples for study in FY 2007. No control of the redox condition of the salt is planned for the tests with ceramics.

A more detailed technology development plan for materials for salt-service will be developed in conjunction with the other high-temperature materials development conducted by DOE.

A1.4.3 Design and Evaluation Methods

The NGNP Design and Evaluation Methods (D&EM) Program will develop the state-of-the-art software analysis tools and supporting data required to calculate the behavior of the NGNP system during normal and off-normal scenarios. The software tools discussed here include those necessary to calculate the neutronic and thermal-hydraulic behavior, the interactions between neutronics and thermal-hydraulics, and the structural behavior where necessary. The NGNP D&EM 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 it will include their feedback through a peer-review process.

The D&EM R&D implementation methodology shown in Figure A1.10 is as follows:

1. Utilize a phenomena identification and ranking table (PIRT) to select the most challenging scenarios together with the dominant phenomena in each
2. Internally validate the software tools and data required to calculate the NGNP behavior in each scenario
3. Externally validate the software tools via non-NGNP Project nuclear engineering community participation in international standard problems
4. Perform R&D through GIF-member and NGNP Project collaborations centered in I-NERIs
5. Perform R&D through university and NGNP Project collaborations centered in NERIs or GIF PMB agreements
6. Develop software when validation findings show that certain models are inadequate
7. Analyze the operational and accident scenarios
8. Review the global process and the process ingredients, using experts outside the program.

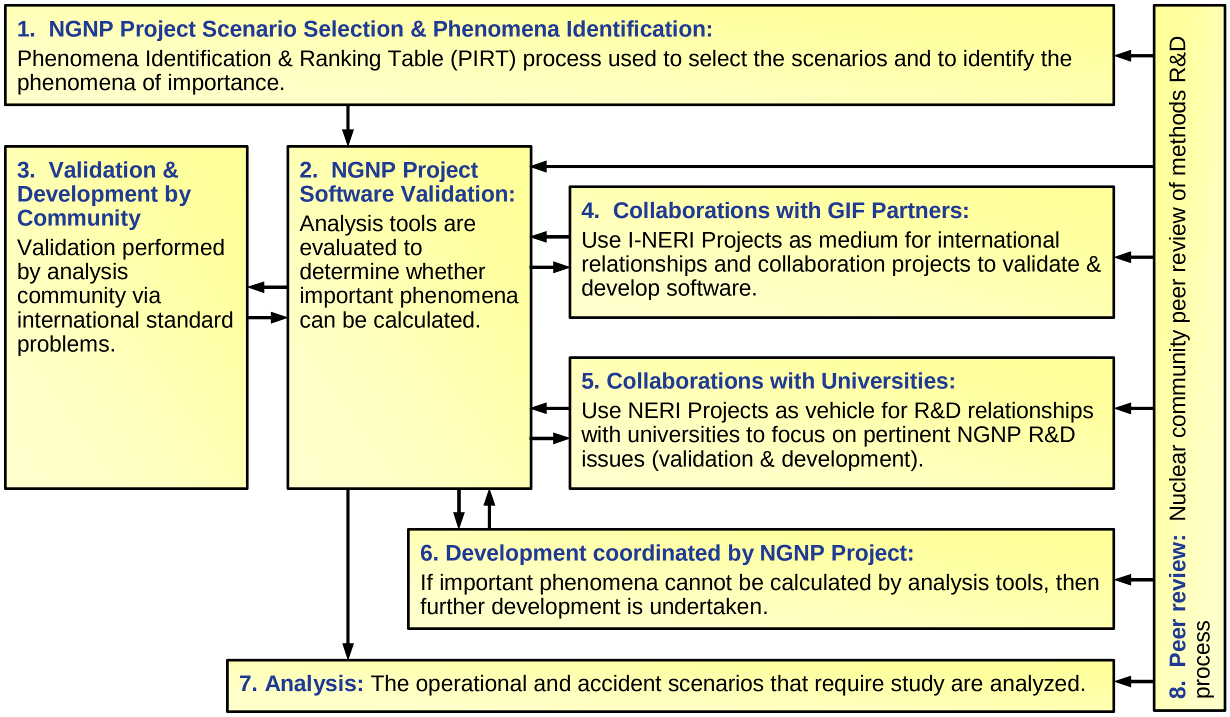


Figure A1.10. NGNP D&EM R&D process.

A rigorous PIRT analysis of the NGNP has not been performed because the design has not yet been identified. However, based on the accumulated knowledge of the AGR vendor community and engineering judgment, a “first-cut” PIRT has been defined and used to specify the R&D for FY 2005 and subsequent years. Once a reactor design is specified, the design and evaluation methods development and R&D requirements will be aligned with the design. The PIRT has identified a number of important phenomena (INEEL 2004). Based on these phenomena, R&D will be focused on five major tasks:

1. Computational fluid dynamics (CFD) code validation experiments: lower plenum, hot channel, and reactor cavity cooling
2. Validation of thermal-hydraulic software, e.g., CFD calculations of exit fluid temperature from hot-channel and lower plenum turbulence; core analysis methods development
3. Core physics methods development
4. Nuclear data tasks
5. Liquid salt-cooled methods development and design assessments.

