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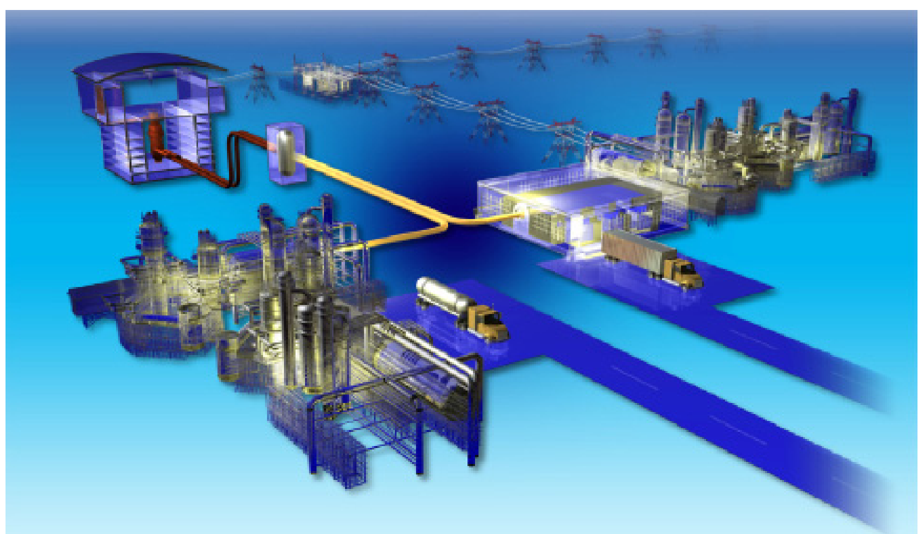
Rev. 2

Next Generation Nuclear Plant Methods Technical Program Plan

September 2010



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

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REVISION LOG

Rev.	Date	Affected Pages	Revision Description
0	DRAFT	All	Newly issued document
1	9/26/08	Chapter 5 All	Chapter 5,"neutronics an Reactor Kinetics Analysis Development" has been completely rewritten to emphasive the needs for new or enhanced neutronic methods such as PEBBED-THERMIX-COMBINE and RELAP-CYNOD-COMBINE. Minor editorial and grammatical changes throughout the document.
2	09/27/10	All	Goals, Assumptions, and Requirements added. Program divided into two major elements: Experimental V&V and Core Simulation. High Temperature Test Facility and potential dust/fission product testing facilities added to the Experimental V&V section. Multiphysics simulations and developments and needs for cross section generation added to the Core Simulation section. Overall condensation of technical detail along with editorial and grammatical changes.

Next Generation Nuclear Plant Methods Technical Program Plan

Idaho National Laboratory

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EXECUTIVE SUMMARY

One of the great challenges of designing and licensing the Next Generation Nuclear Plant (NGNP) is to confirm that the intended analysis tools can be used confidently to make decisions and to assure that all the reactor systems are safe and meet the performance objectives of the Generation IV Program. The research and development (R&D) projects defined in the NGNP Design Methods Development and Validation Program will ensure that the tools used to perform the required calculations and analyses can be are validated and verfied. The Methods R&D tasks are designed to ensure that the calculational envelope of the tools used to analyze high temperature gas-cooled reactor (HTGR) systems encompasses, or is larger than, the operational and transient envelope of the HTGR itself. The methods to be developed and employed for HTGRs with historical outlet temperature (700^{o}C to 850^{o}C) may also be used for reactors with higher temperatures, i.e. the so-called Very High temperature Reactor (VHTR).

Methods research and development focuses on the development and validation of tools to assess the neutronic and thermal fluid behavior of the plant. The fuel behavior and fission product transport models are discussed in the Advanced Gas Reactor program plan. Fuel particle performance and dimensional changes in graphite are also directly related to the neutronic and thermal fluid behavior of the fuel and are being addressed as part of the long-term core simulation effort described in this Plan.

The calculational envelope of the neutronics and thermal-fluids software tools intended to be used on the NGNP is defined by the scenarios and phenomena that these tools can calculate with confidence. The software tools can only be used confidently when the results they produce have been shown to be in reasonable agreement[a] with first-principle results, thought-problems, and data that describe the highly ranked phenomena inherent in all operational conditions and important accident scenarios for the HTGR.

The R&D process is informed by Regulatory Guide 1.203 of the Nuclear Regulatory Commission. It is a well-established process by which key safety scenarios and phenomena are identified and ranked according to their importance to safety and the analysts' state of knowledge of their defining parameters. The calculational envelope of the model must fully encompass the operating envelope of the plant. The code must be shown to reproduce both the individual physics (flow patterns, heat transfer, reaction rates) and the integral behavior of the plant. If the code is incapable of modeling some of the physics with sufficient fidelity, further code development is warranted. The full range of validation studies must be completed prior to performing the required analyses with confidence in the result.

A design for the demonstration HTGR has not yet been selected. Consequently, the R&D process is focused on scenarios and highly ranked phenomena that have already been identified as important by the advanced gas-cooled reactor community for the designs being considered as candidates for the HTGR. This approach has resulted in a HTGR -specific PIRT from which the methods R&D is being defined using the following assumptions:

1. The selected NNGP design could be either a pebble-bed or a block-type reactor.
2. The calculational and experimental needs, and consequently the required R&D, are focused in six distinct areas based on the relative state of the software in each:
    a. Basic differential and integral nuclear cross-section data measurement and evaluation, including mathematically rigorous sensitivity studies of the effects of uncertainties in the differential nuclear data and other independent design variables on key integral reactor properties (the task of

a. Reasonable agreement is achieved when the calculation generally lies within the uncertainty band of the data used for validation and always shows the same trends as the data. Code deficiencies are minor.

characterizing the effects of the nuclear fuel, fission products, moderator, and other relevant materials on the system reactivity, neutron flux distribution, and power production)

b. Reactor assembly cross-section preparation (the task of translating the fundamental data characterized in area (a) into formats and states useful for core burnup and dynamic analysis)

c. Reactor core simulation (the task of computing the core flux, power, temperature, coolant flow, and burnup profiles for all anticipated steady state operating scenarios)

d. Reactor kinetics (calculation of spatial changes in flux, power, and temperature level as functions of time during postulated transients)

e. Fuel and material behavior (calculation of the effect of neutronic and thermal fluid behavior on the fuel and core structures).

f. Fission product transport (determination of fission product movement once fission products have escaped from the confines of the fuel).

Methods R&D is tailored to follow the guidance and timelines defined by the *Energy Policy Act of 2005*. Through 2013, Methods R&D will be performed to enable analyses to be performed that can characterize the behavior of the candidate NGNP designs. Phase 1 covers the period beginning from the passage of the Energy Policy Act until the design is selected. Phase 2 will begin when Phase 1 is completed. During Phase 2, validation of the software tools will be completed using data directly scaled to the NGNP design and the operational, off-normal, and accident behavior of the design will be analyzed.

The commercial companies (for example Areva, General Atomics, and Westinghouse) that are currently designing the future gas-cooled reactors are still, in large measure, using legacy analysis tools to describe the operating and accident characteristics of their designs and intend to use them for licensing purposes. Likewise, for the purposes of evaluating an NGNP license application, the Nuclear Regulatory Commission is developing its own evaluation models using modifications or enhancements of available tools. The levels of uncertainty in the accuracy and fidelity of these tools have yet to be quantified. The role of the NGNP Methods program is to provide an independent and accessible validation, verification, and high fidelity simulation capability against which all NGNP stakeholders can benchmark their tools.

This role is carried out by (1) conducting or coordinating thermal fluid experiments that provide data for validating computational fluid dynamics and system codes; (2) the development of high-fidelity core and plant simulation tools specifically geared toward the modeling of scenarios and phenomena with large uncertainties or that exhibit complex neutronic, thermal-hydraulic, and material interaction; and (3) integrating the diverse but dispersed high-temperature reactor knowledge base through frequent communication with stakeholders. This communication includes:

- Regular telecons with the NRC to discuss HTGR Methods issues
- Steering committees for thermal fluid experiments that include representatives from the NRC, vendors, national labs, and academia
- Leadership of or full participation in code benchmark or comparison activities
- Workshops for exchanging ideas and results and for developing strategy.

To support project objectives and schedule, Methods activities will follow the schedule shown in Figure ES-1.

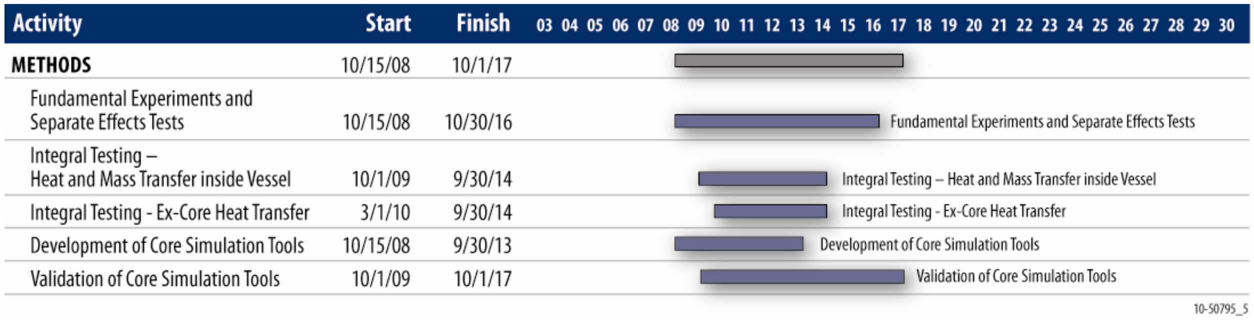

Figure ES-1. High level methods development and experiment schedule.

The highest-priority Methods activities for FY 2011 through 2013 will include: conducting integral experiments in the High Temperature Test Facility, completing and operating the Natural Circulation Shutdown Test Facility for investigation of ex-core heat removal, performing bypass and air ingress experiments with associated computational fluid dynamics model validation, and completion of the development of three-dimensional core simulation tools for analyzing complex core behavior under anticipated normal and off-normal conditions including a range of loss-of-forced-cooling events.

In subsequent years, these tools will be validated and verified using experimental data and used to investigate complex phenomena as required for understanding of HTGR systems.

