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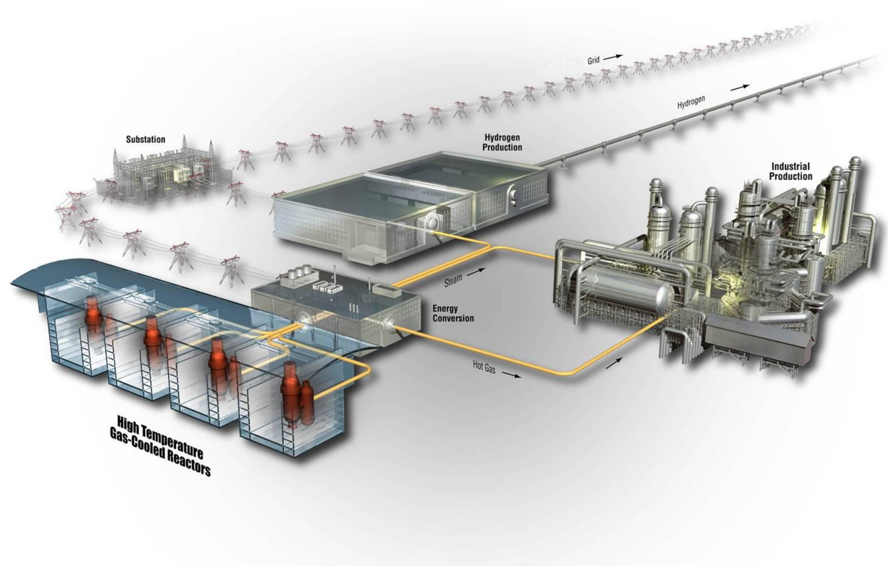
Revision 0

NGNP Program 2013 Status and Path Forward

March 2014



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NGNP Program
2013 Status and Path Forward

March 2014

Idaho National Laboratory
Next Generation Nuclear Plant Program
Idaho Falls, Idaho 83415

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Next Generation Nuclear Plant Program

NGNP Program
2013 Status and Path Forward

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March 2014

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EXECUTIVE SUMMARY

General

High temperature gas-cooled reactor (HTGR) technology can play an important role in the energy future of the United States by extending the use of nuclear energy for nonelectrical energy production missions, as well as continuing to provide a considerable base load electric power generation capability. Extending nuclear energy into the industrial and transportation sectors through the co-production of process heat and electricity provides safe, reliable energy for these sectors in an environmentally responsible manner. The substantial improvement in nuclear plant safety for the protection of the public and the environment afforded by the modular HTGR supports collocation of the HTGR with major industrial facilities. Under U.S. Department of Energy (DOE) direction since 2006, the Next Generation Nuclear Plant (NGNP) project at Idaho National Laboratory (INL) has been working toward commercializing the HTGR technology. In response to a 2011 decision [1] by the Secretary of Energy to reduce the scope of the NGNP project down to a research and development program, however , considerable realignment has taken place. This report: (i) summarizes the accomplishments of the NGNP program from Fiscal Year (FY) 2011 through FY 2013; (ii) lays out the path forward necessary to achieve the ultimate objective of commercializing HTGR technology; and (iii) discusses ongoing technical, licensing, and evaluation activities under the realigned NGNP program considered important to preserve the significant investment made by the government to-date and to maintain some progress in meeting the objectives of the *Energy Policy Act of 2005* (EPAct 2005).

Program Accomplishments

In December 2002, development of the International Generation IV Technology Roadmap concluded that a Very High Temperature Reactor (VHTR) concept based on gas-cooled reactor technology should be one of several nuclear energy systems to be pursued[2]. This concept could provide high-temperature process heat and produce hydrogen for industrial applications. In the September 2003 *U.S. Generation IV Implementation Strategy* report to Congress, the NGNP, based on VHTR gas-cooled reactor technology, was designated the first priority for U.S. development.

Via its November 2004 DOE contract (DEAC07-05ID14517) to manage and operate INL, Battelle Energy Alliance LLC (BEA) was directed to lead U.S. research, development, and exploration of NGNP technologies and carry out this mission in cooperation with other national laboratories, universities, international partners, and the private sector. Subsequently, Congress established the NGNP project through the *Energy Policy Act of 2005*. DOE was directed to form this project to conduct research, develop, design, construct, and operate a prototype nuclear reactor plant. The project was to be conducted in two phases with interim reviews by the Nuclear Energy Advisory Committee.

The NGNP project at INL was formally initiated in March 2006 with the issuance of the *Preliminary Project Management Plan*. Since that time, the project (and now the program) has undertaken (among others) the following selected activities:

- Consolidated all research and development, design, engineering, licensing, quality assurance, and management activities under a single management team.
- Performed a substantial scope of research and development activities for HTGR applications that have:
    1. Demonstrated proof of the tristructural isotropic (TRISO) fuel design concept, developed consistent production quality for said fuel, and demonstration of its performance under operating and postulated accident conditions.
    2. Characterized and now is in the process of qualifying graphite structural materials under irradiated conditions.

3. Extended the high-temperature material characterization data for achieving consensus design code requirements for metals applications.
4. Initiated the analytical modeling qualification for the fuel, graphite, high-temperature materials, and overall reactor and nuclear system behavior.

- Managed and performed preconceptual design activities in 2007 for NGNP/HTGR design concepts by three design teams composed of approximately two dozen companies with interests in development of HTGR technology.[a]
- Performed a broad range of engineering trade studies to evaluate technical areas important to the development and maturation of the NGNP/HTGR concept.
- Developed and initiated the implementation of a technical risk management program that characterized the technology readiness levels of key structures, systems, and components (SSCs) of the HTGR and production of technology development roadmaps for increasing the technical readiness of these SSCs to required levels for deployment.
- Developed new and updated American Society for Testing and Materials (ASTM) International material specifications for nuclear graphite and selected high-temperature metals.
- Developed and implemented a quality assurance program for all NGNP and VHTR activities that is compliant to NQA-1-2008; 1a-2009, "Quality Assurance Requirements for Nuclear Facilities Applications"
- Developed and supported approval and publication of a new American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 5 providing construction rules for high-temperature reactors, including rules for graphite core components.
- Prepared an implementing strategy for licensing HTGR technologies and submitted it to the Nuclear Regulatory Commission (NRC).
- Prepared a series of white papers and responses to Request for Additional Information for policy and high level technical areas important to developing the regulatory requirements in these areas that were submitted to the NRC.
- Evaluated the technical feasibility and economic viability of integrating HTGR technology in a broad range of major industrial processes as an alternative source of energy to the current use of fossil fuels.
- Evaluated several selected proxy sites for technical feasibility and economic viability for integrating HTGR technology with existing industrial processes. The potential licensing of two of the proxy sites by the NRC was evaluated, which included considerations of collocated hazards and other issues important to licensing HTGR technology.
- Upgraded and implemented the quality assurance function to ensure that the information developed within the program could eventually be used directly in submittals and applications to the NRC.
- Provided project management support to DOE.
- Supported formation of an industry consortium (currently known as the NGNP Industry Alliance, Limited), which is composed of energy end-users and off-takers, a major nuclear owner/operator, nuclear system suppliers, and fuel and materials suppliers.

This report describes the culmination of program activities anticipated to occur between early 2012 and late 2013, the effort needed to achieve the project objective of commercializing the HTGR

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a. Plant design work beyond preconceptual design was not performed by the INL NGNP project from April 2009 to the present by direction of DOE.

technology, and the scope of activities in research and development and licensing under the realigned NGNP program to be managed by the INL VHTR Technology Development Office (TDO).

Project Status

On October 17, 2011, the Secretary of Energy forwarded to Congress the report and recommendations of a Nuclear Energy Advisory Committee review of the NGNP project EPAct Phase 1 activities. The Secretary's letter concluded that "…Given current fiscal constraints, competing priorities, projected cost of the prototype, and the inability to reach agreement with industry on cost share, the Department will not proceed with the Phase 2 design activities[b] at this time[1]. The Program will continue to focus on high-temperature reactor research and development activities, interactions with the Nuclear Regulatory Commission to develop a licensing framework, and establishment of a public-private partnership until conditions warrant a change of direction." The scope and schedule of these continuing activities, and the conditions warranting a change of direction have not yet been defined; however, the actual effective scope and schedule are implicitly imposed each year on the Program via the approved appropriations.

The result of the Secretary's letter is that the NGNP project at INL was reconfigured as a research and development program in calendar year 2012 and a considerably reduced scope of work is being managed by the VHTR TDO at INL. The reduced scope supports a limited set of ongoing research and development priorities and continues the pre-application licensing activities built around the series of white papers, associated responses to NRC Request for Additional Information, and the pending NRC policy issue assessment reports. No preliminary design work is being performed, which is consistent with the direction from DOE in April 2009. Such design work is considered necessary to support these licensing activities and to otherwise further the development and deployment of the HTGR technology.

Although the Secretary's October 2011 letter did not provide conditions or a schedule for restarting full NGNP project activities, this report is based on the premise that the INL-managed NGNP program assumes that a resumption of full scope activities for development and deployment of the HTGR technology may occur at some future date. Such resumption could be affected either by the federal government, one or more private industrial entities (which would exploit the accomplishments achieved thus far), or a partnership thereof. This report provides a baseline from which future development and deployment of the HTGR technology can progress. This baseline is derived from results of the considerable development work completed by the NGNP project/program at the time of this writing. This report also includes insights of the NGNP program on the work that is needed to complete technology development, design, and licensing to commercialize the technology. In the meantime, the immediately recommended activities are specifically directed at maximizing the future value gained from the considerable investment in technology development by DOE over the past 8 years and minimizing the startup time to resume a larger scope of development and deployment activities at some future time.

Future Activities to Commercialize High Temperature Gas-Cooled Reactor Technology

The capabilities and safety characteristics of the HTGR have attracted the attention of an ever-increasing number of industries as an option to address ongoing environmental concerns, large price variability, and uncertain local availability associated with traditional fossil fuels used for energy and feedstock. However, the HTGR option will exist only if the necessary investment is made to complete its development and commercialize the technology through initial deployment in industry. Given the regulatory and financial barriers to commercializing advanced reactor technology, this investment has thus far been funded primarily by the federal government in collaboration with private industry which has thus far provided limited complementary investment. The goal of reducing fundamental risks to private

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b. Phase 2 as defined in Section 643, Project Organization, of the EPAct.

investors are those associated with modifying the NRC technical and policy infrastructure to support licensing of HTGRs and ensuring that viable business cases can be built around the economics of HTGR nuclear energy systems, including that which takes into account competition from alternative fuels.

As a result of the Secretary's decision, an alternative business model will need to be developed if industry wants HTGR technology to be available as an option over the longer term. Potential models include:

- Government could complete development and construct a small modular demonstration reactor as a national priority. For example, this national priority could be established to provide electric power for a DOE or private industrial site, along with the production of synthetic transportation fuels for Department of Defense purposes. The facility would demonstrate the viability and licensing of both an advanced reactor concept and a hybrid energy production plant. Based on this demonstration, it is anticipated that industry would commercialize the technology for those applications demonstrating a viable business case.
- A single industrial concern, in partnership and cost sharing with government, could complete development and construction of the NGNP demonstration small modular reactor in a first-of-a-kind commercial application as part of a multiple module plant, serving the diverse energy needs of an industrial customer. Such an endeavor would be pursued, if a viable business case can be demonstrated for this multiple-module plant.
- A consortium of private sector companies representing, for example, utilities, vendors, and customers that would combine their resources and commitment with those of national and/or international entities. These private sector companies could complete development and construction of the demonstration small modular reactor as part of first-of-a-kind commercial application within a multiple module plant. The pursuit of such an endeavor is predicated on the conditions that a viable business case can be demonstrated for this multiple-module plant. The consortium may request limited U.S. and foreign government assistance (e.g., tax credits or loan guarantees).
- One or more private vendors, using the fuel design and materials being qualified by the NGNP program, could complete development and construction of a small modular reactor in a first-of-a-kind commercial application for customers with limited access to both the electrical grid and fossil fuel infrastructure, presuming a viable business case can be demonstrated.

These alternative scenarios or variations thereof can be enabled by the following U.S. Government actions in the near-term:

- Continue to pursue a common ground for a comprehensive public-private partnership or other collaborative arrangement between DOE and industry to implement a substantive scope of development activities aimed at deployment and commercialization of HTGR technology. Substantive scope refers to deliverable activities for which industry can identify an acceptable return on cost share or other form of investment with an appropriate level of risk. This will require demonstrated planning and commitment for out-year funding by the government that would provide the confidence necessary for the private sector to invest in such a partnership.
- Complete the following activities to protect and maximize the value derived from the over $500 million investment made by the U.S. in HTGR technology over the past 8 years:
  - Continue the research and development program to the point of qualifying and codifying the TRISO fuel, graphite structural materials, high-temperature metals, and applicable analytical methods for reactor outlet temperature applications up to 925°C.
  - Prepare technical reports as part of the research and development program, on the fuel, graphite, high-temperature metals, and analytical methods for use by an applicant in developing topical

reports for submittal as part of an application for a combined license and/or design certification from the NRC.

- Support continuing interactions with NRC to ensure the subjects of the extant licensing white papers, responses to the Request for Additional Information, and NRC assessment reports are brought to conclusion and, where needed, formally acted upon by the Commissioners. This will help ensure that the foundation for regulatory requirements and a review process exist for HTGR technology and are formally recognized via topical reports and changes to regulatory infrastructure, as appropriate. This support may require collaboration with one or more industry applicants for HTGR technology Design Certifications and/or combined licenses.

The various sections of this report describe the remaining research and development and licensing activities consistent with these activities. This scope of work is reduced from the original NGNP project planning to be consistent with the Secretary's letter of October 17, 2011, and is the minimum estimated to be required to maintain a minimal level of progress in meeting the objectives of commercializing the HTGR technology. The reduced scope of work is consistent with the reconfiguration of the INL NGNP organization from a project to a research and development program managed by the INL VHTR TDO.