A1.4.3.1 Validation of Thermal-Hydraulics Software and Computational Fluid Dynamics Codes Including Computational Fluid Dynamics Validation Experiments

The thermal-hydraulics of the NGNP encompasses the heat generation by the fuel; its transport to the helium coolant; and the laminar, transition, or turbulent flow of the helium as it flows from the upper plenum through the core, into the lower plenum, then out the exit duct to the IHX or power generation vessel. Also included are the heat losses from the reactor vessel during normal operation as well as accident scenarios that may occur from failures in the system. The system designed to remove the heat in the event of an accident, the reactor cavity cooling system, is also included in the thermal-hydraulics of the NGNP.

Advanced simulation tools are available to simulate turbulent flow and heat transfer in complex systems. These tools must be validated for application on the NGNP. CFD codes are needed to simulate regions of complex turbulent flow in the plant. Because of the size and complexity of the plant, thermal-hydraulics systems analysis codes will also be applied, in conjunction with CFD codes, to analyze the plant.

The high-priority research areas identified in the “first-cut” PIRT include (1) the core heat transfer, (2) mixing in the upper plenum, the lower plenum, hot duct, and turbine inlet, (3) the heat transfer in the reactor cavity cooling system (RCCS), (4) air ingress following a system depressurization, and (5) the behavior of the integral system during the key scenarios, including the contributions of the BOP. These R&D areas are outlined in Table A1.4 together with a summary of the key needs.

A1.4.3.1.1 Computational Fluid Dynamics Code Validation Experiments. The experiments that stem from the areas identified in Table A1.4 are discussed at a summary level below. More detailed discussion can be found in INEEL 2004. Some potential issues identified to date include hot streaking in the lower plenum evolving from hot channels in the core, 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. 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.

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 (see Figure A.1.11); 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. Heat transfer experiments will be developed and accomplished for the same purposes. Existing databases from experiments, direct numerical simulations (DNSs), and large eddy simulations (LESs) will also be utilized where appropriate.

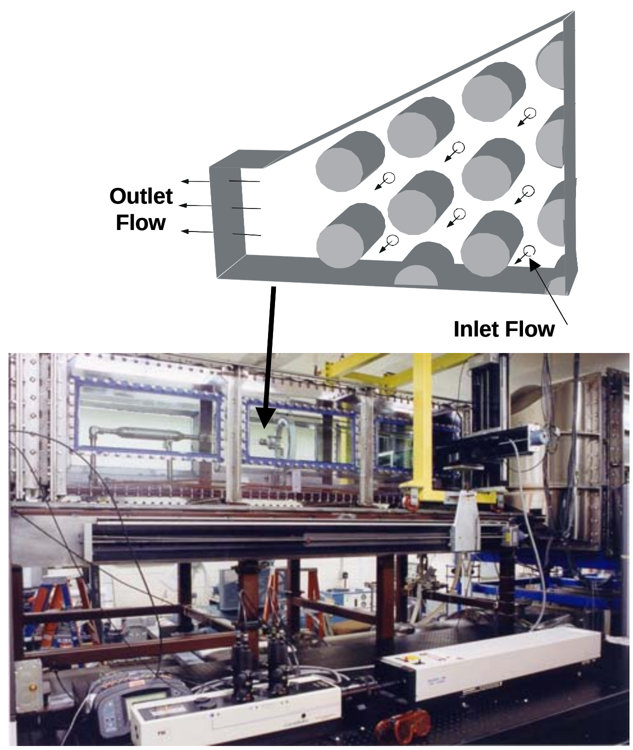


Figure A1.11. MIR flow system and a conceptual model design to study important flow features in a VHTR LP.

Table A1.4. NGNP thermal-hydraulics D&EM R&D 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, and systems analysis codes (S)	The core heat transfer, both with cooling flow (operational conditions) and without cooling flow (depressurized conduction cooldown [DCC] and pressurized conduction cooldown [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 and Lower Plenums (UP and LP)	HTTR UP and LP, HTR-10 UP and LP, Matched-Index-of-Refraction (MIR), heated experiments, scaled vessel, jets and cross-flow data, UP experiments, system performance enhancements.	E and CFD	Circulation in the UP 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 LP mixing determines both the temperature variations and the maximum temperatures that are experienced by the turbine blades, the LP, hot duct, and power generation vessel structural components. The LP mixing will determine the material selection and configuration in the NGNP LP, 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, and 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, and 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 period of time. It must be demonstrated that the system would 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 BOP components (IHX, turbine, compressor, reheater), analyses of pre-conceptual design, conceptual design, preliminary design, and final design	E, CFD, and 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.

The purpose of the following experiments is to produce data to validate the NGNP CFD software.