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ACRONYMS

AVR	Arbeitsgemeinschaft Versuchsreaktor
ANL	Argonne National Laboratory
CFD	Computational fluid dynamics
CFR	Code of Federal Regulations
DCC	Depressurized conduction cooldown scenario
DOE	U.S. Department of Energy
EMDAP	Evaluation Model Development and Assessment Process
FY	fiscal year
GIF	Generation IV International Forum
HTGR	High Temperature Gas-Cooled Reactor
HTR-10	Chinese High Temperature Gas-Cooled Reactor
HTR	High-temperature reactor
HTTR	High-Temperature Engineering Test Reactor
IAEA	International Atomic Energy Agency
INET	Institute of Nuclear Energy Technology
INL	Idaho National Laboratory
JAEA	Japan Atomic Energy Agency (formerly JAERI)
LWR	light water reactor
MHTGR	modular high temperature reactor
NGFM	Nodal Green’s Function Method
NGNP	Next Generation Nuclear Plant
NRC	U.S. Nuclear Regulatory Commission
PBMR	pebble-bed modular reactor
PBR	pebble bed reactor
PCC	pressurized conduction cooldown scenario
RCCS	reactor cavity cooling system
R&D	Research and Development
TRISO	tri-structural isotorpic (ceramic-coated-particle fuel)
V&V	Verification and validation

1. INTRODUCTION

President George W. Bush signed the *Energy Policy Act of 2005* on August 8, 2005. As summarized in the September 2005 issue of Nuclear News, page 15, in the article, "Compromise Energy Bill Becomes Law," the nuclear provisions specific to the Next Generation Nuclear Plant (NGNP) are:

> The DOE shall establish the Next Generation Nuclear Plant project, with a prototype to be sited at Idaho National Laboratory. The centerpiece is to be the development of reactor, fuel, and associated technology for the production of hydrogen as well as electricity. The DOE and the NRC are to submit jointly a licensing strategy to Congress within three years after enactment. Hydrogen production technology and initial reactor design parameters are to be chosen by September 30, 2011, or an alternative date is to be submitted to Congress by that time. The reactor is to begin operation by September 30, 2021, or an alternative date is to be submitted to Congress by that time. The project is authorized to receive $1.25 billion over fiscal years 2006 through 2015, and such sums as are necessary thereafter.

Research and development (R&D) specific to the NGNP mentioned in the *Energy Policy Act* and conducted to date is based on the gas-cooled very high temperature reactor (VHTR) concept promulgated in the Generation IV Technology Roadmap [Generation IV International Forum 2002]. Presently, the most likely VHTR candidates are the prismatic and pebble-bed designs. Consequently, the R&D described in this document is focused on these types of gas-cooled thermal reactors.

The purpose of this revision is to describe the updated plan for the development and validation of the methods used to analyze and license the NGNP and incorporates the experience and knowledge gained from ongoing and completed work since the release of the original technical plan in April 2007 [Schultz 2007]. This revision also summarizes the progress made in the program to date and the activities remaining to complete the program.

In the Gen-IV analyses, two HTGR concepts excelled in meeting program goals, the prismatic reactor and the pebble bed reactor (PBR). The overall layout, functionality, and behavior of the concepts are quite similar with the primary difference being the geometry and mobility of the fuel. The general approaches used in the simulation of core neutronics and thermal fluid behavior, therefore, are the same. The differences do have an impact on the specific steps taken, so are they are briefly discussed in this plan.

1.1 The Need for NGNP Analysis Methods Development and Qualification

The status of methods currently available for designing and analyzing the HTGR can be summarized with the following statements:

- State-of-the-art software and advanced, detailed methods are not ready to perform design and analysis to the standard required by the HTGR. Considerable validation, and limited development of the necessary software tools, is required.
- The above conclusion also applies to present software capabilities to perform NGNP licensing calculations.
- The practices and procedures acceptable for both validating and developing the necessary software tools for the HTGR must be defined and implemented to a standard defined by the engineering community.

These statements are true because (a) the tools developed under the early high temperature gas-cooled reactor (HTGR) programs were subject to limitations and and assumptions that were required of the simulation capabilities available decades ago but can now be relaxed or refined, and (b) these methods do not account for some of the detailed and complex phenomena anticipated for these reactors and thus yield results with significant uncertainties, (c) software tools that have a low calculational uncertainty will be required to analyze the behavior of the HTGR to enable the plant to operate safely at a high efficiency with a competitive economic margin, and (d) most of the software tools that will be used have not been validated for the scenarios and phenomena that must be analyzed for licensing. For example, although systems analysis software has been validated for selected cases, a full validation has not been performed nor are the data available that will enable a full validation to be performed. Also, computational fluid dynamics (CFD) software, which will be widely used to analyze the HTGR behavior, have never been used in large measure to perform auditing, design, or licensing calculations for a nuclear plant. If a license application were to be submitted today using the existing tools, it is anticipated that the regulator would request a considerable amount of additional information and validated quantification of the uncertainties associated with the reported results.

Finally, the risks involved with a complex engineering project of this nature can be greatly reduced with a rigorous, front-loaded modeling effort that can resolve key performance and safety issues early in the design process. The cost of design changes increases exponentially as the project matures and the licensing process has begun. A comprehensive evaluation model with a low degree of calculational uncertainty can facilitate the design process and increase the confidence in the results obtained by the designers and regulators.

1.2 Categories of, and General Approach to, Core Simulation

For licensing purposes, the complete set of codes and models used to analyze a nuclear plant is called the Evaluation Model. Other codes may supplement these for purposes of core and plant design. Codes and models may be categorized by their ultimate use or by how the uncertainty in their results is treated. These are described in the following subsections.

1.2.1 End Use Categorization

As is the case with light water reactors (LWRs), reactor core simulation entails the use of neutronics calculations to determine the neutron flux, power, and burnup (transmutation) profiles and thermal fluid calculations to obtain the temperature and coolant mass flow distributions at selected times during a fuel cycle. The specific type of calculation depends upon the parameter(s) of interest, and the objective of the analysis and the tools to be used will vary in functionality and fidelity. The types of calculations needed may be categorized as follows:

- Design (optimization, sensitivity analyses, economics, proliferation)
- Benchmarking and verification and validation (V&V; fundamental and separate effects phenomena, integral experiments)
- Safety analysis (normal and off-normal transients, fission product transport, cliff-edge behavior).

The engineered coolant channels in the blocks enable one-dimensional (1-D) subchannel analysis of coolant temperature inside the fuel columns, which is adequate for many design and safety calculations. In pebble bed analysis, the bed is assumed to be a homogeneous but porous medium in which the bulk coolant temperature is coupled to the solid temperature through a heat transfer correlation. Validated empirical heat transfer and pressure drop correlations exist for such geometries. The flow distribution in the core is determined from a common core pressure drop between the upper and lower plenum. The local coolant temperatures, combined with the power profile obtained from the neutronics calculations, can be used to compute the temperature distribution within the block or pebble bed. The codes that perform such calculations (e.g., RELAP [Shultz 1993] and TINTE [Reitsma 2005]) run fast enough to support extensive design calculations and larger plant simulations that explore the time-dependent coupling between the core and the balance of plant.

Error and uncertainties arise with certain phenomena and scenarios. The flow of hot helium out of the core, into the lower plenum, and into the hot duct is not captured well using low-order codes like RELAP. The helium jets emerging from the core and impinging upon the core support structures of the lower plenum may exhibit large variations in temperature and velocity leading to uncertain impact on the mechanical integrity of these structures. The graphite core and reflector blocks shrink and swell as complex functions of irradiation and temperature. This leads to the formation of gaps between the blocks through which coolant will flow. The nature of this bypass flow is still under study in order to assess the impact on temperature profiles within the fuel blocks. Computation of the flow and temperature distribution in the lower plenum or for a given gap geometry is possible with the use of CFD codes. CFD calculations are computationally demanding and thus not yet practical for routine whole core simulations. Standards and practices for their use in reactor analyses are still being developed, and thus CFD is still considered as much art as science. Nonetheless, CFD may play an important role in the understanding and licensing of HTGRs.

Likewise, neutronics codes can also be segregated into low-order tools suitable for whole-core design and transient analysis and high resolution, high fidelity tools for benchmarking and separate effects studies. MCNP [Briesmeister] is the workhorse of the latter category because of its extensive user base, ability to model geometry exactly, and continuous energy treatment of neutron interactions. Like CFD, Monte Carlo simulations are, at least for now, computationally too demanding to be used for extensive design optimization, whole-core fuel management, and transient analysis. They are, however, the preferred option for investigating neutronic behavior in specific components. When coupled to a suitable depletion code like ORIGEN, burnup and irradiation damage in small components or for specific core conditions can be studied in considerable detail. Monte Carlo calculations can yield very accurate results for critical experiments in which burnup and temperature variations are negligible or well-characterized. More recently, deterministic transport codes have progressed to the point in which whole core calculations are possible in limited cases. These codes offer the fidelity of Monte Carlo simulations but can also generate detailed flux and power profiles.

Low-order (usually multigroup diffusion) approximations to the neutron transport equation are still preferred for fuel management and transient analysis of a large HTGR core, because of the computational efficiency. The use of such codes requires the generation of multigroup diffusion parameters (cross sections, diffusion coefficients, and kinetic parameters) using a procedure called homogenization. Higher-order transport codes are used to simulate local regions with high fidelity and generate average values that are then used in the whole core diffusion code. The challenge in HTGRs is to generate these nodal parameters in such a way that captures the effects of the fuel and core geometry and scattering in graphite. Well-established techniques and codes used for LWR simulation and the early HTGR programs do not do this with sufficient accuracy to address current design needs.

Fuel management codes must compute the local flux, power, and temperature profiles and then use this information to vary the composition in the blocks or pebbles as a function of time. The fuel management module must be able to compute and store composition data for each block or batch of pebbles as they are burned during and shuffled between each cycle. The power profiles at each specified burnup step provide an initial condition for transient analyses. Thermal fluid analyses generate the coolant and fuel temperature data that determine if the core is operating within design specifications. These analyses are dominated by the thermal coupling between the solid structures and the coolant, and neutronic feedback is captured with simple reactivity coefficients or quasi-static kinetic modules to generate time-dependent power data.

1.2.2 Best Estimate versus Conservative

Figures-of-merit for the present fleet of LWRs were traditionally calculated using conservative assumptions and approaches that were guaranteed to yield calculated results with very large safety margins. Models of this sort were based on prescription of sometimes arbitrary restrictions (for example, neglecting heat transfer for certain phases of a scenario) to ensure a large safety factor was present in the licensing calculations. The approach codified in Title 10 of Part 50.46, Appendix K of the Code of Federal Regulations is the most widely known example. The major drawback to the Appendix K approach is that the calculational uncertainty, while known to be large and conservative, is not quantified.

Subsequent to the Appendix K approach, best-estimate approaches were developed and have been used to perform some plant license reevaluations. The best estimate approaches have the advantage of enabling the calculational uncertainties to be defined and quantified. However, 1-D fluid flow models were almost exclusively used to calculate average or bulk values of the figures-of-merit in the various regions of the plant. Thus, to account for potential deviations from 3-D behavior in the 1-D model results, safety factors have been used to provide a sufficient margin from the limiting value.

The concepts discussed in the above two paragraphs are illustrated in Figure 1. The best estimate approach for calculating the safety margin gives more operational latitude to the plant operator than using a conservative approach with prescribed arbitrary models.