This reduced scope of work is not sufficient to meet the objective of commercializing HTGR technology. This scope is a stopgap measure to maintain some progress and protect prior investment toward the ultimate objective until the means necessary to accomplish this objective have been identified, acquired, and deployed. The main body of this report (Sections 1 through 8) summarizes the accomplishments of the project/program at the time of this writing and identifies the additional needed developments in order to meet the objective.

CONTENTS

FIGURES

TABLE

ACRONYMS

ACRS	Advisory Committee on Reactor Safety
AGC	Advanced Graphite Creep
AGR	Advanced Gas Reactor
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
B&W	Babcock & Wilcox Nuclear Operations Group
BEA	Battelle Energy Alliance
CCNY	City College of New York
CO	Carbon Monoxide
DOE	Department of Energy
FIMA	Fissions per Initial Metal Atom
FY	Fiscal Year
HFEF	Hot Fuel Examination Facility
HTGR	High Temperature Gas-cooled Reactor
HTTF	High Temperature Test Facility
HTTR	High Temperature Test Reactor
INERI	International Nuclear Energy Research Initiative
INL	Idaho National Laboratory
IPyC	Inner PyroCarbon
ISU	Idaho State University
LWP	Laboratory-Wide Procedures
LWR	Light Water Reactor
MHTGR	Modular High Temperature Gas-cooled Reactor
MOOSE	Multiphysics Object Oriented Simulation Environment
MTS	Methyltrichlorosilane
MTHM	Metric Tons Heavy Metal
NDE	Nondestructive Examination
NDMAS	Nuclear Data Management and Analysis System
NEA	Nuclear Energy Agency
NEAMS	Nuclear Energy Advanced Modeling and Simulation
NGNP	Next Generation Nuclear Plant
NQA	Nuclear Quality Assurance

NRC	Nuclear Regulatory Commission
NSTF	Natural Circulation Shutdown Test Facility
OECD	Organization for Economic Co-operation and Development
OPyC	Outer PyroCarbon
ORNL	Oak Ridge National Laboratory
PARFUME	PARticle FUel ModEl
PEBBED	Pebble Bed Burnup and Diffusion
PGS	Precision Gamma Scanner
PIE	Post-Irradiation Examination
QA	Quality Assurance
QAP	Quality Assurance Program
QAPD	Quality Assurance Program Description
R&D	Research and Development
RCCS	Reactor Cavity Cooling System
SiC	Silicon Carbide
SNU	Seoul National University
STEM	Scanning Transmission Electron Microscopy
TAMU	Texas A&M University
TDO	Technology Development Office
TRISO	TRistructural ISOtropic
UCO	Uranium Carbon Monoxide
UO_2	Uranium Dioxide
UP	University of Pittsburgh
USU	Utah State University
V&V	Validation and Verification
VHTR	Very High Temperature Reactor
VHTRC	Very High Temperature Reactor Critical

NGNP Program
2013 Status and Path Forward

Although the original demonstration mission of the Next Generation Nuclear Plant (NGNP) project was abandoned by the Department of Energy (DOE) in 2011 [52], research and development (R&D) activities in many areas continue to varying degrees. As fuel performance is the linchpin of high-temperature gas-cooled reactor (HTGR) safety and is the most complex of the plant systems, fuel development and qualification comprise the bulk of the continuing NGNP effort. R&D in graphite and high-temperature alloy qualification and design and safety methods are also described in the following sections followed by the status of nuclear data management, licensing, quality assurance, and overall project planning and controls.

1. AGR FUEL DEVELOPMENT AND QUALIFICATION

1.1 Fuel Fabrication

1.1.1 Background

Modular HTGR designs were developed to provide natural safety, which prevents core damage under all design basis accidents and presently envisioned severe accidents. The principle that guides their design concepts is to passively maintain core temperatures below fission product release thresholds under all accident scenarios. This level of fuel performance and fission product retention reduces the radioactive source term by many orders of magnitude and allows potential elimination of the need for evacuation and sheltering beyond a small exclusion area. This level, however, is predicated on exceptionally high-quality fuel fabrication quality and performance under normal operation and accident conditions. High-quality fuel for pebble bed HTGRs was developed and demonstrated in Germany in the 1980s, but no U.S. manufactured fuel had exhibited equivalent performance prior to the Advanced Gas Reactor (AGR) Fuel Development and Qualification Program. The design goal of the modular HTGRs is to allow elimination of an exclusion zone and an emergency planning zone outside the plant boundary fence, typically interpreted as being about 400 meters from the reactor. To achieve this, the reactor design concepts require a level of fuel integrity that is better than that claimed for all prior US manufactured tristructural isotropic (TRISO) fuel, by a few orders of magnitude. The improved performance level is about a factor of three better than qualified for German TRISO fuel in the 1980s.

At the start of the AGR program, without a reactor design concept selected, the AGR fuel program selected to qualify fuel to an operating envelope that would bound both pebble bed and prismatic options. This required a fuel form that could survive at peak fuel temperatures of 1250°C on a time-averaged basis and burnup in the range of 150 to 200 GWd/MTHM (metric tons of heavy metal) or 16.4 to 21.8% fissions per initial metal atom (FIMA). Although Germany has demonstrated excellent performance of TRISO-coated uranium dioxide (UO_2) particle fuel up to about 10% FIMA and 1150°C, UO_2 fuel is known to have limitations because of carbon monoxide (CO) formation and kernel migration at the high burnups, power densities, temperatures, and temperature gradients that may be encountered in the prismatic modular HTGRs. With uranium oxycarbide (UCO) fuel, a mixture of (UO_2 and UC_x), the kernel composition is engineered to prevent CO formation and kernel migration, which are key threats to fuel integrity at burnup, temperature, and temperature gradients higher than anticipated for HTGRs designed to run on UO_2. Furthermore, the poor fuel performance of UO_2 TRISO fuel recently observed in irradiated pebbles sponsored by China and Germany along with historic data on poorer fuel performance in safety testing at elevated temperatures and burnup [3] have raised concerns about the suitability of UO_2 TRISO above 10% FIMA and 1150°C. This continues to be an active area of study internationally.

1.1.2 Objectives

The overall objective of the AGR Fuel Development and Qualification Program [4] is to qualify UCO TRISO-coated particle fuel for use in the HTGR. TRISO-coated particles must be fabricated at industrial scale, as opposed to small batches in a laboratory, for use in qualification testing. The testing consists of a variety of experiments and examinations that enable an understanding of the behavior of TRISO-coated fuel under the radiation and temperature environment expected in an HTGR under both normal operation and accident conditions. The program also contains experiments to provide an understanding of how the fission products are retained by or transported through the fuel particle kernel, coatings, and the graphite matrix that comprise the reactor core to support the reactor source term calculations. Another important part of the program is the development of fuel performance and source term modeling and simulation computer tools and the associated physical testing required to validate those tools for use in the HTGR design and safety analysis [5,6,7].

1.1.3 Fabrication

The TRISO-coated particle is a spherical-layered composite, about 1 mm in diameter. It consists of a kernel of UO_2 or UCO surrounded by a porous graphite buffer layer that absorbs radiation damage and allows space for fission gases produced during irradiation. Surrounding the buffer layer is a layer of dense pyrolytic carbon called the inner pyrolytic carbon (IPyC) layer, a silicon carbide (SiC) layer, and a dense outer pyrolytic carbon (OPyC) layer. The pyrolytic carbon layers shrink under irradiation and create compressive forces that act to protect the SiC layer, which is the primary pressure boundary for the microsphere. The IPyC layer also protects the kernel from corrosive gases present during deposition of the SiC layer. The SiC layer provides the primary containment of fission products generated during irradiation and under accident conditions.

The uranium-containing kernels are made by a sol-gel process, followed by washing, drying, calcinations, and sintering to produce UCO kernels. UCO is a mixture of UO_2, UC, and UC2 and is formed by dispersing carbon in the sol-gel bead and performing a carbothermic reduction after calcination. The coatings are applied in a fluidized-bed coater in a sequential, continuous process. The coating process for the buffer is based on chemical vapor deposition from a mixture of argon and acetylene. The inner and outer pyrolytic carbon layers are deposited from a mixture of acetylene, propylene, and argon. The SiC layer is deposited from a mixture of hydrogen and methyltrichlorosilane (MTS) with or without the addition of argon. Graphite powder and organic binders are used to produce a powder matrix that is applied to the TRISO particles as an overcoat. The overcoated particles are then pressed to form the cylindrical compact. The compacts undergo carbonization and heat treatment at high temperature to produce the final fuel form.

Rigorous control is applied at every step of the fabrication process to produce high quality, very-low-defect fuel. Specifications are placed on the diameters, thicknesses, and densities of the kernel layers; the sphericity of the kernel and coated particle; the stoichiometry of the kernel; the maximum anisotropy of the pyrocarbon layers; the microstructure of the SiC; and the acceptable defect levels for each layer. Statistical sampling techniques are used to demonstrate compliance with the specifications, usually at the 95% confidence level.

At its inception, the AGR Fuel Development and Qualification Program had to reestablish the capability to fabricate and characterize TRISO-coated particle fuel in the U.S. after about a decade long hiatus. Many of the characterization procedures and associated equipment used in the past were still available but needed to be modernized to take advantage of current measurement technology. New procedures and personnel had to be qualified to meet NQA-1-2008; 1a-2009 requirements. In some cases, new, more accurate, and repeatable methods were developed. Specific changes include:

- Removal of high variability human interactions in the process by (a) replacing tabling with 3D sieving of coated particles; (b) replacing multi-step matrix production (resin solvation, matrix mixing, kneading, drying and crushing) with dry mixing and jet milling of the matrix material; (c) replacing the rotary overcoater with an automated fluidized bed overcoater that produces highly spherical, uniformly overcoated particles; and (d) using automated die filling and punch travel to form compacts

- In the fabrication of kernels, internal gelation was used to improve sphericity, compared to international efforts that use external gelation, and the method of carbon addition modified to improve distribution of oxide and carbide phases
- Improvements were made in chemical vapor deposition process control, including (a) argon dilution during SiC coating, (b) development of a coater "chalice" and multiport nozzle to improve yields (>95%), (c) using mass flow controllers to control gas flows during deposition of each coating layer, and (d) implementing an improved MTS vaporizer (leveraging computer chip industry) to evaporate MTS for the deposition of SiC.

In the area of measurement science, computer measurements of thicknesses greatly improved anisotropy measurements and improved density measurements using better density column materials.

The result has been more controlled and reproducible fabrication and much more accurate and precise characterization of this fuel form. Better control of the processes, removal of high variability human interactions in the process and better measurement technologies all contribute to better quality TRISO fuel. [3,4,5] Figure 1 illustrates the standard deviation in performance of historical German, Japanese, and US fuels along with recent results from the AGR program. In all cases, the standard deviation of the coating layers is as good as, or better than, the historical data indicating tighter process control associated with chemical vapor deposition and enhanced characterization techniques that provide greater precision to the measurements.

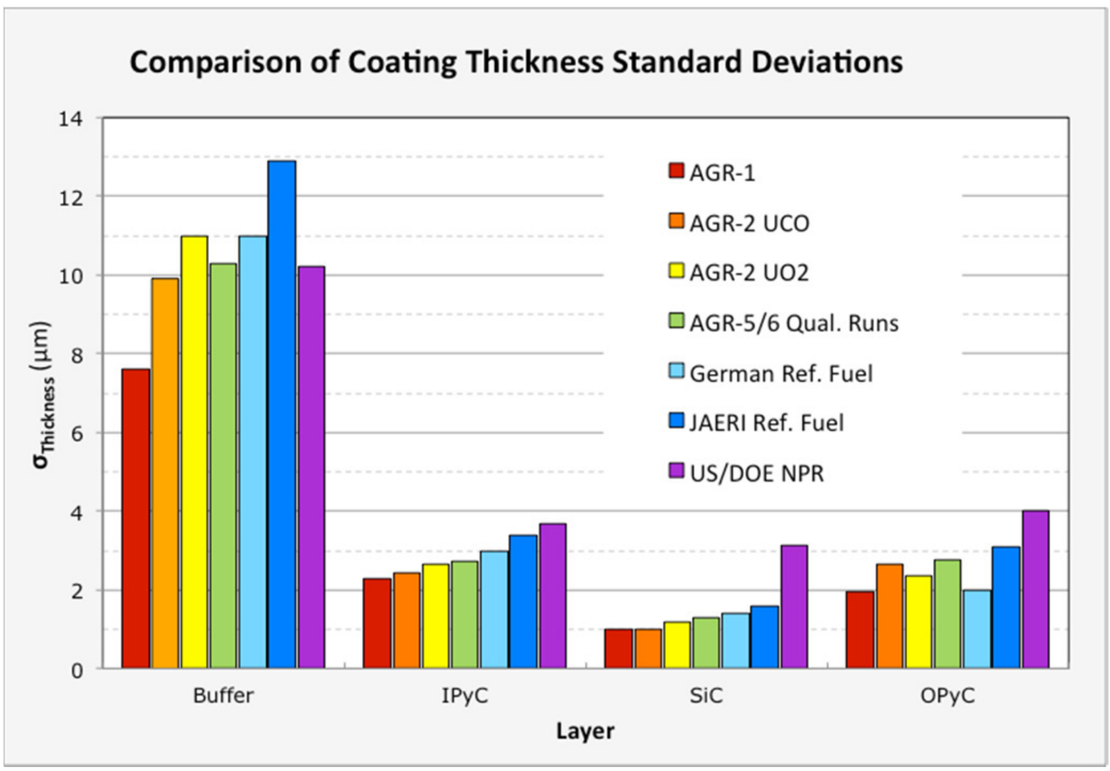

Figure 1. Comparison of standard deviation of coating layers from AGR program with historic US, German, and Japanese TRISO fuel.

Systematic fabrication studies, combined with improved characterization capabilities, have also enhanced the understanding of how to fabricate high quality TRISO fuel. The program is now fabricating high quality TRISO-coated fuel particles in an engineering-scale coater that exhibit a historically low rate of less than five defects in every 100,000 particles because of flawed coatings (SiC or inner PyC) or exposed uranium). Placing a U.S. fuel vendor in position to fabricate high quality TRISO fuel with an improved fundamental understanding of the relationships between the fuel fabrication process, fuel properties, and fuel performance enhances credibility in establishing the safety case, and in establishing the licensing basis with the U.S. Nuclear Regulatory Commission (NRC). Today, the US fuel vendor (Babcock Wilcox Nuclear Operations Group [B&W]) has all the technologies necessary to fabricate TRISO coated UO_2 or UCO fuel in compact form. A pilot line has been established and fuel for final fuel qualification testing has begun.