- Core heat transfer experiments:
  - Turbulence and stability data from vertical cooling channels. Experiments defined to study (a) the natural circulation-driven jets that emerge from the hottest channels and that may impinge on the vessel upper head structures during a pressurized conduction cooldown scenario and (b) the factors that influence the hottest coolant exit temperatures during normal operation.
  - Bypass flow. Experiments designed to characterize the variation in the bypass flow during the lifetime of the NGNP as well as the influence of the bypass flow on other variables of importance, e.g., cooling flow and fuel temperatures.
  - Exit flows in pebble beds. The factors that influence the exit flow from the pebble bed into the downstream plenum are characterized and correlated to define their effect on downstream power conversion equipment.
- Upper plenum (UP) and lower plenum (LP) fluid behavior experiments:
  - Fluid dynamics of LPs. A series of isothermal experiments designed to study the core coolant jet interactions and mixing as a function of location and flow conditions (flow rates and wall interactions) using the MIR facility shown in Figure A1.11.
  - Heated flows in LPs. An experimental series that complements item 2 in Table A1.4 by measuring the influence of the buoyant contributions when exit jet temperatures are considered.
  - Interactions between hot plumes in a UP and parallel flow instabilities. Validation data are measured to characterize the coolant jet plumes, and their interactions, in the UP during a pressurized conduction cooldown scenario. The plumes from the hottest channels will impinge on the UP structural members.
- Air ingress experiments:
  - Heat transfer and pressure drop of mixtures of air and helium. Intended to supplement existing data (e.g., International Atomic Energy Agency benchmark data) by recording measurements specific to the intended NGNP design; the experiments provide scaled data for the depressurized conduction cooldown scenario when air component gases move into the lower plenum and hence the core by diffusion.
- Larger scale vessel experiments:
  - Behavior in the core and in the plenums, and the interactions between them. A series of experiments will be performed in a scaled mockup of the near-final or final NGNP design to characterize the flow behavior specific to the defined geometry, e.g., characteristic turbulent behavior stemming from the area expansions, contractions, and changes in direction.
- Integral experiments:
  - Phenomena interactions throughout HTTR and HTR-10 test reactors. The only integral advanced gas-cooled experimental facilities in the world (prismatic and pebble bed respectively) will be used to obtain representative data that includes various phenomena interactions throughout the reactor system to enable validation of the phenomena interaction calculations to be performed.
- RCCS experiments:
  - Behavior of RCCS. Data representative of the behavior of this key heat removal system will be measured for validation purposes to characterize the system behavior during operational, DCC, and pressurized conduction cooldown (PCC) conditions. These data are crucial in

validating the models that directly influence the fuel temperatures both at steady state and accident conditions.

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.

A1.4.3.1.2 Thermal-Hydraulic Design Methods Development, Validation, and Analysis. The modeling strategy chosen for this effort is to make use of both thermal-hydraulic systems analysis and CFD software codes. The reference codes chosen are the Reactor Excursion and Leak Analysis Program (RELAP) RELAP5-3D© systems code and the Fluent and STAR-CD CFD codes. However, other codes, such as the Graphite Severe Accident Code (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 (such as, the IHX, turbine, compressor, reheaters, etc.). CFD software is needed to analyze the 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, the IHX or turbine inlet region, and the RCCS.

Two commercial CFD reference codes (Fluent and STAR-CD) are presently being used, and a university-developed or a national laboratory-developed CFD code may also be used. However, it is suspected that none of them will meet all of the NGNP analysis requirements, and thus some modifications will be required. Consequently, the following 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
- 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 currently available CFD software. As the CFD software is validated, it may become necessary to add new turbulence models or pursue other modeling strategies, such as LES or DNS, thus following Track 2. Track 3 is designed to ensure that more efficient and capable simulation software will be available in the future.

The key technical issues identified by the first-cut PIRT, as summarized in Table A1.4, are the basis for defining the code development, validation, and analysis R&D program. The R&D activities are organized based on the five key technical issues listed below:

1. Core heat transfer model validation, development, and analysis: (a) Convective Heat Transfer during Normal Operation—CFD analysis and systems analysis calculations, and experimental data to be used for validation; (b) Convective Heat Transfer during PCC; (c) Axial and Radial Conduction during PCC and DCC; and (d) Core bypass

2. UP and LP coolant flow mixing validations, development, and analysis: (a) LP (CFD model validation) and (b) upper plenum flow (validation of coupled RELAP5-3D/Fluent model) (Figure A1.12)