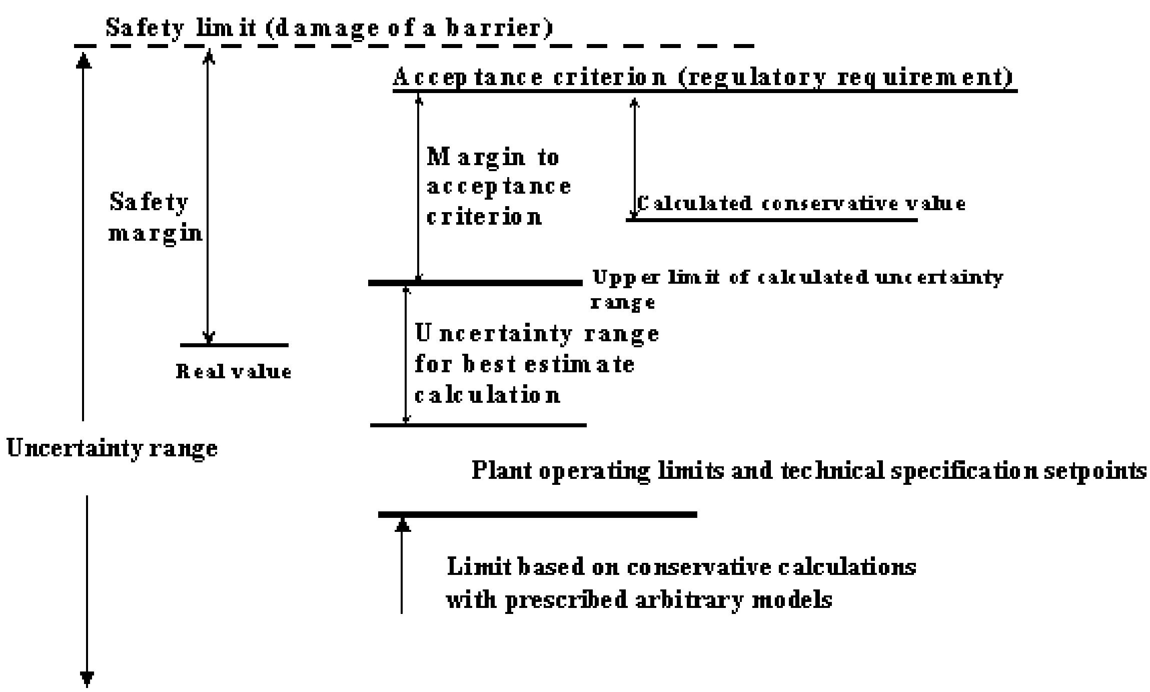

Figure 1. Stacked uncertainties in a conservative analysis.

1.3 The Role of Legacy Tools

The vendors often use tools that were developed for the early HTGR program to calculate the plant operational and accident behavior for licensing purposes. This *legacy* software relies mostly on low-order, low-dimension, approximate methods that would run efficiently on the computers of the day. For example, the core temperature distribution for a prismatic reactor would be generated using a rigorous 1-D calculation of the bulk temperatures that does not have the capability to calculate localized hot spots or cross flow between blocks. Similarly, the few-group cross sections used in the core simulation are generated at the compact (pincell) level and homogenized for use in block lattice calculations. The calculations do not account for the interpenetration of spectra from the surrounding fuel and reflector blocks, nor can they accurately treat large asymmetrically-placed control rods and burnable poisons. To allow room for the uncertainty in the power and temperature fields, prescribed safety factors are used to ensure that local material temperatures do not exceed material property limits. Consequently, since the prescribed safety factors account for the large uncertainties inherent to the use of legacy tools to calculate the localized core power distributions and maximum outlet jet temperatures, the NGNP will have to operate at a derated power conditions or with larger design margins as a function of the magnitude of the prescribed safety factor. These tools still have considerable value in design and scoping calculations; however, greater accuracy and precision can be achieved with modern lattice, nodal diffusion, and multidimensional thermal fluid and CFD tools that can capture the physics with far less loss of fidelity and accuracy.

1.4 The Role of Multiphysics and Higher Order/High Fidelity Simulations

Advances in computing power and algorithms for efficiently solving large systems of partial differential equations have enabled the development of high resolution multiscale, multiphysics analysis tools. Whereas 1-D or 2-D neutron transport is used in legacy evaluation models for generating local reaction rates and cross sections, 3-D discrete ordinates or Monte Carlo transport calculations of entire cores is now possible in some cases, particularly for confirming the ability of lower order tools to capture the important core behavior. High fidelity neutron transport and CFD calculations can be used to homogenize complex structures and localized phenomena to generate coarse mesh parameters that can be solved quickly with core simulation codes with a minimal loss of accuracy.

Multiphysics codes can be used to investigate phenomena too complex for system codes such as power and temperature peaking near the pebble bed reflector interface. The large change in porosity of the bed near the wall enhances cooling, which is counteracted by the additional thermalization of neutrons caused by the reflector. The conduction of heat from the pebble bed into the reflector is not described well by the correlations used in systems codes. Since the highest power densities, and thus the highest fuel stresses occur near this boundary, it is desirable to be able to model this region with full neutronic and thermal fluid feedback. Current thermal fluid codes do not address the change in porosity. The anisotropic scattering is not captured fully by neutron diffusion codes. A fully coupled CFD-transport calculation with an explicit representation of pebbles near the boundary can capture the physics and simulate power peaking and heat transfer far better than system codes.

Similarly, system codes do not explicitly model the change in the dimensions of graphite blocks under irradiation. This deformation is a complex function of irradiation and temperature history and thus the bypass flow is constantly changing over the life of the core. In modeling the core with system codes, a certain fraction and distribution of bypass flow is usually assumed based on empirical knowledge. A full coupled neutronic-CFD-mechanical model of the core can be used to simulate and predict bypass flow and reduce the uncertainty in core temperature analyses.

1.5 Differences between the HTGR Concepts that Must Be Addressed in Methods

As mentioned previously, the prismatic and pebble bed versions of the HTGR possess the same general functionality and bulk design. TRISO particles are embedded in a graphite matrix to form fuel elements that occupy a tall annular or cylindrical region inside the vessel and are surrounded by graphite reflector blocks. Helium is circulated through cooling pathways in the core to carry off the fission energy and be converted into the desired energy product. Reactivity control is affected by either helium inventory variations or with control rods inserted into the side reflector. The material, neutronic, and thermal fluid behavior of the concepts are largely the same and thus can be addressed with the same general modeling approach. There are, however, key differences in the fuel geometry requiring different heat transfer correlations, fuel management techniques, and other modeling assumptions that prohibit a prismatic core simulator from being used to model a pebble bed core and vice versa. Thus the NGNP Methods program has developed two simulation code capabilities in parallel but with largely the same functionality and underlying physics. The differences are described here.

1.5.1 Prismatic HTGR

In the prismatic reactor, cylindrical fuel *compacts* are stacked firmly inside channels drilled into graphite blocks in the shape of right hexagons (blocks). Surrounding the fuel channels are a number of open channels through which the helium coolant is directed. These fuel blocks are stacked firmly against each other in columns (red region of the left core in Figure 2). The columns form an annulus between an inner reflector and an outer reflector, both of which consist of rings of unfueled graphite blocks. The fuel blocks are loaded and reshuffled at regular intervals (18 to 24 months), much like a LWR. Because prismatic reactors are batch-loaded, additional fuel must be loaded into the blocks to maintain criticality through the cycle. This results in excess reactivity at the beginning of the cycle that has to be held down by burnable poisons or control rod insertion. Burnable poisons are also used to flatten the power profile and achieve a more uniform fuel burnup and load on the fuel particles. Neutronic codes are geared toward solving the transport or diffusion equation in hexagonal geometry with special accommodation for cylindrical burnable poison pins and control rods. Cylindrical coolant channels are drilled vertically through the blocks and between the columns of compacts. Most of the coolant flows through these engineered coolant passages. They are not connected except at the upper and lower plena. In the simplest thermal fluid analysis, the coolant can be modeled as 1-D flow through a set of independent pipes. By assuming no bypass flow (flow between the blocks). This can yield a first order map of temperatures within the core for fuel management studies and transient analysis. More sophisticated thermal fluid codes must model the radial and axial flow between the blocks to get a firmer picture of the temperature distribution within the fuel and core.

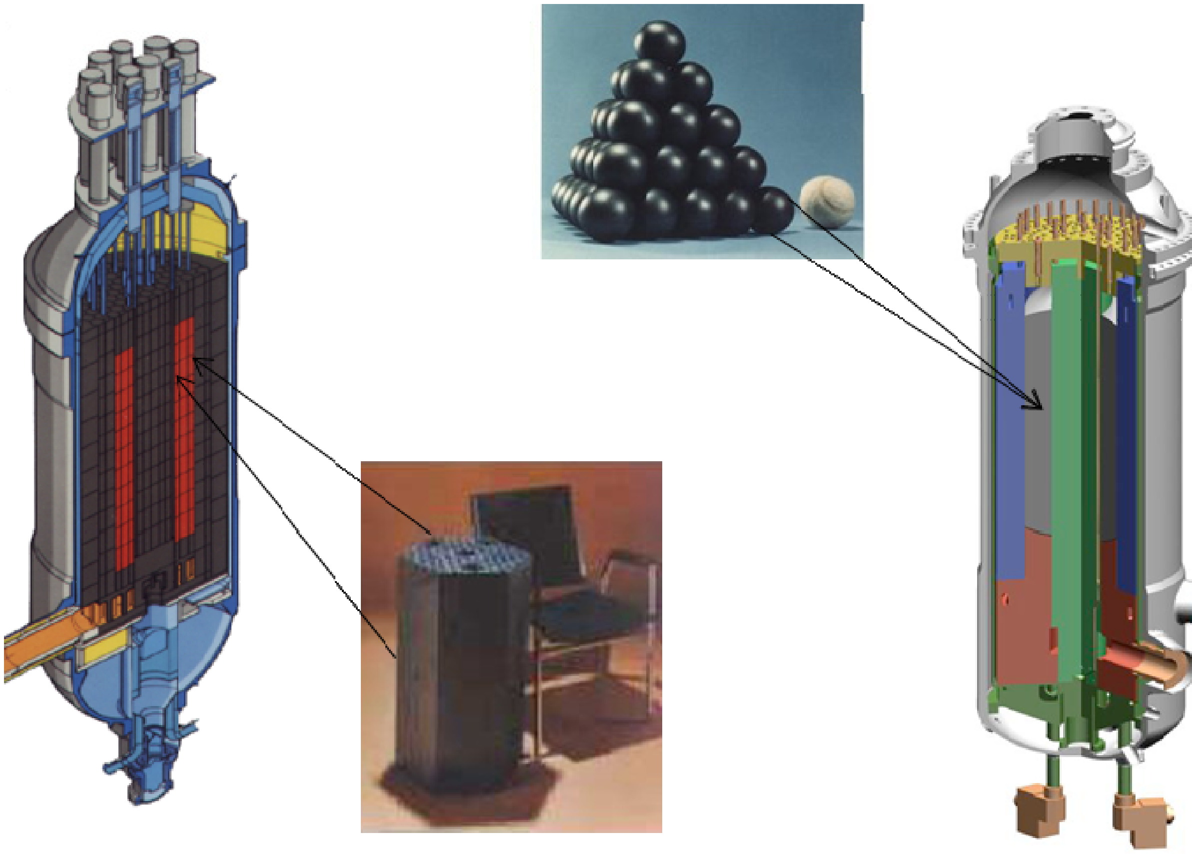

Figure 2. Prismatic block and reactor (left) and a pebble bed reactor (right)