1.2 Performance under Irradiation and Accident Safety Testing

The fuel irradiation activities will provide data on TRISO-coated fuel performance under normal operation. The objective of the post-irradiation examination (PIE) and safety testing is to characterize and measure the performance of TRISO fuel during irradiation and under accident (elevated temperature) conditions. Under normal operating conditions, the particles attain a temperature of up to 1200°C. In the event of a loss of forced cooling; however, decay heat and the lack of active cooling will increase peak fuel temperatures by hundreds of degrees. PIE and safety testing support the fuel development effort by providing feedback on the performance of kernels, coatings, and compacts. Data from PIE and accident testing will supplement the in-reactor measurements as necessary to demonstrate compliance with fuel performance requirements and to support the development and validation of computer codes.

1.2.1 AGR-1

The first irradiation test, AGR-1, ended in 2009 after approximately 3 years of irradiation. The UCO fuel in AGR-1 was irradiated to a peak burnup of 19% FIMA, a peak fast-neutron fluence of about 4.5 × 1025 n/m^2, and a maximum time-averaged fuel temperature of about 1250°C. About 300,000 TRISO fuel particles were irradiated without a single particle failure, as indicated by the fission-gas measurements on the purge gas from each of the capsules. [8] Fission gas release measured from both the AGR-1 and AGR-2 experiments is compared to historic German and US irradiations in Figure 2.

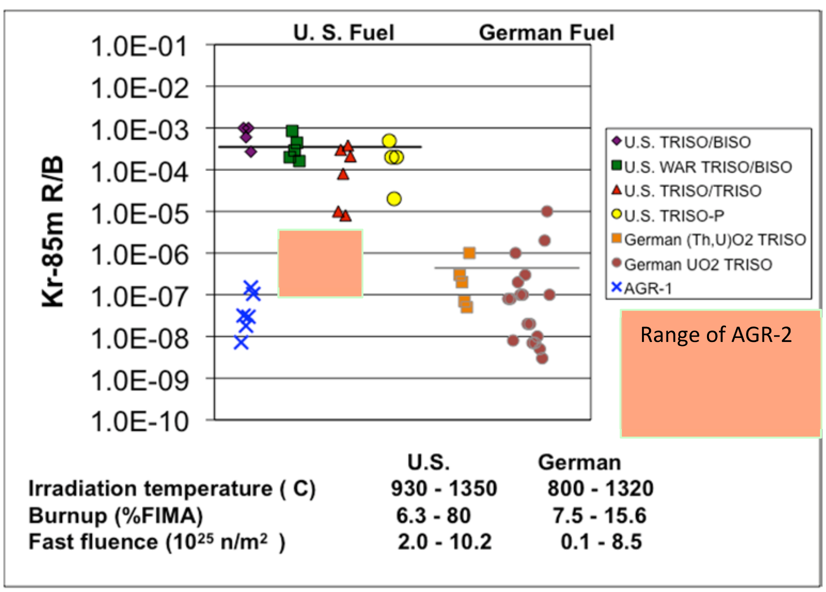

Figure 2. End-of-life Kr-85m fission gas release for AGR-1 and AGR-2 compared to historic performance in US and German TRISO fuel irradiations.

This is the best irradiation performance of a large quantity of TRISO fuel ever achieved in the US, and exceeded the German levels of burnup by almost a factor of 2. This irradiation has confirmed the expected superior irradiation performance of UCO at high burnup. No kernel migration, no evidence of CO attack of SiC, and no indication of SiC attack by lanthanides has been observed.

Zero fuel failures among the 300,000 AGR-1 particles translates into a 95% confidence failure fraction is <1E-5, a factor of 20 better than the design in-service failure fraction requirement of 2E-4. The more severe AGR-1 irradiation conditions compared to the vast majority of historic modular HTGR designs suggest substantial fuel performance margin.

PIE of UCO TRISO fuel irradiated in AGR-1 is complete. The very low release of key metallic fission products (except silver[c]) measured in PIE is confirming the excellent performance measured under irradiation [9,10]. Accident safety testing of UCO TRISO from AGR-1 is nearing completion and is demonstrating expected robustness of the fuel. An example of fission product release from 1600°C safety testing is shown in Figure 3. Very low releases have been measured in post irradiation accident simulation heatup testing ("safety testing") after hundreds of hours at 1600 and 1700°C and no through-particle coating failures (no noble gas release measured) have been observed [11]. No full particle failures observed in testing to date corresponds to a 95% failure fraction of ~ 8E-05, a factor of 7.5 margin relative to the HTGR prismatic reactor specification.

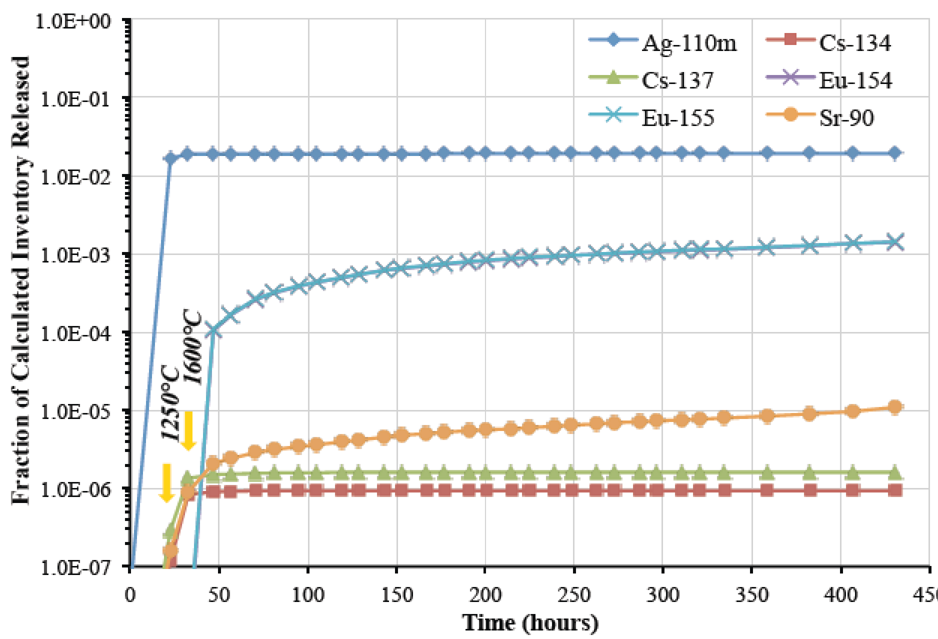

Figure 3. Fission product release from heating of AGR-1 compact 6-4-3 at 1600°C.

1.2.2 AGR-2

The second irradiation, AGR-2, is complete. It contains separate compartments to test UCO and UO_2 TRISO produced at industrial scale from U.S. and international collaborators (France/AREVA and South Africa/PBMR). The UCO has been irradiated under prototypical prismatic core conditions while the UO_2 TRISO temperature experienced conditions typical of a pebble-bed HTGR. The in-pile results to date are very good as shown in Figure 2 [12]. PIE and safety testing are scheduled for 2014 through 2016.

c. The migration of silver is expected due to diffusion through the coating layers and is not an indication of coating defects or failure. The extent of silver migration increases with fuel temperature, is primarily a potential issue for plant maintenance, and is not an important contributor to the postaccident source term.

Collectively, these irradiation and accident safety results provide a high level of confidence that the AGR fuel program will successfully demonstrate the superior performance capability of TRISO fuel required by the modular HTGR concept. Additional safety testing of industrially produced TRISO fuel is planned in 2014.

1.2.3 AGR-3/4 and AGR-5/6/7

The third irradiation test, AGR 3/4 designed to study fission product release from intentionally failed fuel and retention in graphitic components, will complete in April of 2014. The design of the fuel qualification irradiation test capsule (AGR-5/6/7) is under way [13].

1.3 AGR Fuel Development and Qualification Future Work/Path Forward

The AGR program has been extremely successful to date. The three key programmatic goals of the program are: (1) stand up a domestic vendor capable of fabricating HTGR TRISO fuel, (2) qualify the TRISO fuel, and (3) qualify the source term used in HTGR safety analysis. The key remaining activities that are necessary to meet these programmatic goals are:

Development of US vendor (95% complete as of this writing).

- Complete fabrication of qualification test fuel.

Qualify TRISO fuel (70% complete as of this writing).

- Complete AGR-1 safety testing and PIE to confirm robustness of TRISO fuel under accident conditions.
- Complete AGR-2 safety testing and PIE.
- Complete development of a furnace to evaluate moisture and air ingress effects on fuel under accident conditions.
- Complete irradiation and accident safety testing in the AGR 5/6/7 campaign (including margin testing in AGR-7).

Qualify Source Term (25% complete as of this writing).

- Complete AGR-3/4 irradiation, safety testing, and PIE to demonstrate fission product retentiveness of graphite and fuel matrix materials and update database on fission product transport rates in HTGRs to support source term evaluations.
- Complete fission product validation irradiation and safety testing in AGR-8.
- Complete fission product plateout and liftoff studies to support source term evaluations in the primary system and reactor building.
- Complete tritium permeation testing for potential intermediate heat exchanger alloy systems.

Beyond the AGR program, once the preliminary design activities are nearing completion, there is a need to reevaluate the fuel acquisition strategy and the specific test conditions being used in the program to ensure they meet the needs of the design and the deployment strategy for the project.

2. GRAPHITE DEVELOPMENT AND QUALIFICATION

The objective of the graphite program [14] is to develop the qualification data set of thermomechanical and thermophysical properties for unirradiated and irradiated candidate grades of graphite for HTGRs. Five major graphite grades, suitable for use within both pebble bed and prismatic HTGR designs, have been selected for further evaluation. These include NBG-18 and NBG-17 (SGL in Europe), PCEA (GrafTech Inc. in the U.S.), IG-110 (Toyo Tanso in Japan), and 2114 (Mersen, formerly known as Carbonne Lorraine, in the U.S.). Historical samples and minor grades, such as PGX, HLM, PCIB, and H451, have also been incorporated into the program to help further elucidate the impact of fabrication processes and coke sources on the resulting microstructure of the graphite and its performance under irradiation. The planned activities will demonstrate the performance of various graphite types under bounding conditions, including irradiation dose levels, anticipated applied stress levels, and maximum core temperatures.

The graphite program consists of statistical characterization of unirradiated graphite material properties to establish the lot-to-lot, billet-to-billet, and within-billet variability of the material. This characterization will establish a quantitative baseline of material properties from which changes under irradiation can be understood. Significant effort has gone into establishing the analytical measurement laboratories required to perform the extensive characterization of nuclear graphite under consideration for HTGRs being developed by the NGNP project. An extensive characterization effort is currently under way at INL and ORNL to establish the material properties before irradiation on a series of large graphite billets for the five major grades selected.

As of this writing, the baseline statistical characterization of the thermomechanical and thermophysical properties of five billets of two major graphite grades is complete, and billets from each of the three remaining grades are currently being characterized. Characterization of over 2,000 samples for the Advanced Graphite Creep (AGC) test series AGC-1, AGC-2, AGC-3, and AGC-4 prior to irradiation is also complete. Test series capsules AGC-1 and AGC-2 have completed irradiation. AGC-3 capsule is currently under irradiation in the ATR. Design and construction of AGC-4 test series capsule is anticipated to be completed in early FY 2015. PIE of the AGC-1 graphite specimens has been completed, and AGC-2 is being disassembled to remove specimens from the test series capsule. AGC-2 PIE activities will begin directly after capsule disassembly has been completed. Other important activities are as follows:

- Currently conducting extensive studies on graphite-air and graphite-water oxidation to better understand mechanisms of oxidation as a function of temperature, microstructure, and air/water concentration. The underlying mechanisms must be understood to predict material performance for chronic oxidation rates, which might aggravate acute accident safety evaluations. Limited studies on acute oxidation rates of irradiated graphite were conducted to determine effects of irradiation on oxidation rate of selected nuclear grades.
- As a result of lessons learned in the AGR Fuels PIE activities, initial oxidation studies of boron-doped graphite have been conducted. Initial results indicate at least a 5-fold decrease in oxidation rates of graphite due to limited boron doping. More extensive studies are anticipated to ascertain whether this boron-doped material could be used as oxidation-resistant graphite in nuclear applications.
- Continuing to evaluate advanced failure models for graphite based on measurements of graphite in complex combinations of potential multiaxial stress states. Potential K_{IIc} (Mode II, critical stress intensity factor) testing standards have been explored and are being evaluated for ongoing fracture studies in the compression-tension stress state domain. Once a viable test standard or procedure has been established, a new American Society for Testing and Materials (ASTM) K_{IIc} test standard will be written for future testing.

- The new graphite core component design and construction code was approved by ASME on November 1, 2011, as part of the new ASME Boiler and Pressure Vessel Code Section III, Division 5. Parts of the code, specifically the irradiation material property changes, are still being developed. Two outstanding issues have recently been resolved including: (1) establishment of nuclear graphite acceptance criteria and (2) approval of graphite irradiation data in ASME Section III. Other remaining issues will be addressed in the future as data become available.
- Completed a new ASTM Guideline for nondestructive examination (NDE) methods employed on nuclear graphite components. The guideline summarizes the currently available techniques with the maturity needed for component inspection, implementation of the various techniques, challenges, and issues for interrogating large graphite components for each method, and assumptions underlying each technique. No new NDE techniques are addressed in the guideline; only mature techniques are addressed. Before an ASTM standard can be developed, universal graphite components must be established as the thickness and graphite grade can significantly alter the analysis from NDE techniques. This is also applicable for the development of new in situ-inspection techniques. These techniques will then be used as ASME inspection criteria within the Graphite Code.
- Initiated fundamental studies through the Nuclear Energy University Program and international collaborations such as the International Atomic Energy Agency to better understand the damage mechanisms and behavior of graphite under irradiation. Specific areas of investigation include:
    - Analysis of microstructural features key to nuclear graphite grade irradiation behavior
    - Irradiation damage mechanisms controlling irradiation property changes
    - Understanding the underlying mechanisms for irradiation induced creep
    - Understanding fundamental graphite oxidation mechanisms
    - Detecting and predicting defect and microstructural evolution (under irradiation, stress, and temperature).