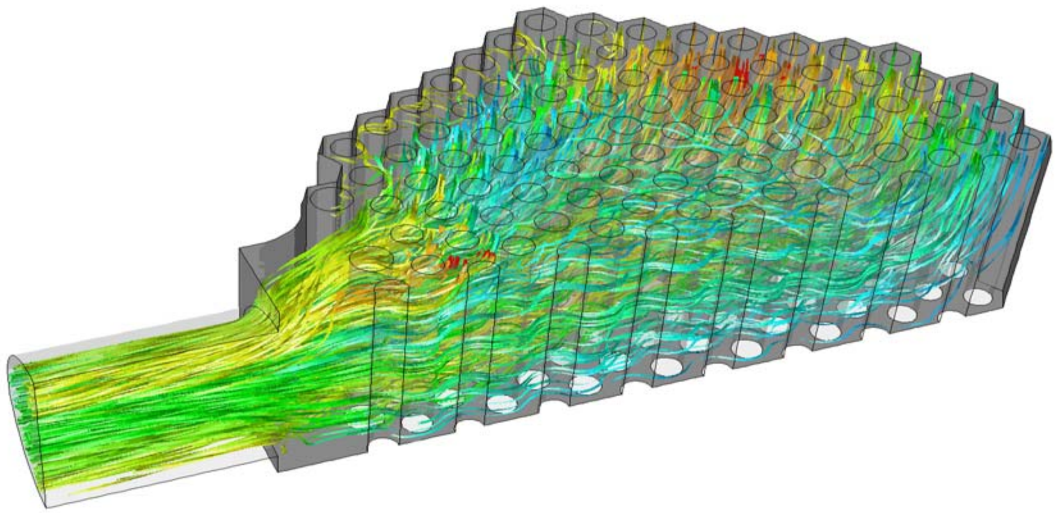


Figure A1.12. Preliminary CFD calculation of mixing in lower plenum (courtesy of Fluent, Inc).

3. RCCS validation, development, and analysis
4. Air ingress and Fission Product Release validations, development, and analysis including (a) air ingress scenarios calculations and RELAP5-3D multi-species model extension, and (b) linking of RELAP5-3D to coupled models Particle Fuel Model (PARFUME) and VICTORIA
5. Integral system behavior validations, development, and analysis.

A1.4.3.2 Reactor Kinetics and Neutronics Analysis Development

The design and operational analyses of the NGNP must have the ability to carry out the following reactor physics computations: (1) fuel cell and assembly spectrum cross section generation calculations to produce effective nuclear parameters for subsequent global reactor analysis, (2) static reactor analysis for core design and fuel management, (3) kinetics, thermal module coupling, and feedback, (4) material-neutronics interface, and (5) V&V and ongoing improvement of code suite. This section 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.

INL and ANL will lead 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. It is anticipated that the bulk of the code development effort will be completed in the first few years of the overall effort (through FY 2009). After that, code maintenance, validation, and application to the ongoing NGNP design effort will continue for the duration of the project.

A1.4.3.2.1 Fuel Cell and Assembly Spectrum and Cross Section Generation. 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 required form. Current 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 have been calculated by INL's COMBINE code or by MICROX-2, both of which need enhancements to properly handle certain phenomena that are characteristic of graphite-moderated reactors with highly heterogeneous fuel arrangements. The following subtasks have been identified under this task:

- Develop a method for improved treatment of double heterogeneity using improved Dancoff Factors.
- Develop the interface between a spectrum code and a pebble bed reactor core simulator. 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).
- Identify or develop an assembly code for prismatic block cross-section generation. The lattice transport code, used for group constant generation, must be able to treat the double heterogeneity properly and account for spectral variations across the basic lattice unit via appropriate neutron transport computations.

The analysis of the VHTR steady-state and transient conditions requires weighted few-group cross sections that accurately represent the complex resonance shielding effects in the doubly-heterogeneous structures consisting of fuel particles and pebbles for the pebble bed concept and fuel particles and fuel compacts for the prismatic concept. Over the past several years, a new fuel assembly and cross-section generation sequence called TRITON has been developed as part of the Standardized Computer Analyses for Licensing Evaluation (SCALE) computer software system. This sequence uses a point-cross section approach for very accurate resonance processing that has the ability to treat double heterogeneous fuels and has a general-geometry transport methodology. Therefore, this code represents a significant improvement in methodology over previous traditional methods. The following subtasks have been identified:

- Apply the CENTRM-based TRITON methodology for improved resonance processing for use with core-level analysis codes (such as PEBBED, DIF3D, BOLD VENTURE, and Purdue Advanced Reactor Core Simulator [PARCS]) for pebble bed and prismatic designs. Included is the development of a unified coupling strategy for the cross-section processing codes and efficient improvements for on-line generation of cross sections with feedback.
- Evaluate and improve CENTRM/TRITON for use as a cross section/spectrum code for pebble bed and prismatic block configurations as needed to improve predictions based on benchmark evaluations.
- Investigate improvements in the treatment of scattering in the resonance region, which nearly all codes treat with a simplified model. Improvements in the treatment of scattering in the resonance region can result in better predictions of Doppler feedback effects.
- Update cross-section libraries to incorporate ENDF/B-VII data in order to utilize the latest cross-section evaluations and covariance data using the AMPX system (a modular code system for generating coupled multigroup neutron-gamma libraries). Continue to update master cross-section libraries as improved evaluations become available.