1.5.2 Pebble Bed Reactor

In the PBR, the core consists of a bed of randomly ordered fuel pebbles. A pebble consists of TRISO particles embedded in the central fuel region (usually 5.0 cm in diameter) and surrounded by a 0.5 cm thick pure graphite shell. Helium coolant is blown through the interstitial void that makes up about 39% of the pebble bed volume. The PBR has the unique characteristic of continuous fueling and pebble movement through the core during operation. Fresh pebbles are dropped onto the top of the bed, trickle down through the core region and out through one or more discharge chutes. After measuring its burnup, the pebble is either reloaded for another pass through the core or discharged to a spent fuel storage container. Pebbles pass a number of times through the core before being discharged. Neutronic codes solve the transport or diffusion equations in spherical (pebble) or cylindrical (core) geometry. Burnable poisons are not required as the online refueling eliminates the need for excess reactivity. Fuel management is a unique challenge because of the semicontinuous movement and reloading of the fuel. Two- or three-dimensional thermal fluid codes model the bed as a porous medium with known pressure drop and heat transfer correlations for pebble beds. Coolant flow can be in any direction, but analyses indicate that the axial flow assumption is a reasonable to first order. PBRs do suffer experience bypass flow when the coolant is redirected through channels in or between the reflector blocks rather than through the pebble bed itself. High fidelity thermal fluid analyses must account for this phenomenon.

2. QUALIFICATION OF CODES AND MODELS

2.1 Analysis Methods Research, Development, and Qualification Process

To ensure that the analysis software is capable of calculating the NGNP plant behavior for the scenarios of interest, a rigorous approach is required that starts from a definition of needs and concludes with a demonstration of the capabilities of the software to fulfill those needs. Such an approach has been adopted by the NGNP Project. Once the software is demonstrated to be adequate for performing representative plant behavior calculations, plant behavior analyses are initiated.

The process of identifying R&D needs and then formulating plans is straightforward, although there are many unknowns and the process itself is iterative. In essence it is a five stage process that consists of (1) identifying the scenarios of importance, (2) identifying the key phenomena for the scenarios of importance, (3) determining whether the tools to be used to analyze the scenario progressions are adequate, (4) correcting or completing existing software and carrying out any software development that may be needed to ensure that the analysis tools are adequate, and finally, (5) performing the required analyses. This process is illustrated in Figure 3.

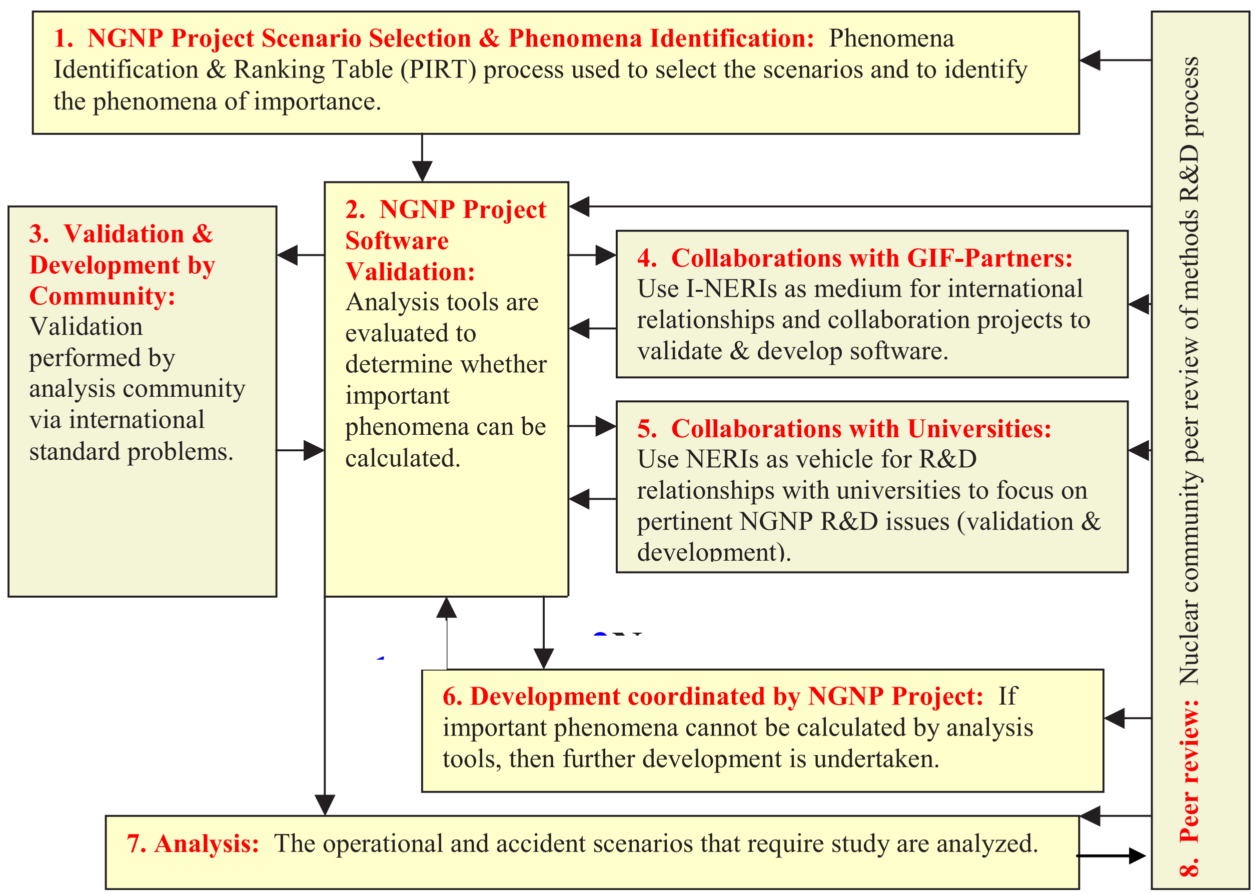

Figure 3. Methods R&D process.

The Methods R&D process consists of eight interacting activities or steps:

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

The first iteration on Step 1 has been completed and the results are discussed in Section 2.3 as they are specific to the NGNP. The PIRT process will be repeated at major steps in the design and licensing phases as knowledge and maturity of the design increases. Standard practices for the selection, validation, and verification of software have been adopted for the NGNP Methods Program. As these are not specific to the NGNP, they are described in detail in Appendix A.

2.2 Qualification of Evaluation Models – Regulatory Guide 1.203

The U.S. Nuclear Regulatory Commission (NRC) describes a process, in Regulatory Guide 1.203 [NRC 2005], they consider acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. In general the Evaluation Model Development and Assessment Process (EMDAP), described in Regulatory Guide 1.203, consists of: (1) determining the requirements for the evaluation model,[b] (2) developing an assessment base[c] consistent with the determined requirements, (3) developing the evaluation model, (4) assessing the adequacy of the evaluation model, (5) following an appropriate quality assurance (QA) protocol during the EMDAP, and (6) providing comprehensive, accurate, up-to-date information.

Although a specific NGNP design has not been selected, the NGNP Methods development effort has proceeded by examining and postulating the evaluation model requirements in conjunction with making a preliminary formulation of the required assessment base [Lee, Wei, and Schultz et al 2005], i.e., Steps 1 and 2. Because the NGNP will likely be either a prismatic or a pebble-bed type gas-cooled thermal reactor with known general characteristics, the various steady-state and transient characteristics are known in general. The assessment base (benchmark experiments) cannot be defined and selected until the final design selection because many of the thermal-fluids experiments are very geometry specific and very dependent on initial conditions that would reflect initial operating and accident conditions. The most probable HTGR design basis scenarios are described in Chapter 3.

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b. An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design-basis accident.

c. That is, either certifying existing experimental data as being adequate or designing physical experiments that will provide high-quality, acceptable data.

The evaluation models have been selected (Step 3) as will be noted in Chapter 3. A different suite of methods software is required to calculate the reactor physics behavior for the prismatic as opposed to the pebble-bed gas-cooled reactor. However, the software used to calculate the thermal-fluids behavior is the same for both reactor types.

In essence, much of this plan deals with defining the appropriate experiments to enable methods software validation to meet Regulatory Guide 1.203 requirements and the practices and procedures that must be developed and used to ensure the evaluation models are deemed adequate. Thus, much of this plan addresses Steps 2 and 4 of the EMDAP; discussion on these topics is given in Sections 4, 5, and 6.

In summary, the NGNP Methods development program's R&D are being planned and executed in conformance with the approach, practices, and methodologies recommended in Regulatory Guide 1.203.

2.3 HTGR Phenomena Identification

To show that the NGNP meets all safety requirements, proven analysis capability must be available to model not only the normal operational conditions, but also the accident conditions. Also, various aspects of the core behavior must be modeled, including:

1. Operational characteristics of the tri-structural isotorpic (TRISO) fuel throughout the NGNP's life cycle, e.g. the: the fuel temperature profile, migratory characteristics of the fuel kernel within the fuel micro-sphere, shrinkage and swelling of the various pyrolytic carbon coatings, and stress distributions in the coating layers. All of these operational characteristics are modeled numerically in the PARFUME software [Miller, Petti, Maki, and Knudson 2004; Petti, Hobbins, Kendall, and Saurwein 2005].
2. Fuel power distribution as a function of exposure in both the fuel compacts or balls and in the microspheres.
3. Thermal-fluid conditions during both operating conditions and transient conditions, including the fuel temperature profiles and the maximum temperatures of plant structural members such as the core barrel, core support plate, vessel wall, etc.
4. Mixing characteristics of the fluid inventory in the plena—the lower plenum during operating conditions since the hot exit gases are delivered to the turbine and both plena during a loss-of-forced-flow scenario.
5. Potential for air ingress, water ingress, and graphite oxidation subsequent to a loss-of-forced cooling (LOFC) events.
6. Fission product release and transport as a function of projected TRISO fuel failure rates.

The full spectrum of possible accident scenarios of importance is not fully defined, since it is dependent on the presently undefined NGNP design.[d] As a starting point, however, the following events postulated for Fort St. Vrain and the German Arbeitsgemeinschaft Versuchsreaktor Reactor (AVR), as indicated in the Fort St. Vrain Final Safety Analysis Report, can be used. The applicability of these to the candidate NGNP designs must be established.