  These are the fundamental mechanisms responsible for the changes to the graphite material properties and will dictate the graphite performance while in service as a core component material. Preliminary results are promising. Initial results have yielded detailed microstructural features in both filler particle and binder matrix phases. Nondestructive technique (X-ray Computer Tomography) has revealed certain pore microstructure changes and initial results on material property recovery indicate irradiation defect structures. Extensive collaborations with a sophisticated atomistic model have led to a better understanding of defect evolution and potential irradiation creep response. These early results provide an improved understanding of these fundamental mechanisms required for predictive performance model behavior in graphite oxidation, irradiation creep rates, and internal stress build-up within critical core components, thermal conductivity, fracture mechanics, and whole core behavior and performance during service. Continuing collaborative work on fundamental studies is required for complete understanding of these performance controlling mechanisms.

  The key remaining activities necessary to qualify graphite for use in NGNP are as follows:

- Complete virgin material property (baseline) testing program to provide a statistically representative, as-fabricated material property database for comparison to the irradiated material property changes
- Complete post-irradiation characterization of low-temperature graphite specimens from AGC-2 capsules
- Complete 900°C temperature irradiations (AGC-3 and AGC-4 capsules) and associated post-irradiation characterization
- Complete 1200°C temperature irradiations (AGC-5 and AGC-6 capsules) and associated post-irradiation characterization

- Complete chronic oxidation studies for graphite
- Complete initial studies of boron-doped graphite to determine if it may be a viable alternative grade for nuclear applications
- Continue to analyze the data from the baseline and irradiated material property testing and apply these data to the analytical models and collaborations with other research institutions to improve our fundamental knowledge of graphite behavior under irradiation, stress, and temperature
- Complete modeling activities to improve material behavior predictions of nuclear graphite components
- Address remaining ASME Nuclear Graphite Code issues: (1) assessment of the proposed strength/failure prediction methodology, (2) development of seismic and fatigue sections without design specific conditions, (3) development of NDE acceptance criteria once ASTM standard guidelines have been established for using NDE methods on nuclear graphite components, (4) model benchmarking verification, and (5) development of a document for subsequent additions to the code.

3. HIGH-TEMPERATURE MATERIALS QUALIFICATION

The NGNP Materials R&D Program [15] provides the essential materials R&D needed to support the design and licensing of the reactor and balance of plant, excluding the hydrogen plant. The thermal, environmental, and service life conditions of the NGNP will make selection and qualification of some high-temperature materials a significant challenge; thus new materials and approaches may be required.

3.1 High-Temperature Materials Qualification Background

Design for components that comprise the pressure boundary for nuclear systems must comply with the ASME Boiler and Pressure Vessel Code rules. For austenitic alloys, Section III of the ASME Code is divided into Subsection NB, which is limited in temperature to 427°C (800°F) for elastic design and Subsection NH, which applies above that temperature and allows limited amounts of inelastic deformation to be considered in the design. The high outlet temperature of the HTGR (particularly the VHTR) heat exchanger will require code qualification of metallic materials for temperatures and service times greater than any alloy that is currently qualified for nuclear design.

Based on technical maturity, availability of required product forms, and mechanical properties at elevated temperature; the nickel-based, solid solution Alloy 617 was down-selected from a family of similar alloys by the VHTR program as the primary material of choice for code qualification for the heat exchanger application. The iron-nickel-chromium solid solution alloy 800H is a suitable candidate for a VHTR steam generator operating up to 750°C. At the inception of the VHTR program, Alloy 617 was not in either subsection of the nuclear design code, while Alloy 800H was allowed for nuclear design, but for limited time and to a temperature of 760°C.

In addition to qualification of suitable materials for nuclear design, the design rules that are incorporated in the ASME code were last modified in a significant way to describe the behavior of austenitic stainless steels for fast breeder reactors. These rules assume, for example, that the creep behavior is characterized by an extensive region of steady-state creep rate, followed by a tertiary creep regime that is dominated by cavitation and rapid failure. Alloy 617 and 800H do not exhibit steady-state creep. The majority of creep life is spent in a deformation regime exhibiting increasing creep rate with increasing time (or strain), and the onset of apparent tertiary creep does not typically correspond to extensive formation of creep cavitation. More accurate design rules for multi-axial stress states and to properly treat creep-fatigue deformation are also required for VHTR design.

3.2 High-Temperature Materials Qualification Objectives

The main focus of the VHTR high-temperature materials program [16] is to develop sufficient data and analysis of the behavior of Alloy 617 to code qualify the material for use in the primary nuclear pressure boundary up to 950°C and to extend the existing code qualification for Alloy 800H up to 850°C and 500,000 hours. An important parallel activity has been to modify the high-temperature design rules that are currently in Subsection NH of the code to properly take into account the behavior of Alloy 617 and 800H, particularly with respect to creep deformation. Simplified design methods for components that will undergo inelastic deformation during service and experimental validation of those methods are also required.

3.3 Material Characterization

Extension of the Alloy 800H code qualification to 850°C and 500,000 hours was proposed based on an extensive effort to acquire and critically assess historical Alloy 800H mechanical property data, along with limited additional testing [17,18]. Testing established that the allowable stresses incorporated in the existing code qualification contained several errors that needed to be corrected and that the temperature limit above which strain-rate sensitivity becomes significant is 650°C. Other than these relatively limited changes, it was determined that the existing data fully supported extending the code allowable time and temperature.

A model describing the effect of impurities in the high-temperature helium environment that are anticipated to exist in a VHTR has been developed based on a modified Ellingham diagram. The model has been validated by extensive testing as part of the VHTR program and through an International Nuclear Energy Research Initiative (INERI) with Commissariat à l'Énergie Atomique (CEA)-Saclay. It was demonstrated that for VHTR temperatures, there will always be some interaction between the VHTR helium and Alloy 617. The environmental interaction is minimized by formation of a thin, stable, and adherent Cr_2O_3 scale that limits either decarburization (and resulting loss of strength) or carburization (and embrittlement). The influence of environment on properties, particularly creep-fatigue behavior, has also been characterized [19,20,21,22,23,24].

Creep-fatigue of both solution annealed plate and weldments of Alloy 617 has been characterized at temperatures of 850 and 950°C. It was determined that in base metal at the lower temperature, failure is dominated by fatigue crack propagation from the specimen surface, while at the higher temperature, failure is a result primarily of linking of internal creep cavities [25,26]. These observations are consistent with an observed transition in dominant deformation mechanism from fatigue at 850°C to solute drag-controlled creep at 950°C [27]. The creep-fatigue life of weldments is significantly reduced compared to the base metal. And at both temperatures, the fracture is along the dendrite boundaries in the weld metal.

An extensive database of Alloy 617 properties was used to code-qualify the alloy for design of non-nuclear pressure vessels. Under the program, the quality of those data has been assessed and testing has been conducted on the impact of extended aging at temperatures thought to represent the most potential for degradation to the mechanical properties [28,29,30]. In general the mechanical and physical properties used for the nonnuclear design from the historical database compare well with the values determined from contemporary heats of Alloy 617. It was found that there is a temperature dependence of the Poisson's ratio that is not accurately captured in the current code values.

Development of constitutive models for Alloy 617 behavior over a wide range of temperatures, strain rates, and loading conditions is ongoing. Experimental validation of the models is also under way. The influence of bi-axial loading on stress rupture life compared to typical laboratory tests of uni-axial specimens is being characterized using pressurized tube creep specimens. Deformation resulting from elastic follow-up is also being investigated using both finite element simulations and two bar ratcheting tests.

3.4 High Temperature Materials Qualification Future Work/Path Forward

Extension of the Alloy 800H code qualification to higher temperatures and longer time is essentially complete. Some additional long-term testing of Alloy 617 to develop a complete experimental database to support code qualification remains to be completed. A fatigue design curve is currently being developed; upon completion of the design curve in FY 2014, sufficient information will be available to proceed with qualification for Section III, Subsection NB. Additional creep and creep-fatigue testing of Alloy 617 base metal and weldments will be completed in FY 2015. Characterization of the influence of long-term aging of Alloy 617 base metal and weldments on the tensile properties and fracture toughness will be completed in FY 2015. Code qualification also requires that allowable stresses for design be determined from experimental data, and isochronous stress-strain curves for a number of temperatures must be developed from models and experimentally validated. These activities will be completed in FY 2015. A special task group has been established within Section II of the code to support submission of an Alloy 617 Code Case in FY 2015. Elevated temperature design rules and associated multi-axial experimental validation will be completed as results become available in FY 2014 and FY 2015. It is anticipated that modification of the draft Alloy 617 Code Case and response to questions from the approving committees will continue in FY 2016 and FY 2017, with the goal of final approval in FY 2017.

4. DESIGN AND SAFETY METHODS

4.1 Design and Safety Methods Background

The NGNP Design Methods and Validation R&D program focuses on the development and validation of tools to assess the neutronic and thermal fluid behavior of the NGNP design [31]. The complete development, verification, and validation of thermal, neutronic, and fluid codes cannot be performed without a parallel experimental program that supplies these new tools with data that envelop the anticipated NGNP designs. Some verification can be achieved by comparing results to those from other, qualified codes if there is sufficient overlap among the calculational envelopes. Validation data may also be obtained from a conservatively operated demonstration reactor.

Following the selection of the prismatic HTGR design as the preferred NGNP design in FY 2012, the methods program was scaled down to focus on prismatic HTGR benchmarking and experimental validation. A limited number of pebble bed tasks were completed as well.

Accurate benchmarking and simulation of prismatic high-temperature reactors for burnup and transient analysis, including the rigorous treatment of uncertainties propagated through all phases of core simulation, remains a challenge. The treatment of both neutron scattering in graphite and resonance capture are complex and not adequately captured using the methods traditionally used in either HTGRs and in light water reactor (LWRs). For burnup calculations, a second level of heterogeneity must be resolved to capture the local effects of burnable poisons and fuel compacts while accurately propagating their effects on a significantly larger spatial scale than the traditional LWR approach. Temperature feedback in HTGRs is dependent on the geometry and composition of the TRISO fuel form, but limits in computation allow full core models to resolve phenomena at this scale only through careful averaging over space and energy. To achieve reasonable benchmark results, different approaches have been developed and implemented in the core simulation tools available at INL and elsewhere.

The goals of the NGNP Experimental Validation and Verification (V&V) R&D are focused on the definition, specification, and implementation of the steps necessary to ensure the software analysis tools that will be used to analyze the behavior of the NGNP-selected VHTR are capable of calculating the plant behavior for steady-state operations and the challenging abnormal operation and accident scenarios for which the plant must be licensed.

The definition, specification, and implementation of the steps necessary to achieve the above goals will be accomplished in the framework described in the Evaluation Model Concepts section of the NRC Regulatory Guideline 1.203 to ensure the software analysis tools, when deemed adequate to perform the required work, are qualified using practices and procedures acceptable to the U.S. NRC.

4.2 Design and Safety Methods Objectives

The two primary objectives of the NGNP Design Methods and Validation R&D program are the development of experimental datasets that envelop the anticipated prismatic HTGR operational regime, and the development V&V of assessment codes and simulation models for generic HTGR designs and computational benchmarks.

4.3 Experimental Validation Activities

To achieve the desired goals, the NGNP Experimental V&V must:

- Define the VHTR scenarios required for VHTR licensing approval
- Identify the key phenomena and figures-of-merit of importance in the scenarios of interest
- Define the experimental and validation data matrices to provide the basis for performing the validation calculations

- Determine whether adequate experimental data are available to complete the validation matrices
- Specify and design the experiments needed to complete the validation matrices using acceptable scaling practices
- Define the metrics needed to perform the validation calculations
- Perform the adequacy calculations and adequacy evaluations using acceptable validation practices and procedures
- Report the capabilities of the analysis tools to meet the NGNP calculational obligations.

The experimental V&V R&D effort has focused on achieving the goals and objectives listed in the above paragraphs. To date, the NGNP scenarios required for analysis have been identified [32], the key phenomena and figures-of-merit have been documented, and the validation matrix has been formulated. The validation matrix, shown in Figure 4, is composed of either already recorded data or data to be recorded via experiments designed using a rigorous scaling technique first documented by Zuber [33], a hierarchical two-tiered scaling approach. The practices and procedures used to construct the NGNP experimental V&V validation matrix are outlined in Schultz [34], and the body of experiments consists not only of experiments performed at INL but also experiments funded through the Nuclear Energy University Program at various universities.

The centerpiece of the experimental V&V R&D is the integral-effects facility under construction at Oregon State University: the High Temperature Test Facility (HTTF). As of December 2013, final assembly of the facility is nearing completion. Acceptance and shakedown testing are scheduled for February 15, 2014, followed by actual experiments in March 2014. Ten experiments are planned in FY 2014. Refurbishment of an ex-vessel experiment (the Natural Circulation Shutdown Test Facility [NSTF]) at Argonne National Laboratory will be completed in 2014 and will provide data for validation of codes and models that simulate the operation of the Reactor Cavity Cooling System (RCCS).

Separate-effects experiments in various stages of construction or test are ongoing at City College of New York, Idaho State University (ISU), Oregon State University, Texas A&M University, University of Idaho, University of Pittsburgh, University of Wisconsin, and Utah State University via Nuclear Energy University Program funding. In addition, experimental data are also obtained via INERI programs with Korea (Seoul National University (SNU), Korea Advanced Institute of Science and Technology, and Korea Atomic Energy Research Institute).

The validation matrix, shown in Figure 4, correlates the required phenomena versus the scenarios of interest, the required phenomena versus the available experimental facilities, and the available experimental facilities versus the scenarios of interest. The experimental contributions, which are either being contributed or will be contributed by university programs, are identified by:

- City College of New York (CCNY)
- Oregon State University HTTF (HTTF)
- Idaho State University (ISU)
- Seoul National University (SNU)
- Texas A&M University (TAMU)
- University of Pittsburgh (UP)
- Utah State University (USU).