A1.4.3.2.2 Static Reactor Analysis for Core Design and Fuel Management. The fundamental quantity in reactor physics analysis, determining all other aspects of core behavior, is the neutron flux. Extremely accurate calculations of the neutron flux, accounting for great geometric detail, can be made with Monte Carlo codes such as the Monte Carlo N-Particle Transport Code (MCNP). However, Monte Carlo codes are still prohibitively expensive for use in iterative design calculations involving evaluation of local power distributions and small reactivity effects. Nor can current coupled Monte Carlo-depletion codes be applied to the pebble bed reactor (Figure A1.13). This section focuses on development of deterministic codes for use on pebble bed static reactor core design and fuel optimization. It includes work in the following areas:

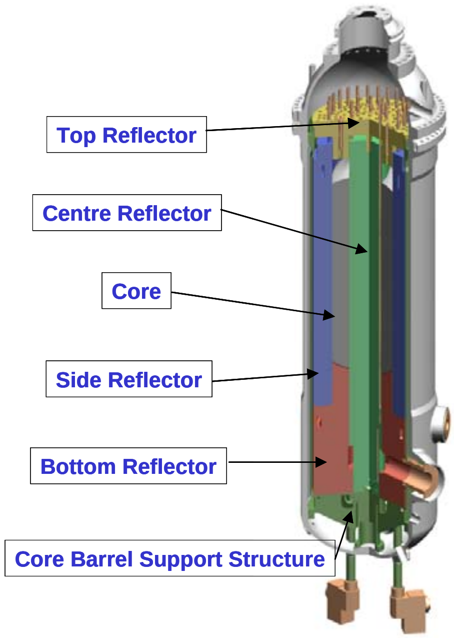


Figure A1.13. Sketch of the PBMR.

- *Fuel management and design optimization*: PEBBED possesses an advanced optimization routine (a genetic algorithm) that allows automated searches for optimal core designs and fuel loading patterns. Such methods have not yet been developed for the prismatic reactor and will be developed. Core optimization and fuel loading will also be automated to produce viable cores within practical time limits. One or more advanced optimization approaches will be explored and implemented in DIF3D/REBUS.

- *Neutron Transport*: The pebble bed reactor contains a large gas plenum above the core through which pebbles are dropped. A transport calculation will be developed to accurately predict neutron transport in these regions. Also, a NERI project led by F. Rahnema, of the Georgia Institute of Technology, is investigating neutron transport techniques accurate treatment of gas plena and control rod regions in the pebble bed reactor. Several codes will be explored for use in three dimensional deterministic transport calculations for some applications, such as accurate gamma transport and deep shielding. The codes include Deterministic Core Analysis based on Ray Tracing (DeCART), Attila, and Even Parity Neutral Particle Transport (EVENT)

- *Isotope Depletion*: The DIF3D/REBUS-3 code system is capable of multi-group flux and depletion calculations in hexagonal-Z geometry, which will be adapted to prismatic NGNP reactor problems with limited effort compared to other codes.

- *Pre-asymptotic core analysis in the pebble bed reactor*: A theoretical formulation for pre-asymptotic core analysis is under way at 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*: A method and code will be developed that link together depletion zones along the true flow path of pebbles in a pebble bed reactor, which is not strictly axial, particularly near the discharge tubes.

The SCALE system includes a Monte Carlo code, KENO, which is based on group cross sections that is much more efficient than continuous energy Monte Carlo codes. Therefore, KENO represents a good alternative for reference solutions and detailed calculations incorporating the exact geometry of pebbles and compacts, while avoiding the need to model each fuel particle. KENO has also been coupled with Oak Ridge isotope generation (ORIGEN) to allow detailed depletion calculations. The development of an independent analysis capability using TRITON/PARCS/GRSAC (or THERMIX) will also be considered. Work to be performed includes:

- Applying KENO code systems to perform independent analysis of benchmark problems including depletion problems. The result will be detailed power distributions and isotopic compositions to compare to the deterministic solution methods.
- Evaluating ORNL-developed neutron transport methodologies for improved analysis capability with a high-level of detail (after FY 2007).
- Evaluating ORNL-developed advanced methods for the treatment of the interaction of neighboring fuel regions to improve the efficiency and accuracy of the calculation process. This and similar approaches will be examined for use in improving pebble bed and prismatic design predictions and to incorporate reflector effects.
- Developing of advanced system-wide optimization methods for integrated core and plant design. An approach has been developed for application to advanced reactor systems based on Genetic Algorithms and will be considered for enhancement and utilization to reduce the design and trade-study effort for NGNP.

A1.4.3.2.3 Kinetics Thermal Module Coupling and Feedback. High-fidelity kinetics methods are important for core transients involving significant variations of the flux shape, but these methods have not been systematically applied to graphite-moderated, helium-cooled reactors. In the future, integrated thermal-hydraulics and neutronics methods will 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.

A cooperative research effort between Penn State University and PBMR, 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 Nodal Expansion Method (NEM) code. INL has been invited 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.

Nodal diffusion and transport kinetics capabilities have been developed for the DIF3D code. These capabilities have been successfully applied for transient analysis of thermal reactor systems by integrating them in a system analysis code, SASSYS. Initial estimates 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/Advanced Thermal-Hydraulic Energy Network Analyzer (ATHENA), that can be utilized for the analysis of the NGNP.