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d. While the VHTR design is being formulated, the modular high temperature gas-cooled reactor (MHTGR) is serving as the reference reactor to define methods validation experiments for the prismatic design. A place-holder prismatic reactor design has not been specified to serve as the basis for defining methods validation experiments.

1. Anticipated operational occurrences:
   a. Main loop transient with forced core cooling
   b. Loss of main and shutdown cooling loops
   c. Accidental withdrawal of a group of control rods followed by reactor shutdown
   d. Small break LOCA (~1 in 2 area break).
2. Design basis accidents (assuming that only safety-related systems can be used for recovery):
   a. Loss of heat transport system and shutdown cooling system (similar to scenario 1b above)
   b. Loss of heat transport system without control rod trip
   c. Accidental withdrawal of a group of control rods followed by reactor shutdown
   d. Unintentional control rod withdrawal together with failure of heat transport systems and shutdown cooling system
   e. Transient without scram
   f. Earthquake-initiated trip of heat transport system
   g. LOCA event in conjunction with water ingress from failed shutdown cooling system
   h. Large break LOCA
   i. Small break LOCA.

On the basis of the prior experience of gas-cooled reactor designers and experimentalists (Ball 2003; Krüger et al. 1991), scenarios 2a and 2g (hereafter referred to as the pressurized conduction cooldown [PCC] scenario and the depressurized conduction cooldown (DCC) scenario, respectively) are considered expected to be the most demanding and most likely to lead to maximum vessel wall and fuel temperatures. Hence, first-cut R&D specifications are based on calculation of the hot-channel temperatures and mixing characteristics in the lower plenum during normal operation, and the PCC and the DCC scenarios from the accident envelope.

2.3.1 Phenomena Identification and Ranking Tables

The Phenomena Identification and Ranking Tables (PIRTs) process entails carefully identifying the most demanding scenarios, followed by prioritizing the phenomena that are found in the most demanding scenarios. Key phenomena are those exerting the most influence on the path taken during the most demanding scenarios. Thus, as discussed in the previous paragraphs, the key phenomena for the PCC and DCC scenarios, or most highly ranked phenomena, are those that exert the greatest influence on the peak core temperatures and peak vessel wall temperatures. During normal operation, other key phenomena such as stresses or irradiation-induced dimensional changes may be important.

Because the specific NGNP design has yet to be selected, a detailed PIRT cannot be completed. However, during the interim, first-cut PIRTs have been used instead as a guide for the initial R&D work and planning for both block-type and pebble-bed-type gas-cooled reactors. The first-cut PIRTs are based on observations from seasoned gas-cooled reactor experts and engineering judgment; these factors were used by a team assembled to define the first PIRT for the prismatic and pebble-bed reactors and described in Appendix A of Schultz et al. [2008] and is documented in detail in Lee, Wei, and Schultz et al. [2005] and Ball et al. [2008]. The results of the “first-cut” PIRTs for steady-state operation, PCC, and DCC scenarios are given in Table 1 for the upper and lower plena, the core, and the reactor cavity cooling system (RCCS). These scenarios and the experiments and modeling used in the investigation of them are described in Section 4.

Table 1. Results: PIRT for normal operation and conduction cooldown scenarios.

Scenario	Upper Plenum	Core	RCCS	Lower Plenum
Normal operation	i. Flow distribution ii. Pressure drop	i. Reactivity feedback behavior ii. Core configuration (bypass) iii. Pressure drop iv. Heat transfer v. Flow distribution vi. Power distribution	i Heat transfer at operational conditions ii Natural circulation in cavity	i Flow distribution ii Heat transfer iii Thermal striping iv Jet behavior
DCC	i. Mixing and stratification ii. Hot plumes iii. Thermal resistance of structures	i Thermal radiation and conduction of heat across the core ii Axial heat conduction and radiation iii Natural circulation in the reactor pressure vessel iv Air and water ingress v Potential fission product transport vi Power distribution vii Core configuration viii Decay heat ix Flow distribution x Material properties xi Pressure drop	i Laminar-turbulent transition flow ii Forced-natural mixed convection flow iii Heat transfer—radiation and convection in duct	i Thermal mixing and stratification ii Flow distribution iii Air ingress
PCC	i. Mixing and stratification ii. Hot plumes iii. Thermal resistance of structures	i Thermal radiation and conduction of heat across the core ii Axial heat conduction and radiation iii Natural circulation in the reactor pressure vessel iv Power distribution v Core configuration vi Decay heat vii Flow distribution viii Material properties ix Pressure drop	i Laminar-turbulent transition flow ii Forced-natural mixed convection flow iii Heat transfer—radiation and convection in duct	i Thermal mixing and stratification ii Flow distribution

3. GOALS, ASSUMPTIONS, AND REQUIREMENTS

An overall set of programmatic goals, assumptions, and requirements developed to guide preparation of this Technical Program Plan (TPP) is presented here. In the preparation, the scope was subdivided into two major program elements:

1. Experimental Validation of CFD and System Codes
2. Core and Plant Simulation.

Detailed goals, assumptions, and requirements were developed to guide the planning of these program elements. A high-level set of goals, assumptions, and requirements from the perspective of the overall program are identified in the following sections.

3.1 Overall Methods Program Goals

Overall Methods Program goals are as follows:

- Define the calculational envelope required to be able to analyze the candidate HTGR reactor systems.
- Define an NGNP evaluation model that should be capable of performing all the required calculations encompassed by the calculational envelope developed in the above bullet. This evaluation model shall provide reference results against which licensee and regulator simulation results can be compared.
- Design and execute a matrix of experiments that will produce a comprehensive data set that can be used to validate and verify NGNP evaluation models developed by the Department of Energy (DOE), NRC, and vendors.
- Define and qualify the components of the NGNP evaluation model using an approach that is in conformance with the NRC's Regulatory Guide 1.203.
- Support near-term deployment of the NGNP for process heat and electricity production in the United States (2021) by reducing market entry risks posed by technical uncertainties associated with thermal fluid and neutronic phenomena
- Develop uncertainty and sensitivity analysis capability that can be used to identify and prioritize gaps in the ability of an evaluation model to compute safety and performance parameters within confidence intervals.
- Utilize international collaboration mechanisms to extend the value of DOE resources (e.g., GIF VHTR activities).

3.2 Overall Assumptions

Overall Methods Program assumptions are as follows:

- Government and potential industry cosponsors of the NGNP recognize that a stable, long-term, disciplined methods development and code qualification effort offers the greatest probability of success.
- Proposed NGNP designs will impose more demanding service conditions than the German High Temperature Reactor module and require codes that simulate plant scenarios and phenomena with greater accuracy and lower uncertainties than can be achieved with legacy tools.

- It is technically feasible to develop and qualify codes and models for licensing purposes at reasonable cost and on a schedule consistent with the proposed demonstration plant deployment schedule(s).
- Adequate DOE funding will be available to support the methods development and qualification activities outlined in this TPP.
- Computational Fluid Dynamics models of thermal fluid phenomena can simulate complex flow and heat transfer phenomena with a level of fidelity and reproducibility that is acceptable to a regulator.
- Nodal diffusion, depletion, and assembly homogenization techniques can be developed that generate reaction and leakage rates in HTGR fuel with uncertainties comparable to state-of-the-art LWR simulation tools.
- Dedicated critical facilities for the NGNP will not be constructed; validation of neutronic codes will rely upon data from critical facilities that operated in the past and on existing HTGR engineering scale reactors (Japan High Temperature Test Reactor [HTTR] and Chinese HTR-10).
- Activities relating to the qualification of a reactor vendor's evaluation model by the NRC Office of Nuclear Reactor Regulation and meeting the NRC mandate of 10 CFR 50, Appendix B, QA and control are outside the scope of this program., As appropriate, however, software QA requirements will be applied to the codes and methods developed under NGNP.
- No major difficulties that could significantly impact the schedule are encountered during the experiment execution and development and validation of HTGR analysis tools.
- Experimental data and simulation results will be made available to all U.S.-based NGNP stakeholders for qualification of their own methods and tools.

3.3 Overall Requirements

Overall Methods Program requirements are as follows:

- Establish a HTGR methods development and qualification program that will
  - Address safety and performance issues identified in the NGNP PIRT as being both importance in terms of safety and having a high degree of uncertainty with regard to behavior and ability to model.
  - Produce standards for using CFDs in analyzing and licensing HTGRs.
  - Provide high quality (NQA-1) data to NGNP stakeholders for validating system and CFD codes.
  - Improve understanding of thermal fluid behavior in components and during both normal and off-normal scenarios.
  - Establish a capability to perform 3D simulations of core burnup and transients in both pebble bed and prismatic reactors.
  - Design, or inform the design of, thermal fluid experiments that are scaled properly to achieve conditions anticipated in the NGNP.
  - Develop pertinent technical information that supplement the NGNP reactor vendor's own licensing/qualification data in the topical report supporting NGNP licensing.
- Implement this plan such that it supports both prismatic and pebble-type fuel designs. The effort dedicated to each design will be proportionate with its associated level of industry interest and commitment. Current stages of the program should support both designs by concentrating on integral, separate effects, fundamental tests, and lattice physics improvements common to both designs.

- Implement this TPP in accordance with the DOE QA requirements specified in 10 CFR 830, "Nuclear Safety Management," Subpart A, "Quality Assurance Requirements" and in DOE Order 414.1B, "Quality Assurance." All activities that have direct input to the irradiation test specimen fabrication and irradiation campaigns will be conducted in accordance with national consensus standard NQA-1-2000, "Quality Assurance Requirements for Nuclear Facility Applications," published by the American Society of Mechanical Engineers. Each participating organization shall prepare specific QA plans for its assigned scope of work and may prepare additional project-specific plans for individual work breakdown structure elements as appropriate.

4. PROGRAM ELEMENTS

4.1 Experimental Validation of CFD and System Codes

4.1.1 Goals, Assumptions, and Requirements

Goals for validating CFD and system codes are as follows:

- Demonstrate that thermal-fluids analysis software are capable of calculating the HTGR plant behavior for typical operational conditions and for those accident scenarios Identified as necessary:
  - In the PIRT reviews.
  - Using the approach outlined in the NRC Regulatory Guide 1.203.
- Create a spectrum of standard problems[e] centered on the NGNP thermal-fluids experiments, that will form the backbone of the HTGR thermal-fluids analysis software qualification/certification for both DOE and Generation IV.
- Define the validation matrix to provide the basis for validating the thermal-fluids analysis software using the spectrum of standard problems that will make up the NGNP experiment V&V matrix.
- Prescribe the methodology for qualifying thermal-fluids software (both systems analysis and CFDs) for performing HTGR plant behavior analyses that meet the standards required by NRC licensing.
- Demonstrate that the methodology for qualifying thermal-fluids software is feasible and conforms to NRC recommended guidelines.
- Develop and use an uncertainty quantification method to evaluate the calculational uncertainties of the above plant behavior calculations.