When data from one of the listed facilities can be used to directly validate a key phenomenon, the data are identified with a "+." When the data are partially applicable, then the data are identified with a

“O,” while if the data are not applicable to a particular phenomenon, they are identified with a “-.” Data that are presently being planned in the future are identified with a “P.”

The validation data identified in Figure 4 will be used to quantify the adequacy of the NGNP thermal-hydraulic analysis tools. The techniques used to quantify the software tool adequacy are outlined in the description of the experiment plan [35]. The documented approach will be supplemented with more advanced techniques that stem from validation R&D completed recently in the NRC’s effort to validate the TRACE systems analysis code [36].

		Scenario of Interest				Experimental Facility										
		Depressurized Conduction Cooldown	Pressurized Conduction Cooldown	Air Ingress	Normal Operation	INL MIR core bypass	ISU heated core bypass	SNU core bypass	TAMU core bypass	CCNY core heat transfer	TAMU plenum-to-plenum NC	UP LP mixing	INL MIR LP mixing	USU transient mixed convection	INL air ingress	TAMU air ingress
Phenomena	Natural circulation	+	+	+	-	-	-	-	-	P	P	-	-	P	+	+
	Lower plenum mixing	+	+	+	+	-	-	-	-	P	-	P	+	-	+	+
	Upper plenum mixing	+	+	+	+	-	-	-	-	-	P	-	-	-	-	-
	Jet impingement: upper plenum	+	+	+	+	-	-	-	-	-	P	-	-	-	-	-
	Jet impingement: lower plenum	+	+	+	+	-	-	-	-	-	-	P	+	-	-	-
	Core bypass	+	+	+	+	+	P	+	+	O	-	-	-	O	-	-
	Core heat transfer	+	+	O	+	-	P	-	-	P	-	-	-	O	-	-
Test Facilities	INL MIR core bypass	-	-	-	+											
	ISU heated core bypass	-	-	-	+											
	TAMU core bypass	-	-	-	P											
	SNU core bypass	-	-	-	+											
	CCNY core heat transfer	P	P	-	P											
	TAMU plenum-to-plenum NC	P	P	-	-											
	UP mixing in LP	-	-	-	P											
	INL MIR mixing in LP	-	-	-	P											
	USU transient mixed convection	P	P	-	P											
	INL air ingress	-	-	+	-											
	TAMU air ingress	-	-	+	-											
	HTTF integral & separate effects	P	P	P	-											

Figure 4. NGNP thermal-fluids V&V matrix for software validation.

DOE is also negotiating a scope of work for collaborative research and development in high-temperature reactor technology with the Japan Atomic Energy Agency under a Bilateral Agreement. If successful, the INL and Japan Atomic Energy Agency will design and execute coordinated experiments using the Oregon State HTTF and the High Temperature Test Reactor (HTTR) in Oarai, Japan. The HTTR is the only operating prismatic core high-temperature reactor in the world and thus can generate valuable data for the validation of multi-physics and system analysis codes.

4.4 Core Simulation Development Activities

Many of the physics of prismatic and pebble bed reactors are similar, and some tools can be applied to both. However, the prismatic core is generally more complex and less amenable to the 1-D transport approach successfully applied to pebble bed reactor simulation using the PEBBED (Pebble Bed Burnup and Depletion) code [37]. Cross sections for prismatic core simulation (transient and burnup analyses) need to be generated for each fuel and reflector block with great care and resolution, while diffusion and transport equations must be solved in 3-D hexagonal geometry.

Progress had been made on the development of a nodal solution to the diffusion equation, but this line of pursuit was suspended in 2011 in favor of other advanced tools and approaches already under development at INL. Specifically, an HTGR simulation code called PRONGHORN is being constructed on the MOOSE (Multiphysics Object Oriented Simulation Environment) platform to allow high fidelity finite element-based 3-D core simulations using hundreds or thousands of processors [38,39].

One of the focus areas in FY 2013 was on the development and testing of sophisticated lattice physics calculations to generate appropriate few-group cross sections that can be used with confidence in full-core burnup and transient simulations such as those specified in the international benchmarks discussed below. Accurate burnup calculations in prismatic fuel require transport calculations that account for inter-block leakage and a correct representation of burnable poisons. Traditional methods rely on assumptions about these phenomena that are of questionable validity and unknown accuracy. Modern lattice and Monte Carlo tools were applied to the problem in a series of comparison and sensitivity studies that yielded an approach to cross section generation that can be used with confidence. A series of supercell depletion studies were conducted to delineate an analysis approach for the prismatic modular reactor design [40]. These studies show that the traditional two-step method of analysis is not accurate enough to represent the neutronic effects present in the prismatic high-temperature reactor concept. The long-range spectral effects induced by the presence of reflectors, burnable poisons, and operational control rods have been shown to be significant because of the long migration lengths of the neutrons in graphite-moderated reactors. A larger validation study was performed in FY 2013 on the VHTR Critical facility experimental dataset [41]. In this study, lattice cross-section generation and depletion results are compared for single block, super cell, and core geometries using stochastic (Monte Carlo and SERPENT) and deterministic codes (HELIOS, DRAGON, PRONGHORN). The VHTR critical facility was constructed and operated in Japan in support of the design of the HTTR. It featured a simpler geometry and operated at low temperatures to facilitate accurate validation of HTGR physics codes (Figure 5).

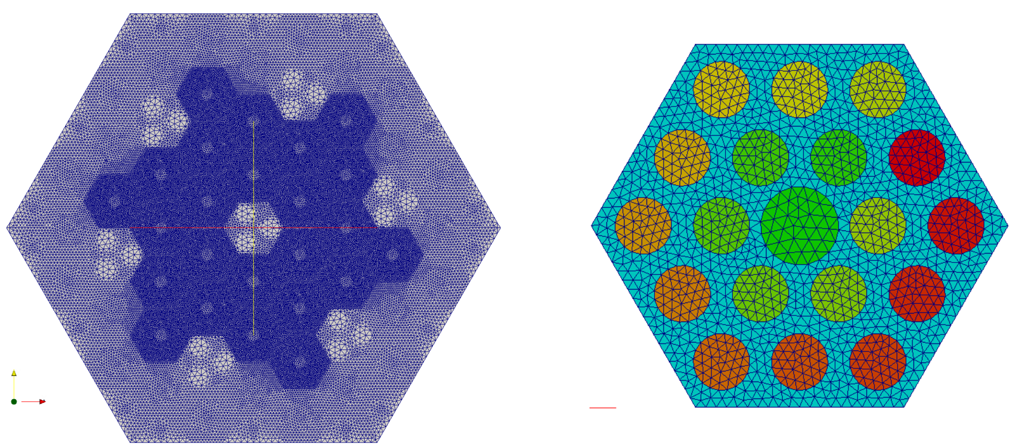

Figure 5. Very High Temperature Reactor Critical (VHTRC) pin-resolved mesh (left) and a single fuel block (right) generated from this mesh.

NGNP Core Simulation staff led the construction of the Organization for Economic Cooperation and Development (OECD) Modular High Temperature Gas-cooled Reactor (MHTGR)-350 Transient Benchmark for comparing and evaluating prismatic HTGR analysis codes [42]. The benchmark is sponsored by the OECD's Nuclear Energy Agency, and the 3-year project will yield a set of reference steady-state, transient, and lattice problems that can be used by DOE, NRC, and vendors to assess their codes. The Methods group is responsible for defining the benchmark specifications, leading the data collection and comparison task, and presiding over the annual technical workshops. Participants from Korea, Germany, U.S. universities, and other national laboratories will remain involved in this activity until FY 2015, assuming adequate funding. Several code development and verification activities were based on the MHTGR-350 benchmark.Two conference publications on the preliminary results for Phase I have been published [43]. As part of the benchmark simulation activities, the PHISICS reactor physics package was coupled to the system thermal fluid code RELAP5-3D to enable temperature feedback in core simulations. RELAP5-3D does not provide the spatial resolution of higher fidelity thermal fluid codes, but it does have a powerful plant simulation capability. A detailed study was performed on the spatial resolution required in flux and temperature profiles to support accurate burnup and transient analyses by comparing the traditional ring model approach, depicted in Figure 6, with a much more detailed triangular geometry model (Figure 7).

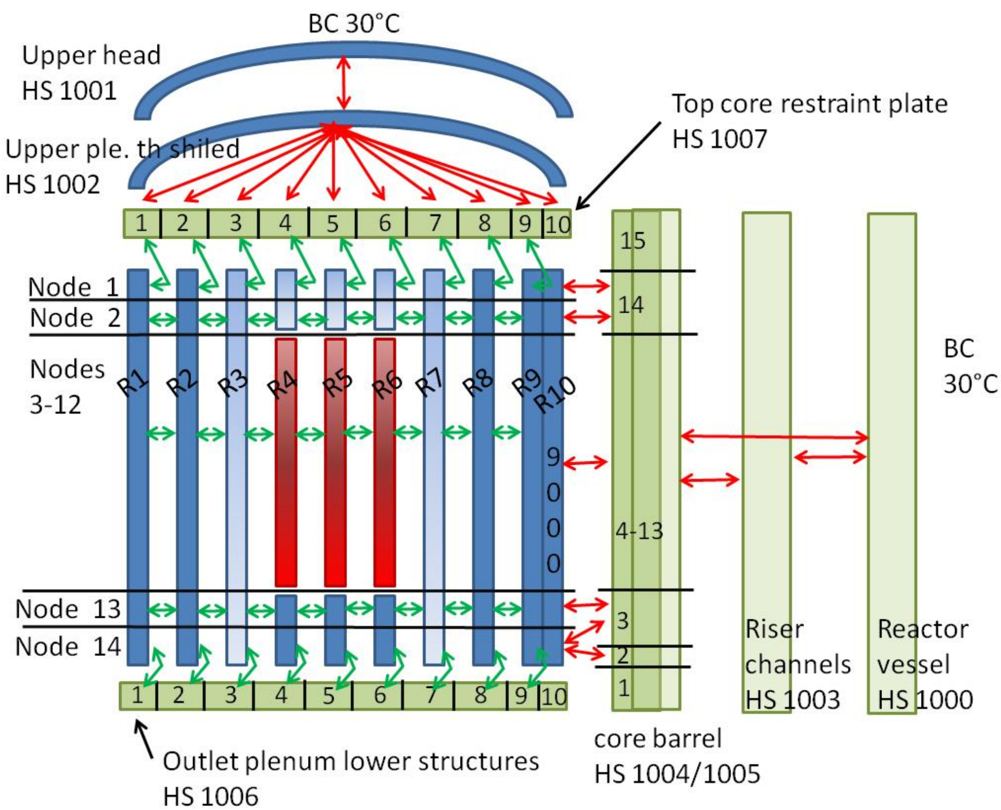

Figure 6. RELAP5 3-D ring-based nodalization of the MHTGR-350.

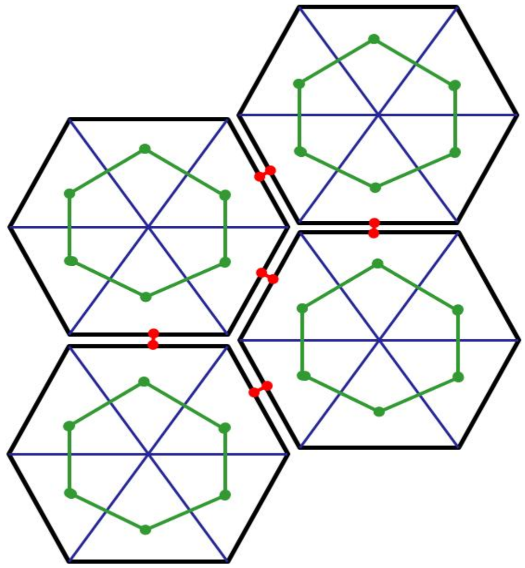

Figure 7. Triangle-based RELAP5-3D model of a cluster of fuel blocks; green and red segments represent conduction and radiation pathways, respectively.

The second international benchmark activity involves the quantification of coupled uncertainties in HTGR reactor lattice, core, and transient simulations. INL played a leading role in the design and launch of a Coordinated Research Program on HTGR Uncertainty Analysis in Modeling that is hosted by the International Atomic Energy Agency. The HTGR Uncertainty Analysis in Modeling benchmark covers both prismatic (MHTGR-350 MW) and pebble bed (HTR-PM 200 MW) designs. The methodology developed for the coupled propagation of uncertainties is based on the stochastic sampling of data and material input uncertainties and the creation of coupled data libraries that can be used in coupled core transients. The Methods group shall use the 44-group co-variance libraries in the new SCALE 6.2 to perform the cell and lattice calculations for Phase I-II, and to use the PHISICS-RELAP5-3D code for the Phase III-IV transient cases.

In a related effort, INL has developed MOOSE for efficient development of simulation tools that can run on supercomputers as well as stand-alone workstations [38]. Although MOOSE was initially applied to fuels performance modeling, a modest effort to model the neutronics and thermal-hydraulics of high-temperature reactor cores was completed successfully in 2009–2010. With some NGNP Methods funding, that code is now being updated for prismatic analysis and testing against the OECD MHTGR-350 benchmark [40]. In FY 2014, an algorithm for online computation of the view factors required for thermal radiation between elements will be added and tested. Besides conduction, thermal radiation is a principal mechanism for heat transfer during a loss of forced cooling, but it is not treated in the high fidelity, multiphysics codes under development at DOE laboratories.