The GRSAC code is widely recognized as a tool to analyze gas-cooled reactors. While alternative codes such as Methods for Estimation of Leakages and Consequences of Releases (MELCOR) exist, GRSAC is well suited for design studies, independent analysis, and benchmarking activities. This code can be incorporated into the current code suite to provide the following capabilities:

- Incorporation of GRSAC into code suite and application to reference and benchmark problems for independent results and for verification of other analysis tools (such as RELAP and THERMIX)
- Development of an overall system design optimization tool
- Assessment of uncertainties in safety-related parameters.

A1.4.3.2.4 Material-Neutronics Interface. Of particular importance is the change in material properties caused by radiation. For example, the thermal conductivity of graphite is significantly degraded 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 imply, for example, 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.

A1.4.3.2.5 Validation, Verification, and Ongoing Improvement of Code Suite. The resulting suite of deterministic codes developed above will be verified against Monte Carlo and deterministic codes, and against integral experiments. 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 precalculated, multigroup 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).

In October 2005, a reference set of benchmark problems were developed to investigate potential errors in the production of cross sections using spectral codes. Work to be done includes:

- Performing an analysis of numerical benchmark problems that were developed to verify the cross section/spectrum code in comparison with continuous energy MCNP. The benchmarking that has been proposed by INL and ORNL for the pebble bed will allow a systematic investigation and evaluation of the design methods used by INL (COMBINE and MICROX) with results from SCALE/TRITON. Areas identified as needing improvements will be addressed.
- Developing a basis for the validation of the decay heat predictions by utilizing available measurements and applying detailed isotopic capabilities of the ORIGEN-S code.

A1.4.3.3 Nuclear Data Measurements, Integral Evaluations, and Sensitivity Studies

Accurate differential nuclear data libraries and well-characterized and accurate integral benchmark information are 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 above needs will be satisfied through the activities discussed below.

A1.4.3.3.1 Sensitivity Studies. ANL, in collaboration with INL, will quantify uncertainties in computed core physics parameters that result from propagation of uncertainties in the underlying nuclear data used in the various modeling codes. This study will aid in further quantifying the need for additional cross-section measurements and provide a guide in planning future measurements and evaluations. Formal sensitivity and uncertainty analysis will be performed to identify the nuclides that contribute to calculated uncertainties and 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 multigroup covariance data (where available) to derive propagated uncertainties in computed integral reactor parameters arising from the nuclear data. Parameters to be evaluated include reactivity, peak power, reaction rate ratios, nuclide inventory, safety coefficients, etc. ORNL has also developed a covariance library based on integral data as well as an approach to provide covariance data using existing cross section evaluations. This effort will be continue through FY 2007.

A1.4.3.3.2 Integral Nuclear Data Evaluations. The computer codes used in NGNP design and safety analyses must be benchmarked against appropriate experimental data. During FY 2004, under DOE Generation IV crosscut funding, INL and ANL studied various experimental and prototypical HTGRs developed since the 1960s to assess their potential as benchmarks.

The HTR-10 test reactor (Figure A1.14) was chosen for the first pebble bed reactor benchmark that will undergo full evaluation, and HTTR and Very-High-Temperature Reactor Critical (VHTRC) were chosen for the first block-reactor benchmarks.

The next steps involve evaluation and documentation of the identified facilities to provide benchmark specifications accepted for validation of physics modeling codes. The work was initiated 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, and a draft report sent to the publishers in February of 2006. INL, which provides leadership for the IRPhEP Technical Review Group, and ANL will contribute NGNP-specific benchmarks evaluated under this R&D Plan.

Initially, INL will evaluate the HTR-10 test reactor and possibly the PROTEUS reactor. 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.

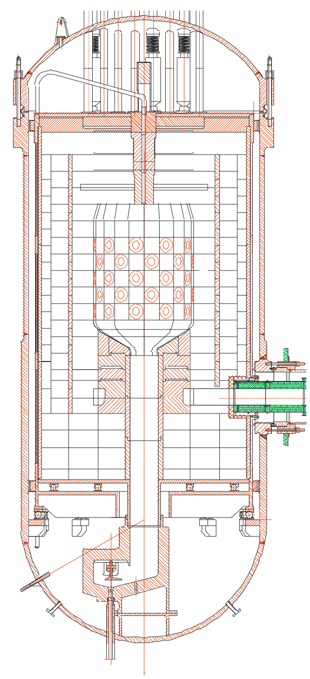


Figure A1.14. Schematic diagram of HTR-10 core and vessel.

The use of previous experiments for the validation of the cross section and core analysis methods are essential. Some of the past experiments are directly applicable with current design parameters and enrichment ranges. It is less clear that other data can be applied given significant differences in the design parameters. The Tools for Sensitivity and Uncertainty Analysis Methodology Implementation (TSUNAMI) developed at ORNL is widely accepted for use in determining the applicability of experiments and can be used to demonstrate the usefulness of these older experiments. The following tasks are proposed in this area:

- Participate in the evaluation and independent review of benchmark problems under the IRPhEP
- Apply ORNL TSUNAMI experiment applicability methodology to determine applicability of older experimental data to current design.