Assumptions for validating CFD and system codes are as follows:

- The validation approach most likely to be acceptable to NRC is outlined in Regulatory Guide 1.203. A working example of this approach is the methodology developed at Idaho National Laboratory (INL) to qualify the thermal-hydraulic analysis software for the NRC Westinghouse AP600 auditing effort.
- The validation approach and the approach for accepting software outlined in Regulatory Guide 1.203 will be used to accept the NGNP thermal-fluids software prior to performing NGNP-certified calculations that describe the HTGR plant behavior for operational and accident conditions.
- The phenomena identification and ranking tables generated to identify the most challenging scenarios, the figures-of-merit, and the dominant phenomena for each phase of each scenario will be periodically updated as the NGNP design is first identified and then matures.
- A standard for developing and using a validation matrix as a tool to demonstrate that the software is acceptable, will be developed, and will be used as the means for demonstrating the capabilities of the thermal-fluids software that is part of the NGNP evaluation model. The standard will be developed within the ASME V&V30 Standard Committee.

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e. A standard problem is an experimental data set that has undergone a rigorous review to ensure the data have achieved the desired quality level, have acceptable measurement uncertainties, are demonstrated to be scalable to the reference reactor system, and has been distilled to a useable format to enable comparison to software validation calculations.

- All experiments used to generate data will be scaled to the reference HTGR reactor system using the acceptable scaling approach [Zuber 1991].
- A set of standard practices and procedures for performing validation calculations for both systems analysis and CFD will be demonstrated and approved. The NGNP thermal-fluids software will be validated using the approved standard practices and procedures before NGNP-generated HTGR plant calculations for operational and accident conditions will be completed.

Requirements for validating CFD and system codes are as follows:

- The experimental data identified from legacy data source and generated in the NGNP experimental V&V program will meet the NQA-1 requirements necessary for the data to be used as a validation reference to validate software that may be used to perform licensing calculations for submittals in licensing documents to the NRC.
- The validation matrix developed in the NGNP experimental V&V program will be comprehensive enough that the software calculational envelope can be shown to encompass the NGNP operational and accident envelopes using the required standard.
- The scaling studies performed to design the experiments used to generate data for the thermal-fluids validation matrix will be accomplished using the approach defined in the Hierarchical, Two-Tiered Scaling Methodology [Zuber 1991].
- The validation matrix and NGNP calculational matrix will be updated periodically to reflect both updates in the reference reactor design and the iterations performed to ensure the PIRT evaluations are consistent with the updated reference reactor design.
- NGNP-approved calculations of the HTGR behavior, with a reasonable expectation of predicting the HTGR behavior for challenging operational and accident conditions, will not be released until the calculational requirements imposed by Regulatory Guide 1.203 have been satisfied using the validation matrix; standard practices and procedures have been executed to qualify the NGNP thermal-fluids analysis software.

4.1.2 Scope

4.1.2.1 Bypass Flow

A series of experiments is envisioned that will test the various theories regarding factors that influence the quantity of bypass (in either the prismatic or PBR) as a function of various factors, including manufacturing tolerances and core configuration changes from irradiation or thermal expansion. It is envisioned that the work scope will be a DOE laboratory-university partnership.

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

The resolution for this phenomenological issue will likely be a statistical approach similar to that used for the classical hot spot/hot channel factors. A high-level stochastic structure involving a combination of materials modeling, both first-principles and correlations, thermal-hydraulics R&D, and manufacturing practice will need to be put in place early. This will guide the R&D. It is anticipated that researchers will investigate the various factors that influence the bypass and develop preliminary models.

For the case of the prismatic reactor, small-scale experiments encompassing both thermal-hydraulics and materials phenomena will be performed (some at INL and some at universities). To fully resolve and characterize the prismatic bypass flow at least three R&D stages are needed as given below.

***Stage 1*:** Characterize isothermal bypass flow and its relationship to the flow in the core coolant channels in the following steps:

a. Characterize flow in the bypass slots as a function of slot width.
b. Characterize slot flow relative to the central vertex that connects the adjacent three slots.
c. Characterize slot flow as a function of local distortion resulting from expansion and/or contraction of block surface.
d. Characterize radial cross flow (leakage) as a function of expansion and/or contraction of block surface in conjunction with potential seal failures.

***Stage 2*:** Characterize heated bypass flow and its relationship to the flow in the core coolant channels in the following steps:

a. Build on the isothermal experiments to study effects of heating on flow in the slots.
b. Conduct supporting separate-effects (fundamental physics) and integral-effects experiments.

***Stage 3*:** Characterize the relationship between the changes in the bypass flow as a function of irradiation, graphite type, and block configuration considering:

a. Graphite material thermal effects such as thermal conductivity, capacitance, and thermal expansion—all as function of direction and temperature.
b. Graphite structural deformation for selected graphite types as a function of irradiation and fuel loading history.
c. Prismatic block seal deformation at upper and lower end of prismatic blocks as function of irradiation, fuel loading history, and structural loading.

Stages 1, 2, and 3 are presently underway.

The R&D effort for the PBR will be planned to complement available data already recorded during extensive experimental programs at the Pebble Bed Modular Reactor (PBMR) Pty. Assuming the PBMR Pty data will be made available through the Generation IV International Forum via the Computational Methods Validation and Benchmark Project Management Board, supplementary experiments will be planned to complement the PBMR Pty data. It is envisioned that the experiments may be performed at both INL and universities.

4.1.2.2 Air Ingress and Graphite Oxidation

Air Ingress

Subsequent to a leak in the hot duct or at other locations on the reactor vessel such as the control rod drive flanges, the reactor vessel is postulated to depressurize. The rate of depressurization is key because for larger leak sizes, where the leak area has the potential to allow a significant quantity of air to enter the vessel, the pressurization rate of the confinement will likely trigger the removal of blow-out panels in the confinement that will open passages from the confinement to adjacent rooms. A significant leak size will likely pressurize the confinement volume enough to push the air from the confinement to the adjacent rooms so the confinement will be largely filled with helium. For smaller leaks, where air will remain in the confinement for some time, it may be assumed the entry rate of air into the vessel will be low and thus of potentially less concern.

R&D for the scenarios summarized above is ongoing. Plans are to study (a) stratified flow of air into the reactor vessel via the leakage site, (b) stratified flow of air into the confinement via blowout passage ports, (c) the rate of mixing between air and helium in the confinement via as a function of leak size and the influence of these, and (d) secondary factors on the rate of air movement into the reactor vessel.

Isothermal air ingress experiments are presently under way. Heated air ingress experiments will be performed using the High Temperature Test Facility (HTTF) as discussed in Section 4.1.2.12.

To provide data required to accurately treat heat transfer and wall friction of air-helium mixtures during air ingress events, isothermal experiments will be ongoing through FY 2011 and hopefully some university research will be conducted by universities through NEUP. Air ingress experiments will be performed in the HTTF from 2012 for at least one year.

Graphite oxidation experiments (in both the kinetic and diffusion-limited regimes) are being performed as part of the NGNP Graphite Characterization program. The data from these experiments will be used in multiphysics air ingress simulations using the GAMMA code.

Water Ingress

Water is normally present in the air in the form of humidity, but if the shutdown cooling system suffers a pipe break, it may enter the core in greater quantities, thereby having greater potential for effecting reactivity. There is presently a high probability that the reference HTGR design will be based on a Rankine cycle with the secondary side designed to operate at a considerably higher pressure than the primary side. It is anticipated that this design will use a set of steam generators where the secondary side is segregated from the primary side by only the wall of a number of steam generator tubes designed to operate with a defined operational differential pressure across them.

For such a reference reactor design, the NGNP will be vulnerable to a new set of scenarios called steam-generator tube rupture (SGTR) events, for which it is assumed that occasional off-normal events and operational occurrences may cause a single tube to rupture. The single tube, as it undergoes depressurization, may be responsible, via pipe whip, for producing ruptures in adjacent tubes. Hence a set of steam generator tubes may rupture almost simultaneously and lead to a depressurization of the water-filled secondary system into the helium-filled primary system.

Water ingress was not identified in the original PIRT to be a high frequency scenario, but the shift to water as a secondary loop working fluid poses a substantially higher risk of water ingress. To investigate the SGTR scenarios, a new PIRT study will be performed following confirmation that the desired reference reactor design includes a high-pressure, water-filled secondary system. From the PIRT results, a set of experiments will be designed that are scaled to represent the reference design and new points will be inserted in the thermal-fluids validation matrix.

4.1.2.3 Core Heat Transfer and Plenum-to-Plenum

The characteristics of the hottest cooling channels at operational conditions are considered a key calculational result, since the hot channel temperature distribution defines the hottest initial condition for the fuel and surrounding materials. Hence, preliminary neutronics and CFD studies have been initiated and validation data are sought from core heat transfer experiments. Under low-flow conditions such as can occur after the primary blowers have tripped, conduction, radiation, and buoyancy-driven flow become the dominant heat transfer mechanisms. This flow can be a mixture of turbulent and laminar flow and thus may be subject to considerable instability. Codes used to analyze this scenario may not capture the heat transfer mechanisms with correlations that are valid in the turbulent or laminar regimes. Also, CFD codes presently do not have a well-developed capability to distinguish whether flow is laminar, in transition, or turbulent. Simulations in the transition flow range are challenging, yet important because the ability of the core to reject heat under these circumstances is tied closely to peak temperatures and the stress on the fuel.

Prismatic HTGR

The characteristics of the hottest cooling channels at operational conditions are considered a key calculational result because the hot channel temperature distribution defines the hottest initial condition for the fuel and surrounding materials. Hence, preliminary neutronics and CFD studies have been initiated and validation data are sought from core heat transfer experiments.

Such experiments will provide documented temperature, velocity, and turbulence field data for forced and mixed convection (buoyancy effects) and gas property variation in HTGR block-type reactor cooling channels in order to validate the turbulence models at reactor conditions for which benchmark data are not available. The proper calculation of turbulence directly influences cooling flow and temperature. Instrumentation will include miniaturized multisensor hot-wire probes developed as a task in a recent Nuclear Energy Research Initiative project for gas-cooled reactors [McEligot et al. 2002]. Both down-flow (normal operation) and up-flow (pressurized cooldown) will be considered.

The distribution of the flow between the various coolant channels in a prismatic reactor (and the complementary behavior in both the upper and lower plena) is important in determining the warmest part of the core and also the location of potential hot spots in the plena. It is necessary to know the flow distribution for both operational and transient conditions.

During normal operation, the core power profile can produce an array of jets that may have large differences in temperature moving into the lower plenum. Potentially large temperature distributions may lead to hot spots or “hot streaks” in the lower plenum structures that consist of particularly hot jets that do not mix well with the surrounding gases prior to reaching downstream components such as intermediate heat exchangers or turbine blades.

During transients initiated by a trip of the helium recirculator, natural convection patterns are driven by the temperature distributions in the core and the plena. These are characterized by upward flow through part of the core and downward flow through other core regions. Hence, hot plumes will emerge from the upward flowing core cooling channels and impinge on the upper plenum ceiling and the control rod hardware located there. Unacceptably large thermal gradients may result in premature structural failure.