Some pebble bed reactor code development occurred despite the emphasis of the program on the prismatic core. In particular, certain capabilities were added to the PEBBED code [44] to assess the effect of nonlocal heating on normal and accident scenarios, to simulate “batch” loading of fuel, and to simulate absorber (control) pebbles in the reflector. These minor improvements were documented in the NGNP FY 2012 monthly reports. The PEBBLES code was used in conjunction with PEBBED to assess the sensitivity of core burnup and temperature profiles to pebble-to-pebble friction. Finally, the INL lattice code COMBINE was modified to use a one-dimensional discrete ordinates transport solver and incorporated into PEBBED as a subroutine for online cross-section generation. In a unique algorithm, few group cross sections for two or three-dimensional pebble bed core analysis are prepared in a multiscale process involving transport solutions for individual particles, pebbles, and the core itself to account for the double heterogeneity of the fuel, resonance overlap, and the leakage between core and reflector regions. Previously incorporated into PEBBED, the thermal fluid solver THERMIX-KONVEK provides steady-state temperature and coolant flow profiles as well as a transient depressurized loss of fluid simulation capability. In 2011, the decay heat correlation in THERMIX was updated according to the (German) DIN 25485 standard, which accounts for spatially varying burnup and different pebble types. Although none of these codes (PEBBED, THERMIX-KINVEK, and COMBINE) have been subjected to a complete V&V effort, together they provide a pebble bed HTGR simulation capability that can be used for detailed and accurate analysis of any pebble bed reactor and for a wide variety of pebble bed reactor trade and sensitivity studies.

Validation of Monte Carlo N-Particle models was performed using data from the Swiss PROTEUS experiments [45, 46, 47, 48]. These evaluations were published in the International Handbook of Reactor Physics Experiments.

4.5 Design and Safety Methods Future Work/Path Forward

The Experimental V&V team will drive the experimental programs identified in Figure 4 with an aim toward completion in 2015. Concurrently with the experimental programs, an adequacy evaluation of the RELAP5-3D systems analysis code will proceed using the practices and procedures outlined in INL PLN-2498 and Lazor's 2012 thesis [35, 36]. To the extent possible with available funding, university-based experimental validation activities will continue to be guided by NGNP Methods staff.

The Core Simulation group will remain involved in the OECD and International Atomic Energy Agency international prismatic code-to-code benchmark activities during the FY 2013 to FY 2015 period, and if resources and funding levels allow, it will perform additional validation studies on the experimental data sets generated by the HTTR. Additional development of thermal radiation models in prismatic HTGR simulations will occur as needed for accurate execution of the international benchmarks using the high-fidelity multi-physics codes (e.g., RattleS$_N$ake, PRONGHORN) being developed under NEAMS and Light Water Reactor Sustainability funding.

Outstanding challenges remain in the ability to model certain HTGR phenomena. Although the effect on core safety and performance parameters is not fully quantified, the following phenomena have yet to be fully captured with existing tools.

- The physics of neutron scattering by graphite, elastic scattering in heavy metals,
- radiation damage effects on thermal properties of graphite,
- shutdown control rod voids (prismatic core), non-axial and transient pebble flow (pebble bed),
- thermal conductivity at the core –reflector interface

Data are needed in core physics (critical experiments and differential cross sections, particularly at high burnup), ingress (air/water) phenomena, bypass and lower plenum flow, core and plenum-to-plenum heat transfer, and seismically induced geometry distortion. Calculations in support of any license application must, therefore, include large and uncharacterized uncertainties and a design that is unnecessarily (and perhaps uneconomically) conservative.

5. NUCLEAR DATA MANAGEMENT AND ANALYSIS SYSTEM

Experiments under way for the VHTR program provide data to support design and licensing of the NGNP. Fuel and materials to be used in the reactor are being tested and characterized to quantify performance in high-temperature and high-fluence environments. Data used by the program must be qualified for use, stored in a readily accessible electronic form, categorized to ensure the correct data are used, and controlled to prevent data corruption or inadvertent changes. The VHTR program has established the NDMAS to address these requirements.

NDMAS supports data qualification, stores data in a controlled and secure electronic environment, identifies the qualification status of data, provides data analysis and modeling products, and makes data available for use by the program. NDMAS is web-based, so program members can access the system and review the data, obtain analytical results including statistics and graphics, create slide presentations, and download data for advanced analysis. By performing these roles, the NDMAS ensures the correct data are used by the program and that data of known quality will be available to support licensing in the future. The processes and controls put in place by NDMAS will ensure that data associated with licensing requirements will meet ASME Quality Assurance Program (QAP) requirements for Nuclear Facility Applications (NQA-1 Version 1a 2009) collection criteria.

Data streams captured and qualified to date in NDMAS include:

- Graphite characterization—4,494 specimens
- Graphite irradiation—28,202,462 records
- Graphite physics—18,919 records
- Fuel fabrication—6,034 records
- Fuel irradiation—77,865,774 records
- Fuel physics—126,048 records
- Fuel fission product monitoring—39,497
- Fuel PIE —6,408 records
- ATR operating conditions—4,031,071 records
- High-temperature materials—428 specimens.

6. LICENSING STATUS AND PATH FORWARD

6.1 Licensing Background

In 2005, the U.S. DOE established the NGNP project at INL to support near-term commercial deployment of a HTGR technology demonstration plant [49]. Thereafter, NGNP undertook a variety of studies and actions related to HTGR technology R&D, conceptual plant design and engineering, and development of a regulatory framework that can support commercial NGNP licensing. These activities were closely coordinated with the NRC and focused on adapting existing regulatory requirements to accommodate HTGRs. This approach was established jointly by DOE and NRC and was communicated to Congress in 2008 [50].

In keeping with this approach, NGNP systematically examined prior HTGR licensing activities and NRC regulations, as they relate to the NGNP safety case, and reviewed associated plant design goals. The scope and results of this examination were closely coordinated and validated with the NRC staff. In 2009, NGNP used those findings to develop a strategic implementation plan [51] for establishing the regulatory licensing basis necessary to support a successful HTGR license application submission to NRC. The plan focused on the key elements of plant development and licensing, which includes the following:

- Development and understanding of the radiological source term (primarily based on particle fuel design and available qualification testing results)
- Prevention/mitigation of the release of the radiological source term to the environment, including methods for the structured and comprehensive identification of licensing basis event sequences, along with establishment of multiple radionuclide release barriers
- Development of an updated emergency planning structure that included consideration of a collocated industrial end-user to assure protection of the health and safety of the public in the unlikely event of a radiological release.

The design and licensing strategy of NGNP centered on the radionuclide retention capabilities of the TRISO-coated particle fuel and relied less on other barriers for maintaining particle fuel integrity. This strategy—along with related HTGR design goals—was aligned with the NRC Advanced Reactor Policy Statement regarding the pursuit of less complex reactor designs with longer response time constants, passive reactor shutdown and heat removal with limited reliance on operator actions, minimization of severe accident potential, and providing multiple barriers to potential radionuclide releases.

The NGNP approach to addressing these key issues was documented in a series of pre-licensing white papers that were developed with input from DOE and NGNP Licensing Working Group. The NGNP Licensing Working Group included three domestic HTGR design firms and a representative of an owner-operator organization, as well as staff from NGNP R&D, NGNP Engineering, and NGNP Regulatory Affairs. These papers were submitted to NRC for review. Feedback was provided using a review process that consisted of extensive interactions through public meetings, conference calls, and written correspondence related to requests for additional information.

The 2010 DOE "Report to Congress" [52] included an update to the NGNP Combined License Application submittal schedule first proposed in the August 2008 report. As was then stated in subsequent letters to select members of Congress dated October 17, 2011, the public-private partnership to be formed under NGNP was to provide further schedule and milestone updates as contained in the 2010 "Report to Congress."

In his 2011 letter to Congress [1], the Secretary of Energy stated that NGNP would continue at a reduced level of activity with a focus on high-temperature reactor R&D, interaction with NRC on developing an HTGR-compatible licensing framework, and establishing the public-private partnership, which subsequently enables Phase 2 activities to proceed. The current status of the licensing framework development activities are provided in Section 6.3.

6.2 Licensing Objectives

The objectives of the NGNP licensing effort are to establish a regulatory framework compatible with modular HTGR licensing and to identify and resolve key licensing framework topics through interactions with the NRC.

6.3 Licensing Recent Activities

NGNP Regulatory Affairs has undertaken a series of additional pre-licensing activities in support of NGNP deployment, as outlined in the NGNP Licensing Plan. These efforts included:

- Development and delivery of a comprehensive HTGR technology training course to a large number of NRC staff
- Performance of two site hazard assessments to inform subsequent NGNP design/licensing strategies to address likely site hazards and constraints in collocated industrial settings
- Performance of a detailed regulatory gap analysis that examined over 2,500 individual regulatory requirements and implementing guidance for NGNP applicability
- Development of key portions of an HTGR-compatible Combined License Application Format and Content Guide that is similar in structure to NRC's current light water reactor guidance (RG 1.206).

In early 2012, NGNP's DOE/INL team and NRC staff jointly identified four key licensing framework topics. The four longstanding topics were confirmed as areas of most significant regulatory uncertainty for the entire HTGR industry, including NGNP, and were targeted for focused resolution so that they could be reviewed and dispositioned by the NRC Advisory Committee on Reactor Safety (ACRS). The four key topical areas were:

- HTGR containment functional performance
- Licensing basis event selection
- Source terms
- Emergency planning.

As a result of extensive interactions with NGNP, the NRC staff provided draft positions on the four framework topics to ACRS for review in March 2013. Final NRC staff assessment results regarding those positions, including the consideration of the related recommendations from the ACRS, are expected to be released sometime in 2014 and will address the ACRS feedback.

In conclusion, NGNP pre-licensing activities and interactions with NRC have resulted in considerable progress in resolving longstanding HTGR licensing policy and technical issues. The efforts summarized above, and the expected content of the forthcoming NRC assessments, have significantly reduced regulatory uncertainties for HTGR designers and future license applicants, even beyond those currently pertaining to NGNP [52,53,54].

6.4 Licensing Path Forward

Once the final NRC staff assessment results covering the four key regulatory framework issues are received in 2014, NGNP plans to incorporate the NRC's feedback into INL/EXT-13-28205, "NRC Licensing Status Summary Report for NGNP," to reflect those results and will interpret the outcomes as necessary. Additionally, NGNP will incorporate into this report additional project licensing recommendations in response to the NRC findings to assist future applicants in continuing development of the overall HTGR licensing framework.

7. RESEARCH AND DEVELOPMENT PLAN SCHEDULE AND COSTS

In 2011, with DOE not electing to continue the NGNP project into the conceptual design phase until a public-private partnership is formed, the NGNP project outlined the path forward to complete the technology development originally identified at the start of the project. At the same time, INL outlined what it thought was a reasonable reduced scope of R&D activities that can and should be continued without the need for a detailed conceptual design. The objectives of these activities are to: (1) resolve technology specific technical issues associated with gas reactors to lower the barrier for when such a system is deployed in the future and (2) continue R&D to leverage historical investments to reduce uncertainties, increase margin, and better define important attributes of the VHTR with outlet temperatures between 850 and 950°C. Given that significant effort has been expended on R&D to date, it was considered most cost effective to continue the research with a focus on the needs associated with the VHTR or provide useful information for cross cutting activities in the NE reactor R&D portfolio.

7.1 Ongoing Activities

The following activities are planned in the reduced scope program:

TRISO Fuel. Continue with existing program to qualify UCO TRISO fuel fabrication and performance for gas reactors (high burnup; 1,250°C peak temperature under normal operation, 1600-1800°C accident temperatures) and to establish the associated mechanistic source term from the reactor core. The program is being slowed down given the lack of schedule driver.

- Complete AGR-1 PIE and safety testing to establish capability of UCO TRISO under accident conditions to inform safety case for HTGR/VHTRs
- Complete performance demonstration irradiation (AGR-2) and follow-on accident safety testing and PIE to characterize behavior of industrially produced TRISO fuel
- Complete source term experiments (AGR 3/4) to measure fission product release from failed fuel and retention in fuel matrix and graphite and associated high-temperature accident testing, critical for defining the mechanistic source term for gas reactors
- Complete moisture and air ingress testing to better understand fuel behavior and source term characteristics resulting from these ingress events
- Complete formal qualification testing (AGR 5/6) to have a complete qualified TRISO fuel performance dataset that can be used in the future when an HTGR is deployed
- Conduct fuel margin testing (AGR-7) and associated accident safety testing to develop understanding of ultimate performance of TRISO fuel at temperatures above 1250°C
- Complete transfer to fuel fabrication technology to B&W to establish a complete fuel fabrication process that can be used by industry when an HTGR is deployed in the future.

Graphite. Complete a slightly modified version of the original qualification program to demonstrate that adequate grades of nuclear graphite are qualified for use in HTGRs/VHTRs, and mature graphite to be a true nuclear material.

- Complete AGC-1 through 5 to qualify two major grades of graphite from 600-1200°C and 3 to 6 dpa
- Complete high-temperature HTV-1 irradiation to understand graphite behavior at high temperature and high dose
- Complete baseline characterization to understand statistical variability within a block, from block to block, billet to billet and lot to lot
- Continue ASTM and ASME code case efforts.

High Temperature Materials. Focus on testing over the broad range of outlet temperatures (750 to 950°C) to support both HTGR and VHTR applications. Complete testing of Inconel 617 between 750 and 950 to develop database necessary for qualification of the material

- Complete testing of 800H for use between 750 and 850°C
- Complete environmental testing for these nickel-based alloys
- Complete environmental testing for A508/A533 pressure vessel steel for use in gas reactors
- Develop code cases as appropriate.

Design and Safety Methods. Using the process outlined in NRC Reg Guide 1.203, develop data from key scaled integral experiments that can be used to support validation activities for VHTR codes and new advanced tools.

- Complete major validation experiments under way at Oregon State (HTTF)
- Complete major RCCS experiments using the NSTF facility at ANL
- Provide validated data with uncertainties for code V&V
- Participate in, and complete, international code benchmark studies specifically, the OECD MHTGR350 Benchmark of steady state, transient, and lattice codes for prismatic reactors and the IAEA Uncertainty Analysis Methodologies for High Temperature Reactors
- Guide the development and testing of high fidelity multiphysics HTGR analysis capabilities on the MOOSE platform.

7.2 Reduction in Scope

The major technology development activities that are no longer part of the VHTR R&D path forward are listed below.