A1.4.3.3.3 Differential Nuclear Data Measurements. The current NGNP design will have a harder thermal neutron spectrum and two to three times the burnup of a light water reactor. As a result, improved cross section measurements are needed in certain neutron energy ranges for some isotopes, in particular ^{240}Pu, ^{241}Pu, and ^{242}Pu.

INL, in partnership with ANL and various university and international collaborators, will conduct a research program of measurements for the actinides of interest at the ANL Intense Pulsed Neutron Source (IPNS). Through FY 2009, the differential data measurement campaign will be focused on development of data for ^{240}Pu and ^{242}Pu, followed by measurements for ^{241}Pu. After this initial campaign is complete and the data published, attention will turn to the higher actinides with specific priorities set by the cognizant international nuclear data working groups.

In addition, the Oak Ridge Electron Linear Accelerator (ORELA) facility has the ability to perform very precise cross section measurements and is currently undergoing refurbishment to improve reliability. Based on the needs and requirements determined by the sensitivity studies, measurements can be performed at ORELA and evaluated at ORNL to support the differential nuclear data work.

A1.5 PROJECT COST AND SCHEDULE

A1.5.1 Project Budget

The current preconceptual estimate for the NGNP Prototype Project establishes the initial estimated cost range for planning at the beginning of the Project Definition Phase.

A1.5.1.1 Basis of Estimate

Currently, no design exists for the project; therefore, the conceptual design for the GT-MHR by General Atomics was used to provide a basis, using a specific analogy cost estimating approach for developing this planning estimate. A form of the GT-MHR is the principal reference concept for both the Generation IV VHTR and the NGNP.

This cost estimate process methodology started with the GT-MHR conceptual design cost estimate, as documented in *Gas Turbine-Modular Helium Reactor (GT-MHR) Conceptual Design Description Report* (General Atomics 1996). The costs, by category, were adjusted using engineering judgment factors to account for differences between the four-module design of the GT-MHR and the single-module reactor of the NGNP Prototype. These costs were then escalated at documented inflation rates per year and spread by year of planned expenditure, based on a high-level planning schedule and engineering judgment.

Technology development costs are based on previous planning under the Generation IV R&D program, which was a general R&D program planned around crosscutting technology applicable to more than one of the six reactor concepts in the Generation IV Program. These costs have been adjusted by INL technology leads for the development areas, considering cursory, high risk and mitigation plans, and the preliminary planning schedule for the project. Changing from a general crosscutting R&D program to project-driven R&D is a major change in direction. Therefore, current R&D planning will be updated during FY 2006, and some changes to R&D costs can be expected.

Costs for the hydrogen development R&D were provided by the NHI National Technical Director. The cost for the Demonstration Facility is based on capital construction costs of \$80/standard cm^3/day × 200,000 cm^3/day assumed output for Thermochemical and \$100/standard cm^3/day × 200,000 cm^3/day assumed output for high-temperature electrolysis. The other project costs were based on a percentage of the construction costs and engineering judgment.

A1.5.1.2 Assumptions

The following assumptions were used in developing the planning cost estimate:

- The plant will be similar to the GT-MHR in that it will be a modular design, with a single reactor module, gas-cooled, prismatic reactor. Outlet temperatures will be 850 to 950ºC.
- Overall project management is provided by INL acting as the owner’s agent (i.e., DOE).
- Vendors will be subcontracted for the design/engineering and construction, licensing support, manufacturing, and construction management activities for the plant.
- Additional costs for a subcontractor project integrator are not included in the estimate.
- Craft labor rates at INL for nuclear construction work will be similar to those of GT-MHR, which were for Milwaukee, Wisconsin.
- INL will act as the owner’s agent and will be the license holder and operator for the plant.
- DOE Order 413.3 and DOE Manual 413.3-1 will be followed for management of the project.
- Sufficient funding will be available as needed and in a manner allowing optimum use of funds as estimated and scheduled.
- This estimate assumes that Preliminary Design will commence with a design competition in FY 2009. The selected vendor will continue with the Preliminary Design in FY 2009 to 2010. The project is scheduled to be completed by the end of FY 2018. Extension of this performance period will result in additional costs for inefficiencies and escalation.
- Construction is on a “greenfield” site and this estimate does not include costs for utilities, roads, and services to the site.
- In accordance with DOE Order 413.3, a formal Conceptual, Preliminary, and Final Design are required.
- A 4 × 10-hour day, 70-hour site construction workweek is assumed for the construction schedule.
- Conceptual Design will begin in the spring-summer of FY 2006.

A1.5.1.3 Cost Summary

The ten-year NGNP Prototype Project Cost Summary is shown in Table A1-5.

Table A1-5. NGNP prototype cost summary ($K).

Base construction cost	357,000
Design and licensing cost	580,000
R&D cost	557,150
Other project costs	124,000
Nuclear hydrogen development and demonstration	268,000
Total Project Cost	1,886,150

A1.5.1.3.1 Fiscal Year 2006 Cost Profile. The FY-2006 budget profile is shown in Table A1-6.