An experiment designed to investigate core heat transfer is shown in Figure 4. The experiment will support the efforts of the current computational task concerning the hot channel issue by providing benchmark data for detailed assessment of its turbulence models for forced and mixed convection with helium property variation. Miniature multiple-sensor hot-wire probes will be inserted through the open exit as shown in Figure 5 to obtain point-wise temperature and velocity measurements. The objectives are to measure the fundamental turbulence structure and obtain benchmark data to assess CFD codes for high temperature gas flows that are in the forced and mixed convection regions, for a range of conditions important in HTGRs. To achieve these objectives the experiment will provide an approximately uniform wall heat flux boundary condition in a tube for helium, either ascending or descending and entering with a fully-developed turbulent velocity profile at a uniform temperature as in coolant channels after passing through an end reflector. Potential apparatus to obtain benchmark turbulence data in heated channels.

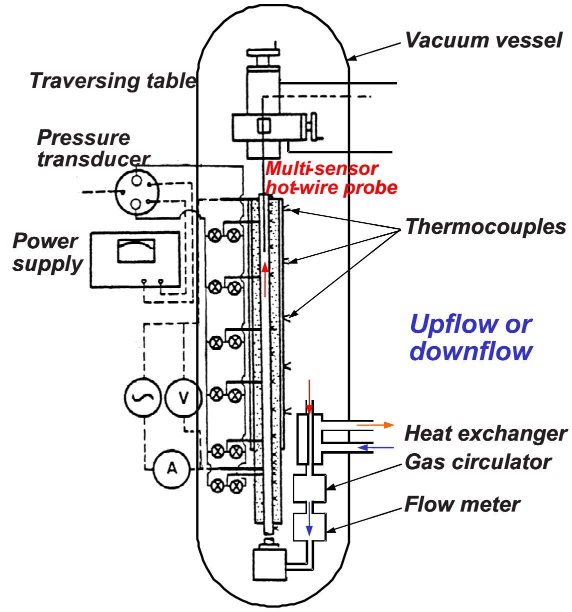


Figure 4. Potential apparatus to obtain benchmark turbulence data in heated channels.

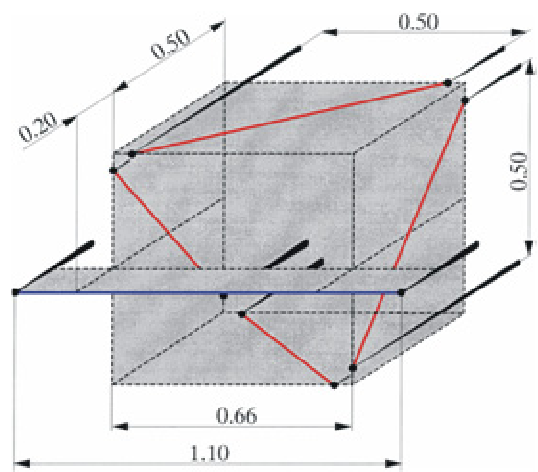


Figure 5. Schematic diagram of miniature five-sensor probe by Vukoslavcevic and Wallace [2003], the dimensions are in millimeters.

Pebble Bed HTGR

A matched-index-of-refraction (MIR) experiment is planned to examine flows near outlets in PBRs. A key difficulty in analyzing the safety of PBR systems is predicting the maximum fuel temperatures and chemical reaction rates locally in the coolant outlet region (e.g., hot spots) where the temperature field is generally high. 1-D system codes have been applied for transient safety analyses and parameter studies during preliminary design. A 1-D calculation predicts quantities that are averaged across the flow (e.g., the core diameter) and does not predict the highest temperatures or their locations. The THERMIX and TINTE codes developed for the early HTGR program in Germany use a 2-D porous medium model (Reitsma 2005). Further, since chemical reaction rates vary nonlinearly with temperature, the average reaction rate is not the reaction rate at an average temperature. While these systems codes are needed, it is desirable to supplement them with 3-D calculations for final designs and for estimating hot spot factors to improve their predictions. Potentially, 3-D CFD codes can be applied using a porous medium approximation to find the coolant velocity and temperature in localized macroscopic regions. Then direct numerical simulations can be used to identify the point-wise peak temperatures and their locations (microscopic treatment). The goal of this research is to develop accurate techniques for predicting maximum temperatures in HTGR concepts that use pebble bed technologies by coupling CFD calculations with experiments in the unique INL MIR flow system.

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

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

4.1.2.4 Upper and Lower Plenum

In typical prismatic HTGR concepts, the complicated transition from coolant channels to the lower plenum provides the inlet conditions for the jets into the lower plenum (see Figure 6); measurements of turbulence distributions and pressure drop (loss coefficients) are needed for CFD predictions and design. Depending on the reactor designs, comparable problems may appear for the upper plenum. Also, at the geometric transition from a lower plenum to its outlet duct, the convergence may cause laminarization of the turbulent flow, leading to reduced thermal mixing [McEligot and McCreery 2004]. Experiments on fluid dynamics of geometric transitions conducted from FY 2009 through FY 2011 will employ the INL MIR flow system (discussed separately under lower plenum fluid dynamics experiments) and gas flow experiments to address the key geometry problems identified in NGNP conceptual designs to that time, both for prismatic and pebble-bed approaches.

In addition, a series of scaled bench-top experiments may prove essential for validation of multi-dimensional and/or multiphysics predictive computational simulation tools for the evaluation of design specific local features specific to the core region such as insulation, baffles, and mixing plates, which improve the performance of the system. As candidate designs are developed, it is anticipated that a list of important local features that are likely to have a limiting effect on system performance will be developed for each design. At this stage, two such features that require simulation tool development and/or validation can be identified as important to all potential designs are (1) the prediction of isothermal flow-

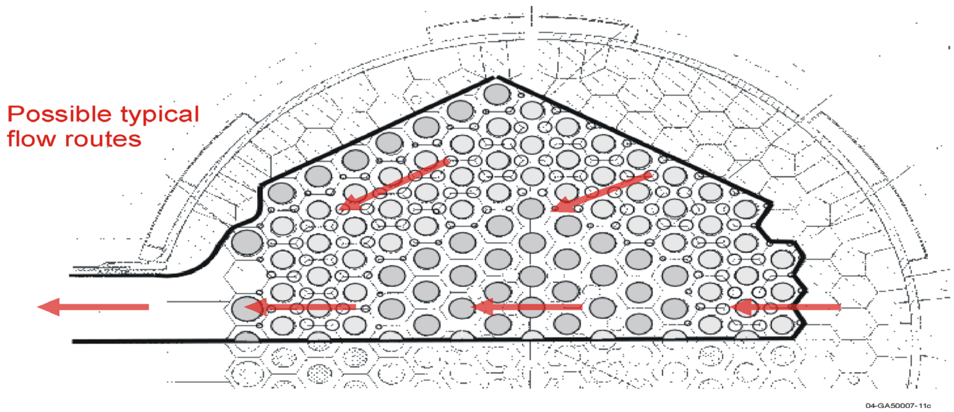


Figure 6. Examples of some possible flow paths in the lower plenum of a typical HTGR block reactor concept.

induced vibration of plate or sheet components (e.g., insulation or thermal radiation shielding), and (2) the prediction of flow-induced vibration and thermal distortion of plate or sheet components (e.g., insulation or thermal radiation shielding. In both cases, the wear and fatigue associated with these fluid-structure interactions may have significant implications on the expected life and probabilistic failure rates of these components, which impact the safety case for the system as a whole. Additional important local features may be identified as potential designs mature. Local structural effects because of large depressurization accidents could be such a possibility.

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

The mixing of hot plumes in the upper plenum of a gas-cooled reactor is of concern during a pressurized cooldown [McEligot et al. 2002]. These plumes come from up-flow in the hot coolant channels during natural circulation in the core and may impinge on the reactor vessel upper plenum structure and control rod apparatus causing localized hot spots that may be prone to failure. The flow rates and temperatures of the plumes may be affected by laminar flow instability caused by variations in the viscosity with temperature [Reshotko 1967] at the low Reynolds numbers in these channels and may possibly cause flow choking. An experiment is planned to investigate interactions between hot plumes and parallel flow instabilities.

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

4.1.2.5 Ex-Core Heat Transfer

Figure 7 shows a likely layout for the NGNP with the reactor pressure vessel and the vessel containing the intermediate heat exchanger and primary coolant system circulator sited below grade. During the PCC scenario in the core region and during both the PCC and DCC scenarios in the RCCS, there is the potential for having convective cooling in the transition region as shown in Figure 8, where an example is shown of convection flow regimes along the heater (reactor core) and cooler (heat exchanger providing ultimate heat sink) at various pressures in a simplified Reynolds-Rayleigh number map [Williams et al. 2003]. Although Figure 8 was generated for a typical gas fast reactor core having hexagonal blocks with circular coolant holes, analogous behavior may occur in the NGNP in various locations and should be investigated. Because the convective cooling contribution is an important ingredient in describing the total heat transfer from the core and thus the ultimate peak core and vessel temperatures, these heat transfer phenomena are potentially important.

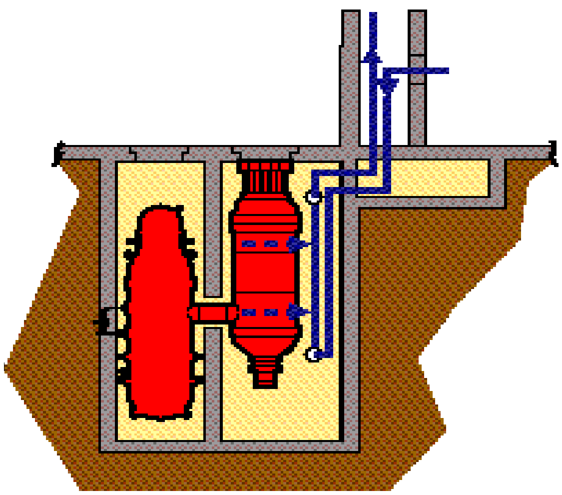


Figure 7. Reactor cavity cooling system configuration.

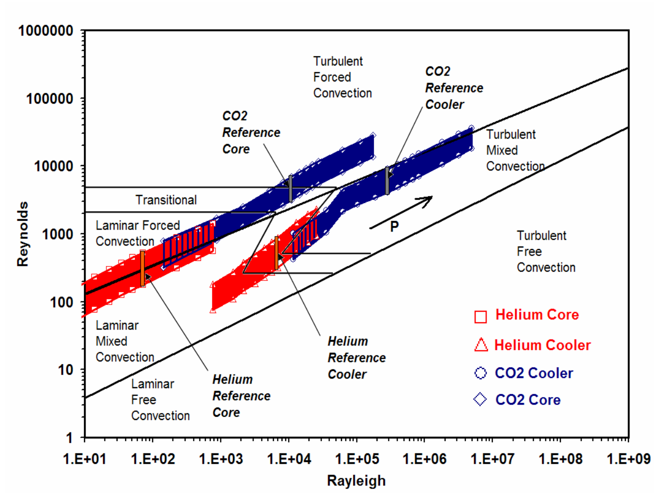


Figure 8. Convection flow regimes at various operating pressures for both helium and CO2 (from Williams, et al, 2003).