TRISO Fuel

- Fission product transport activities beyond the graphite, including out of pile test loops, an integral validation experiment, and testing to understand fission product behavior in the reactor building. These are expensive activities, estimated to cost ~ $100 M in the original AGR program plan. The ability to perform this work requires a completed conceptual design and a licensing/safety strategy that identifies the role that the helium pressure boundary and the reactor building play in reducing the source term. The better than anticipated behavior of the TRISO fuel under irradiation and accident testing indicates that tradeoffs exist between the level of retention a reactor vendor wants to take in the reactor core and the level of holdup outside of the core. Without this critical source term information, the entire safety case for collocation cannot be established for NGNP.
- Fuel performance validation experiment AGR-7. Without an identified failure mode of UCO TRISO fuel, developing a validation experiment at this stage is problematic. AGR-7 as currently envisioned in the existing program will focus on margin testing and not a formal validation test as originally planned. A new AGR-7 fuel performance validation campaign would cost about $25 M to complete.
- Fission product transport validation experiment AGR-8. The objective of this in-pile irradiation and the associated accident testing and PIE was to provide data to validate reactor vendor codes used to describe fission product transport in the TRISO fuel, matrix and graphite under normal and accident conditions. Given the design specific nature of this activity it was not included in the reduced scope program. A new AGR-8 fission product transport validation campaign would cost about $25 M to complete.

- Fuel fabrication studies to optimize a commercially viable TRISO fabrication process. The pilot line at B&W today is a process that has not been optimized for commercial production. Additional studies at a cost of between $10-15 M were envisioned in the original program to optimize kernel fabrication, coating, upgrading, and compacting to a level that would improve throughput in anticipation of beginning fabrication of first core to support the EPAct 2005 planned schedule.
- Proof test of fuel from the pilot/production line AGR-9. This irradiation and accident safety testing was identified in all NGNP fuel acquisition studies. It was anticipated that changes could and probably would be made over the current pilot line at B&W, and further that NRC would require such a confirmatory activity. Given the hiatus between the last AGR fuel fabrication campaign (currently expected to complete in early 2015) and the actual production of first core of NGNP with potential disassembly of equipment and loss of trained personnel, this proof test is even more of an imperative if a positive decision is made relative to NGNP deployment. Thus, the AGR program is, in reality, qualifying a UCO TRISO fuel specification but not the actual production line that will be used to fabricate first core. This testing campaign would cost about $25 M.

Graphite

The last graphite irradiation AGC-6 (1200°C, 7 dpa). While the graphite program would like to complete this irradiation to have a more complete dataset of radiation dose and temperature to understand the irradiation creep behavior of graphite, the combination of temperature and dose does not exist in most HTGR designs because of fuel shuffling and potential reflector shuffling envisioned in the core. The irradiation and post-irradiation activities to accomplish this would cost about $14 M.

In all these cases, archive material exists from the AGR and AGC programs that can be used so new materials are not required.

Methods

Except for the support of testing in the Oregon State High Temperature Test Facility and complementary work funded under Nuclear Energy University Programs, all activities to develop physics and thermal hydraulic data sets necessary for code validation and verification are discontinued. The development of modern reactor design and safety analysis tools for VHTR that can capture the important phenomena and scenarios with quantifiable uncertainties has been discontinued except for those that may be funded through the Nuclear Energy Advanced Modeling and Simulation Program. These items in total would cost about $50 M to complete.

7.3 Schedule Considerations

The VHTR R&D program continues to retain these objectives but reduced budgets and facility and resource limitations have slowed progress in the past few years. Currently, completion of the reduced scope program outlined in the 2011 NGNP Path Forward report is expected in 2022 or 2023.

In the interim, logistical issues have impacted and continue to impact the schedule. The ability of the program to make the same rate of progress over the next decade that it has in the past decade will be difficult because of:

- Less operating reactor availability at ATR. More outage time is required per year to accommodate extra experiment removal/handling because of more frequent high power cycles in the operating schedule, and upgrades and modifications to existing aged safety and operating systems
- Upcoming core internals changeout at ATR in 2016
- Multi-program demands for PIE at the Hot Fuels Examination Facility (NGNP, DOE Fuel Cycle fuels program, RERTR, ATR NSUF, Terra Power, and other INL National and Homeland Security programs)

- Lack of some key resources at Hot Fuel Examination Facility (HFEF) (e.g., operators, system engineers)
- Funding uncertainties, impacting the ability to maintain trained and experienced personnel.

As its overall R&D program places less emphasis on NGNP, the DOE is integrating it into its overall Advanced Reactor Technology program. As such, other objectives such as collaborations with China and Japan and our Generation IV partners gain increasing importance and may impact the future trajectory of the reduced scope VHTR R&D program. Figure 8 indicates and illustrates the proposed cost schedule for the completion of the program.

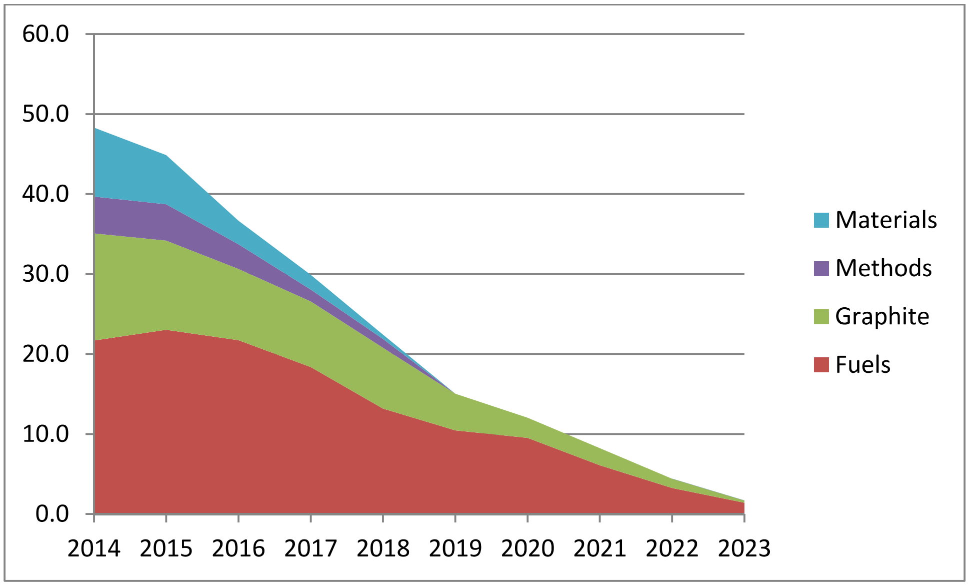

Figure 8. Projected Cost Schedule for Completion.

Table 1 summarizes the total funding that was spent from inception through FY 2013 followed by the projected costs required to complete all the R&D planned as of the end of FY 2013. Project Management costs are included in these numbers. The cost of Licensing (Regulatory Affairs) activities is not included.

Table 1. Actual and Projected Fiscal Year Costs ($M) to Complete the Reduced Scope of R&D Work.

	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017	2018	2019	2020	2021	2022	2023	Total
	ACTUALS											**PROJECTED**										
FUELS	1.3	5.3	7.4	10.5	6.1	18.2	31.0	34.3	32.9	21.9	18.7	25.3	16.0	17.8	15.9	13.5	10.7	11.1	7.5	5.4	0.0	311.0
GRAPHITE			0.6	1.6	3.2	7.6	10.7	10.2	10.0	9.2	6.0	10.6	7.4	6.7	7.1	5.8	4.2	1.3	0.0	0.0	0.0	102.3
HIGH TEMPERATURE MATERIALS	0.3	0.6	2.5	3.9	1.2	2.9	3.6	5.4	6.2	4.4	2.2	1.5	2.5	0.5	0.5	0.0						38.1
METHODS	1.8	2.6	2.1	3.7	1.1	4.7	6.2	7.4	7.8	3.4	1.9	2.2	1.5	1.5	1.0	1.0						49.9
HIGH TEMPERATURE ELECTROLYSIS								3.9	2.7	2.3												9.0
Total by Fiscal Year	3.4	8.4	12.6	19.7	11.6	33.4	51.6	61.2	59.6	41.2	28.8	39.6	27.4	26.5	24.5	20.3	14.9	12.4	7.5	5.4	0.0	510.3

** TDO Project Management costs are spread across the technical areas*

8. QUALITY ASSURANCE

The NGNP project was established under the INL Management and Operations contract between DOE and Battelle Energy Alliance (BEA). The BEA contract requires compliance with 10 CFR 830, Subpart A, "Quality Assurance Requirements," and DOE Order 414.1D, "Quality Assurance," and uses consensus standard NQA-1-2008; 1a-2009, "Quality Assurance Requirements for Nuclear Facility Applications," as the baseline.

The NGNP Quality Assurance Program Description (QAPD) PDD-172 was developed to implement ASME NQA-1-2008, 1a-2009. The QAPD was developed using the Nuclear Energy Institute 11-04 Draft, "Nuclear Generation Quality Assurance Program Description (NG-QAPD)" as a template, and which is based on ASME NQA-1-2008, 1a-2009. The project's QA requirements are based on Regulatory Guide 1.28, Revision 4, "Quality Assurance Requirements (Design and Construction)," and on Regulatory Guide 1.33, Revision 2, "Quality Assurance Program Requirements (Operation)." Regulatory Guide 1.28, Revision 4, states that Part I and Part II requirements of NQA-1-2008, 1a-2009, "Quality Assurance Requirements for Nuclear Facility Applications," provide an adequate basis for complying with the requirements of 10 CFR Part 50, Appendix B, subject to additions and modifications, with specific reference to selected sections of Parts III and IV as identified in the document. The NGNP project was placed under the VHTR Quality Assurance Program Plan (PLN-2690, "VHTR Technology Development Office Quality Assurance Program Plan") in 2012. In many instances, INL procedures are used to implement NGNP QA requirements. Where INL procedures lacked specific requirements and rigor to implement NGNP QA requirements, NGNP project-specific procedures were established and implemented in PLN-2021 "Quality Assurance Program Plan for the Next Generation Nuclear Plant Project," and PLN-2690 "VHTR TDO Quality Assurance Program Plan."

The INL NGNP QAP is described and implemented through a tiered document structure. At the highest level (Tier 1), the QAPD establishes the NGNP project QA requirements, providing an overall description of the QAP and how it is implemented to facilitate understanding of the program scope and structure. The next level (Tier 2) consists of Program Requirements Documents that identify the requirements by program elements contained in standards NQA-1-2008, 1a-2009, ASNT-SNT-TC-1A, and ANSI/NCSL Z540-1, and DOE Guides 414.1-1A, 414.1-2A, 414.1-3, and 414.1-4. The third level (Tier 3) consists of Management Control Procedures, Standards, Guides, Lists, Laboratory Instructions, Technical Procedures, and other site-specific procedures that implement the QAPD at the organizational, functional support area, or facility levels. The objective of the NGNP project is to develop the technology to the extent that a vendor can complete and submit an application to obtain an NRC license to build and operate an HTGR. INL submitted the NGNP QAPD to the NRC for initial feedback.

On September 12, 2012, NRC issued a report titled "Staff Assessment of NGNP Quality Assurance Program Description," concerning the contents of the NGNP QAPD, Revision 3, and related Request for Additional Information responses. In the assessment documented in this report, the staff stated that it covered only the portions of the QAPD applicable to the current scope of NGNP (i.e., nonapplicant activities) and found that the program met the criteria of Appendix B to 10 CFR 50. Thus, the QAPD was deemed acceptable for technology development and high-level design work.

The NRC assessment noted a staff expectation for one of the following:

1. A supplemental QAPD would be submitted by INL should the scope of the NGNP project be expanded to include design and/or construction activities that would warrant INL becoming an applicant in accordance with the guidelines of 10 CFR Part 52
2. Any future applicant or licensee planning to design and/or construct an HTGR based on INL's current R&D efforts would submit an independent QAPD covering the appropriate scope of activities in accordance with the applicable QA regulations and guidance in place at that time.

In October 2012, INL issued Revision 5 of the QAPD that incorporated inputs resulting from NRC interactions and assessment report comments. The QAPD will continue to be applied to NGNP technology development activities until it is accompanied by a license applicant's program as noted in Item 2 above.

A future applicant will most likely utilize the results from the INL's current R&D efforts. INL continues to execute the R&D efforts under PLN-2690. Until a formal applicant for NGNP is identified, PLN-2690 will serve to define the planning, structure, documentation, implementation, and maintenance of the VHTR TDO QAP, which flows from the INL QAPD, PDD-13000, and the NGNP QAPD, PDD-172. A more detailed description of the QAP supporting the NGNP project is provided in PDD-172.

8.1 Contractor Assurance

The Assurance Portfolio [56] for Nuclear Science and Technology identifies VHTR organizational and management-system risks and the assurance activities used to monitor three specific areas of research and design: 1) fuel development and qualification, 2) materials testing and qualification, and 3) design methods and validation. The risk basis determination and associated assurance activities are recorded and managed through the Integrated Assessment System.

Contractor Assurance activities are documented and records maintained in accordance with LWP-13740, "Performing Inspections" [57], LWP-13745, "Performing Surveillances" [58], LWP-13750, "Performing Management Assessments" [59], and LWP-13760, "Performing Independent Assessments" [60]. Assessment issues, observations; and notable practices are documented and resolved through the INL LabWay system (previously Issues and Corrective Action Management System [61]) in accordance with LWP-20000, "Conduct of Research." Lessons Learned are documented in accordance with LWP-13850, "Processing Lessons Learned and Operating Experience Information" [62].

8.2 Path Forward

The VHTR QAP adaptation of ASME NQA-1-2008, 1a-2009 will continue to provide guidance in implementing quality requirements over specific aspects of the program including fuels development, PIE, graphite characterization, and high-temperature metals characterization. Inspections and document reviews will continue to be performed for all AGC and AGR experiments assembled at the Test Train Assembly Facility at the ATR facility.

All quality purchases and contract requisitions will continue to be reviewed and approved by VHTR TDO quality engineering. As required by NQA-1, all activities will continue to be reviewed and observed annually including preparation for DOE audits. Corrective action will be managed through the LabWay system.

The NGNP project scope could be transitioned to another entity (license applicant). In that case, a QAP implemented and maintained by that entity will be required. Selected portions of the remaining INL quality affecting work scope (presumably R&D activity) would be subject to review and acceptance by that entity.