Table A1-6. FY 2006 budget profile for NGNP activities ($K).

Task	FY-06[a]
R&D	42,150[b]
Nuclear hydrogen R&D/Demo	25,000
Total	67,150

a. FY 2006 funding includes FY 2005 carryover funds.
b. Additional NGNP funding is included in the crosscutting section.

A1.5.2 Ten-Year Project Schedule

A1.5.2.1 Summary Schedule

The NGNP Summary Level Schedule is shown in Figure A1.15. The schedule shows the timelines for each of the major R&D areas and the design and construction activities. 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 principals of DOE Order 413.3-1
- Items shown under the Design and Construction will be managed by INL through industry contracts as determined in the Acquisition Strategy
- INL, as DOE's designated owner/operator, will be the NRC licensee
- The RPV will be needed approximately twelve to eighteen months into construction and a 38- to 40-month procurement schedule is needed for the RPV.

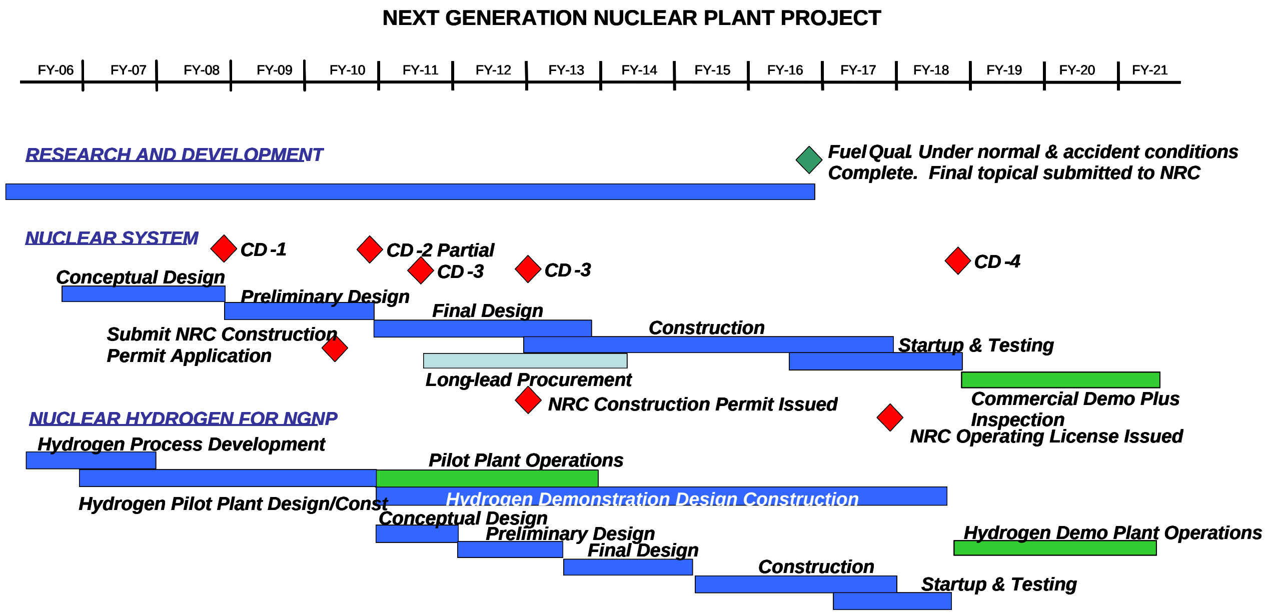


Figure A1-15. NGNP summary project schedule.

A1.5.2.2 Major Milestones

The major milestones for the NGNP project are dictated by the critical path through the design and construction activities of the project. Some high-risk fuel development and materials development activities also have the potential to impact the project schedule. These have been identified in the major milestones listed below:

FY 2006

- Initiate conceptual design/trade studies.

FY 2008

- Approve an alternative selection and cost range for CD-1.

FY 2009

- Complete energy transfer test vehicle
- Complete lab scale UCO TRISO fuel irradiation safety testing and Nuclear Fuel Industries, Ltd. fuel performance testing.

FY 2010

- Codify metallic materials ASME/ASTM
- Approve a performance baseline for CD-2.

FY 2011

- Approve procurement of long lead components for CD-2/3
- Approve preliminary safety analysis report by NRC
- Complete AGR fuel performance demonstration irradiation
- Complete fuel source term irradiation on AGR-3 and AGR-4.

FY 2012

- Finalize power conversion procurement specification
- Approve start of construction for CD-3
- Issue NRC permit to construct.

FY 2014

- Complete fuel performance demonstration safety tests on AGR-2
- Complete qualification irradiation on AGR-5 and AGR-6 and source term accident test.

FY 2016 – FY 2018

- Complete fuel qualification under normal accident condition
- Submit final topical report to NRC
- Issue NRC operating license
- Approve start of operations for CD-4.

A1.6 REFERENCES

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