9. REFERENCES

1. Chu, S., Secretary of Energy's Letter to U.S. Senate Appropriations Committee, October 17, 2011.
2. *A Technology Roadmap for Generation IV Nuclear Energy Systems*, A report issued by the U.S. DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, GIF-002-00, December 2002.
3. Petti, D. A., J. T. Maki, P. A. Demkowicz, and R. R. Hobbins, "TRISO-coated Particle Fuel Performance," In Rudy Konings (ed.) Comprehensive Nuclear Materials Volume 3, Amsterdam: Elsevier BV, 2012, pp. 151–213 (invited).
4. Petti, D. A., J. T. Maki, J. Hunn, P. Pappano, C. Barnes, J. Saurwein, S. Nagley, J. Kendall, and R. R. Hobbins, "The DOE Advanced Gas Reactor Fuel Development and Qualification Program," The Journal of The Minerals, Metals & Materials Society, Vol. 62, No. 9, September 2010, p. 62–66.
5. Nagley, S. G., C. M. Barnes, D. L. Husser, M. L. Nowlin, and W. C. Richardson, "Fabrication of Uranium Oxycarbide Kernels for HTR Fuel," 5th International Topical Meeting on High Temperature Reactor Technology, (HTR 2010), Prague, Czech Republic, October 18–20, 2010.
6. Barnes, C. M., D. W. Marshall, J. T. Keeley, and J. D. Hunn, "Results of Tests to Demonstrate a Six-Inch Diameter Coater for Production of TRISO-Coated Particles for Advanced Gas Reactor Experiments," 4th International Topical Meeting on High Temperature Reactor Technology, (HTR 2008), Washington D.C., October 2008.
7. Phillips, J. A., E. L. Shaber, J. J. Einerson, D. A. Petti, S. E. Niedzialek, W. C. Richardson, and S. G. Nagley, "Compacting Scale Up and Optimization of Cylindrical Fuel Compacts for the Next Generation Nuclear Plant," 6th International Topical Meeting on High Temperature Reactor Technology, Proceedings of the HTR 2012 Tokyo, Japan, October 28–November 1, 2012.
8. Grover, S. B., D. A. Petti, and J. T. Maki, "Completion of the First NGNP Advanced Gas Reactor Fuel Irradiation Experiment, AGR-1, in the Advanced Test Reactor," 5th International Topical Meeting on High Temperature Reactor Technology, (HTR 2010), Prague, Czech Republic, October 18–20, 2010.
9. Jason M. Harp, Paul A. Demkowicz, Scott A. Ploger, "Post-irradiation Examination and Fission Product Inventory Analysis of AGR-1 Irradiation Capsules, Proceedings of the HTR 2012, Tokyo, Japan, October 28 – November 1, 2012
10. Paul A. Demkowicz, John D. Hunn, Robert N. Morris, Jason M. Harp, Philip L. Winston, Charles A. Baldwin1, Fred C. Montgomery, "Preliminary Results of Post-Irradiation Examination of the AGR-1 TRISO Fuel Compacts, Proceedings of the HTR 2012, Tokyo, Japan, October 28 – November 1, 2012
11. Charles A. Baldwin, John D. Hunn, Robert N. Morris, Fred C. Montgomery, G.W. Chinthaka Silva, and Paul A. Demkowicz, "First Elevated Temperature Performance Testing of Coated Particle Fuel Compacts from the AGR-1 Irradiation Experiment" Proceedings of the HTR 2012, Tokyo, Japan, October 28 – November 1, 2012
12. S. Blaine Grover and David A. Petti , "Status of the NGNP Fuel Experiment AGR-2 Irradiated in the Advanced Test Reactor," 6th International Topical Meeting on High Temperature Reactor Technology (HTR 2012), Tokyo, Japan, October 28 – November 1, 2012
13. S. Blaine Grover and David A. Petti, "Design and Status of the NGNP Fuel Experiment AGR-3/4 Irradiated in the Advanced Test Reactor," 6th International Topical Meeting on High Temperature Reactor Technology (HTR 2012), Tokyo, Japan, October 28 – November 1, 2012.

14. Windes, W., T. Burchell, and R. Bratton, “Graphite Technology Development Plan,” INL/EXT-07-13165, Idaho National Laboratory, September 2007.
15. Burchell, T., R.L. Bratton, R.N. Wright, J. Wright, “Next Generation Nuclear Plant Materials Research and Development Program Plan,” Revision 4, INL/EXT-06-11701, Idaho National Laboratory, October 2007.
16. Wright, R. N., J. Wright, T. L. Sham, R. Nanstad,W. Ren, “Next Generation Nuclear Plant Pressure Vessel Materials Research and Development Program Plan,” Revision 4, PLN-2803, Idaho National Laboratory, April 2008.
17. Hurley, D. H., S. J. Reese, R. Kennedy, F. Farzbod, and R. N. Wright, “Characterization of High Temperature Mechanical Properties Using Laser Ultrasound,” International High-Power Laser Ablation Conference, Santa Fe, New Mexico, April 30–May 3, 2012.
18. Wright, J. K., L. J. Carroll, J. Simpson, and T. L. Sham, “Tensile Strain-Rate Sensitivity of Alloys 800H and 617,” *ASME Pressure Vessels and Piping Conference, Paris, France, July 14–18, 2013*.
19. Carroll, L. J., C. J. Cabet, and R. N. Wright, “Creep and Environmental Effects on the High Temperature Creep-Fatigue Behavior of Alloy 617,” *Journal of ASTM International*, Vol. 8, No. 6, June 2011, doi:10.1520/JAI103797.
20. Cabet, C. J., L. J. Carroll, and R. N. Wright, “The Role of Environment on High Temperature Creep-Fatigue Behavior of Alloy 617,” *Journal of Pressure Vessel Technology*, in press.
21. Carroll, L. J., R. N. Wright, and C. J. Cabet, “The Role of an Impure Helium Environment on High Temperature Creep-Fatigue Behavior of Alloy 617,” *ASME Pressure Vessel and Piping Conference, Seattle, Washington, July 18–22, 2010*.
22. Carroll, L. J., R. N. Wright, and C. J. Cabet, “Alloy 617 Creep-Fatigue Behavior in Impure Helium: Effect of Cyclic Loading and Environment,” *OECD Nuclear Energy Agency Second International Workshop on Structural Materials for Innovative Nuclear Systems, Daejeon, Korea, August 31–September 3, 2010*.
23. Wright, J. K., L. J. Carroll, J. K. Benz, W. R. Lloyd, J. A. Chapman, and R. N. Wright, “Characterization of Elevated Temperature Properties of Heat Exchanger and Steam Generator Alloys,” *5th International Conference on High Temperature Reactor Technology (HTR 2010), Prague, Czech Republic, October 18–20, 2010*.
24. Cabet, C., L. Carroll, R. Wright, and R. Madland, “Creep-fatigue of High Temperature Materials for VHTR: Effect of Cyclic Loading and Environment,” *Proceedings of ICAPP 2011, Nice, France, May 2–6, 2011*.
25. Carroll, L. J., W. R. Lloyd, J. R. Simpson, and R. N. Wright, “The Influence of Dynamic Strain Aging on Fatigue and Creep-Fatigue Characterization of Nickel-Based Solid Solution Alloys,” *Materials at High Temperatures*, Vol. 27, No. 4, January 2011, pp. 313–323.
26. Carroll, L. J., C. Cabet, M. C. Carroll, and R. N. Wright, “The Development of Microstructural Damage During High Temperature Creep-Fatigue of a Nickel Alloy,” accepted by *International Journal of Fatigue.*
27. Wright, J. K., J. A. Simpson, L. J. Carroll, and R. N. Wright, “Low-cycle Fatigue of Alloy 617 at 850 and 950°C,” accepted by *ASME Journal of Engineering Materials and Technology.*
28. Totemeier, T. C., R. N. Wright, and J. R. Simpson, “Effect of Extended Aging at 1000°C on Microstructure and Properties of Alloy 617,” *ASME Pressure Vessels and Piping Conference, Vancouver, Canada, July 23–27, 2006*.

29. Wright, R. N. "High Temperature Behavior of Candidate VHTR Heat Exchanger Alloys," *4th International Conference on High Temperature Reactor Technology (HTR 2008), Washington D.C., September 29–October 1, 2008*.

30. Rabin, B. H., W. D. Swank, and R. N. Wright, "Thermophysical Properties of Alloy 617 from 25 to 1000°C," submitted to *Nuclear Engineering and Design Journal.*

31. Gougar, H. and R. Schultz, *Next Generation Nuclear Plant Methods Research and Development Program Plan, Revision 2*, PLN-2498, September 2010.

32. Ball, S. J. and S. E. Fisher, Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), NUREG/CR-6944, March 2008.

33. Zuber, N. "Appendix D: A Hierarchical Two-Tiered Scaling Analysis," *An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution*, NUREG/CR-5609, 1991.

34. Schultz, R. R., "Using CFD to Analysis Nuclear System Behavior: Defining the Validation Requirements," *Proceedings of the OECD Nuclear Energy Agency CFD4 NRS-4 Workshop, Daejon, Korea, September 10–12, 2012*.

35. PLN-2498, "Next Generation Nuclear Plant Methods Technical Program Plan," Rev. 3, Idaho National Laboratory, December 21, 2010.

36. Lazor, M. Thesis on *Evaluation of Assessment Techniques for Verification and Validation of the TRACE Systems Analysis Code*, Pennsylvania State University, 2012.

37. Gougar, H. D., A. M. Ougouag, W. K. Terry, and K. N. Ivanov, "Automated Design and Optimization of Pebble Bed Reactor Cores," *Nuclear Science and Engineering*, Vol. 165. No. 3, July 2010, pp. 245–269.

38. Gaston, D., *MOOSE Enhancements Toward Delivery of an Integrated Framework*, INL/EXT-13-29814, Idaho National Laboratory, July 2013.

39. A. Bingham, *PRONGHORN User Manual*, INL/EXT-12-27784, Idaho National Laboratory, December 2012.

40. Ortensi, J., "Supercell Depletion Studies for Prismatic High Temperature Reactors," Proceedings of HTR2012, Tokyo, Japan (2012) (INL/CON-12-24395).

41. *International Handbook of Evaluated Reactor Physics Benchmark Experiments*, OECD/NEA 7140, March 2013.

42. Ortensi, J., et al., "Prismatic Core Coupled Transient Benchmark," *2011 ANS Annual Meeting, Hollywood, Florida, June 26–30, 2011*, INL/CON-11-20811.

43. Strydom, G. and A. Epiney, "Relap5-3D results for Phase I (exercise 2) of the OECD/NEA MHTGR-350 MW benchmark," *Proceedings of the 2012 International Congress on the Advances in Nuclear Power Plants (ICAPP2012), Chicago, Illinois, June 24–28, 2012*, INL/CON-12-24604.

44. Epiney, A., et al., "New Multi-group Transport Neutronics (PHISICS) capabilities for RELAP5-3D and its Application to Phase I of the OECD/NEA MHTGR-350 MW Benchmark," *Proceedings of 6th International Topical Meeting on High Temperature Reactor Technology (HTR2012), Tokyo, Japan, October 28 – November 1, 2012*.

45. Williams, T., *LEU-HTR PROTEUS: Configuration Descriptions and Critical Balances for the Cores of the HTR-PROTEUS Experimental Programme*, TM-41-95-18, v. 1.00, Paul Sherrer Institute, Villigen, November 25, 1996.

46. Bess, J., *HTR-Proteus Pebble Bed Experimental Program Core 4: Random Packing with a 1:1 Moderator-to-Fuel Pebble Ratio*, INL/EXT-12-27057, Idaho National Laboratory, March 2013.

47. Bess, J., *HTR-Proteus Pebble Bed Experimental Program Cores 1,1A, 2, and 3A: Hexagonal Close Packing with a 1:2 Moderator-to-Fuel Pebble Ratio*, INL/EXT-11-23219, Idaho National Laboratory, March 2013.
48. Bess, J., *HTR-Proteus Pebble Bed Experimental Program Cores 9 and 10: Columnar Hexagonal Point-on-Point Packing with a 1:1 Moderator-to-Fuel Pebble Ratio*, INL/EXT-11-26334, Idaho National Laboratory, March 2013.
49. Public Law 109-58, "Energy Policy Act of 2005," Energy Efficiency and Renewable Energy (EERE), Federal Energy Management Program, July 29, 2005.
50. DOE, Next Generation Nuclear Plant Licensing Strategy: A Report to Congress, U.S. Department of Energy, Office of Nuclear Energy, August 2008.
51. PLN-3202, "NGNP Licensing Plan," Rev. 0, Idaho National Laboratory, June 26, 2009.
52. DOE, Next Generation Nuclear Plant: A Report to Congress, U.S. Department of Energy, Office of Nuclear Energy, April 2010.
53. Petti, D., INL, to NRC, "Contract Number DE AC07-05ID14517 – Next Generation Nuclear Plant Submittal – Confirmation of Requested NRC Staff Positions – NRC Project #0748," CCN 227793; ML121910310, July 6, 2012.
54. Sam, A. J. to R. W. Borchardt, NRC, "Next Generation Nuclear Plant (NGNP) Key Licensing Issues," ML 13135A290, May 15, 2013.
55. Moe, W. and J. Kinsey, "NGNP Licensing Status Summary Report," INL/EXT-13-28205, Idaho National Laboratory, July 2013.
56. PLN-3174, "Assurance Portfolio for Nuclear Science & Technology," Rev. 5, Idaho National Laboratory, May 2013.
57. LWP-13740, "Performing Inspections," Rev. 2, Idaho National Laboratory, January 2011.
58. LWP-13745, "Performing Surveillances," Rev. 1, Idaho National Laboratory, January 2011.
59. LWP-13750, "Performing Management Assessments," Rev. 1, Idaho National Laboratory, April 2008.
60. LWP-13760, "Performing Independent Assessments," Rev. 1, Idaho National Laboratory, April 2008.
61. LWP-13840, "Management of Issues, Observations, and Noteworthy Practices", Rev. 4, August 2012.
62. LWP-13850, "Processing Lessons Learned and Operating Experience Information," Rev. 3, Idaho National Laboratory, May 2009.