The objective of this task is therefore to acquire the model/code validation data for natural convection and radiation heat transfer in the reactor cavity and the RCCS by performing experiments in the Argonne National Laboratory (ANL) Natural Convection Shutdown Heat Removal Test Facility (NSTF). The NSTF (see schematic in Figure 9) will be used as a experiment simulator. The first step will be to determine scalability of the existing data and configuration of the ANL RCCS simulator to the RCCS designs. The scaling studies will identify the important nondimensional parameters for each separate-effects study for both air-cooled and water-cooled systems. Based on the results of the scaling/feasibility study, the range of experiment conditions will be determined as well as the appropriate experiment scale and appropriate fluids to be used that most effectively simulate full-scale system behavior. R&D will include the identification of RCCS design candidates from both the pebble-bed and prismatic options. The range of thermal-hydraulic conditions for normal operating and accident events will be evaluated. An instrumentation strategy will be developed to assure that adequately detailed velocity and turbulence profiles are obtained, as well as surface pressure and/or temperature distributions for the validation of multidimensional simulation tools. Based on the results of these scaling/feasibility studies and the RCCS analyses carried out in Section 6.10, a detailed engineering modification plan for the ANL RCCS facility will be developed.

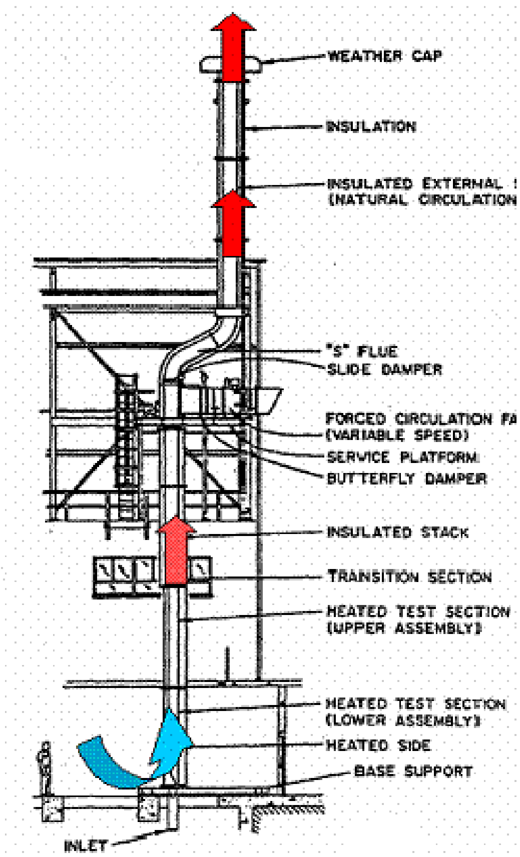


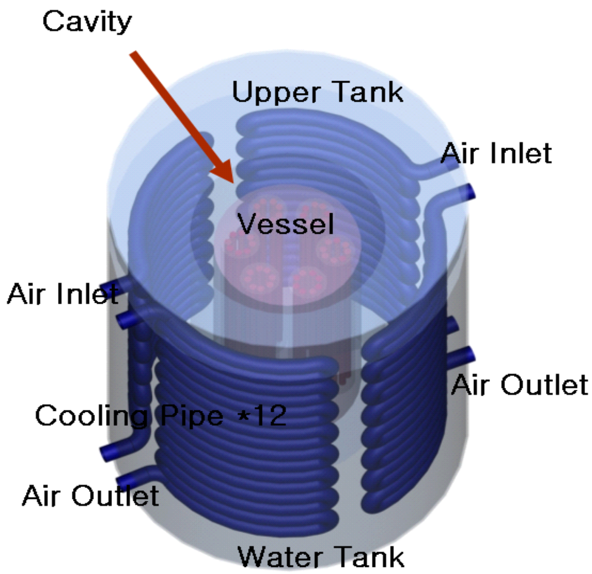
Figure 9. Schematic of ANL Natural Convection Shutdown Heat Removal Test Facility.

Next, a test matrix will be developed, and the indicated test program will be performed. The ANL RCCS experimental results will capture key phenomena expected to be present in the RCCS and provide data of sufficient resolution for development and assessment of applicable CFD (STAR-CD/Fluent) and system codes (RELAP5-3D/ATHENA). Both air and water-cooled RCCSs will be included in the NSTF test plan. The detailed work scope for the modification of the NTSF to perform experiments for the NGNP RCCS has been completed. Refurbishment of the NSTF will commence in FY 2010, with final construction in FY 2011.

A complementary experiment has been constructed at the Seoul National University RCCS facility in Korea. It consists of three parts as shown in Figure 10: the reactor vessel, an air cavity, and a water pool. The SNU experiments are being performed using various gas mixtures in the gap and with various water pool elevations. The temperatures on the various surfaces are measured together with the surface emissivities and water pool characteristics (temperature as a function of position, elevation, etc.). Heat from the reactor vessel is transferred to the RCCS by radiation, natural convection, and conduction. The data provided by these experiments are the basis for validation CFD calculations specific to the behavior of water-cooled RCCS.

In addition to the above experiment, it may be possible to perform depressurized conduction cooldown experiments in the NSTF for the prismatic block reactor to simulate the conduction and radiation of the decay heat away from the fuel in the core and transfer it to the primary vessel metal. In FY 2007, the existing NSTF facility will be evaluated at a high-level to determine if there is the possibility of also using this experimental facility for depressurized conduction cooldown tests for the prismatic block core.

Work accomplished from FY 2006 through FY 2010 included initiating preliminary design of the apparatus, cost and schedule estimation, complementary CFD modeling of the system, and initial reconfiguration of the NSTF. Final design, reevaluation of costs and schedules, and initiation of fabrication are planned for later years with equipment operation, measurements, and documentation occurring from FY 2011 to FY 2013.



a. Schematic of test facility

b. Photograph of test facility

Figure 10. SNU water-cooled RCCS experiment.

4.1.2.6 Fission Product and Dust Transport

The "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program" [Petti 2010] discusses the need for experimental facilities for fission product and source term experiments in Section 3.5.1. The AGR Fuel Qualification Program addresses the generation and transport of fission products out of the fuel and block matrix. The transport of fission products out of the primary system and into the reactor building will be addressed in Methods. Specific experiments are outlined for fission product transport in the vented low-pressure containment. The compartment and spaces in the reactor silo building are connected together to form a long and torturous vent path. During events involving primary coolant leakage into the silo and onto the building, natural processes will act to reduce the level of entrained radionuclides as the gas stream transits. Mechanistic radionuclide retention in the vented low-pressure containment is considered when showing compliance with the *Protection Action Guide* dose limits at the exclusion area boundary with source terms for core conduction cooldown accidents .Data are needed to develop and validate the methods describing the behavior of condensable radionuclides in the building under wet and dry conditions for these accidents. The reactor silo with the cavity and the RCCS piping forms a compartment with internals in this long vent path for the transit of gas. Once the NSTF is configured for the RCCS experiment, the structures and geometry for the condensation of the fission products will be available for an integral large-scale experimental simulation of fission product transport in this cavity and silo. A preliminary scoping evaluation will be the starting point to assess the feasibility of utilizing the NSTF to perform multifunctional integral experiments. Simulant fission product transport experiments will be the focus.

Other facilities and experiments have recently been proposed to provide data for fission product and dust migration through the reactor building. Argonne's ZPR Cell 5 (Figure 11) is a tightly sealed structure rated to take an explosion from 45 kg of TNT and 4 bar of overpressure. It has a nuclear qualified ventilation system that can be rigged to perform aerosol and dust dispersion experiments after a blow down. The building is currently an empty cavity (mostly), but partitions and compartments can be installed to simulate the geometry of an actual power plant.

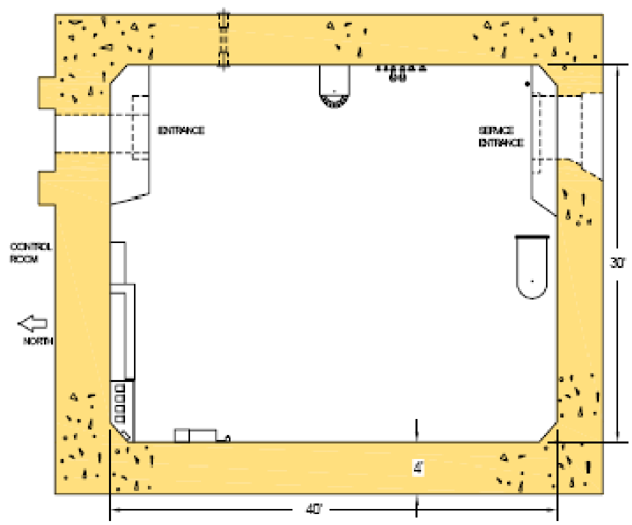


Figure 11. Aerial view of ANL ZPR Cell 5.

A consortium of European organizations (Areva, GRS, and Becker Technologies) is proposing to the Organization of Economic Cooperation and Development (OECD) the so-called THAI facility for investigation aerosol and dust transport in a generic HTGR building (Figure 12). Like ANL Cell 5, the experiments would support gas mixing phenomena and complex flow patterns within multiple compartments. A primary deliverable of this series of experiments would be fission product retention factors (amount released from the building/amount released from the core) which could be incorporated directly into a safety analysis or PRA. The work at these large facilities would be added to the separate effects experiments taking place at some universities under the NEUP Program (Tokuhiro 2009, Loyalka 2009).

The proposal is being considered by both the NGNP Project and the US NRC; however, it is necessary to conduct system and sensitivity analyses on the candidate NGNP design to determine what experimental data is desired and to guide the design of the experiment.

4.1.2.7 Reactor Physics

Integral benchmark experiment data for existing critical configurations that are neutronically similar to contemplated NGNP designs are required for physics code validation and quality assurance, both as part of the reactor design process as well as for licensing applications. Modern computational simulation techniques for reactor physics are capable of very high accuracy, and can in some cases replace significantly more costly mockups and critical experiments, but only if the accuracy of the simulation is carefully established by rigorous validation of physics codes against appropriate integral experiment physics data. In addition, mathematically rigorous sensitivity studies for representative HTGR core designs are required as an aid in guiding the design of any needed critical experiments that cannot be replaced by simulations (because experiments with sufficient similarity are determined to be unavailable), and perhaps most importantly, for quantification of the propagation of uncertainties in computational simulations because of uncertainties in the underlying nuclear data and other parameters that make up the input to the simulation models.

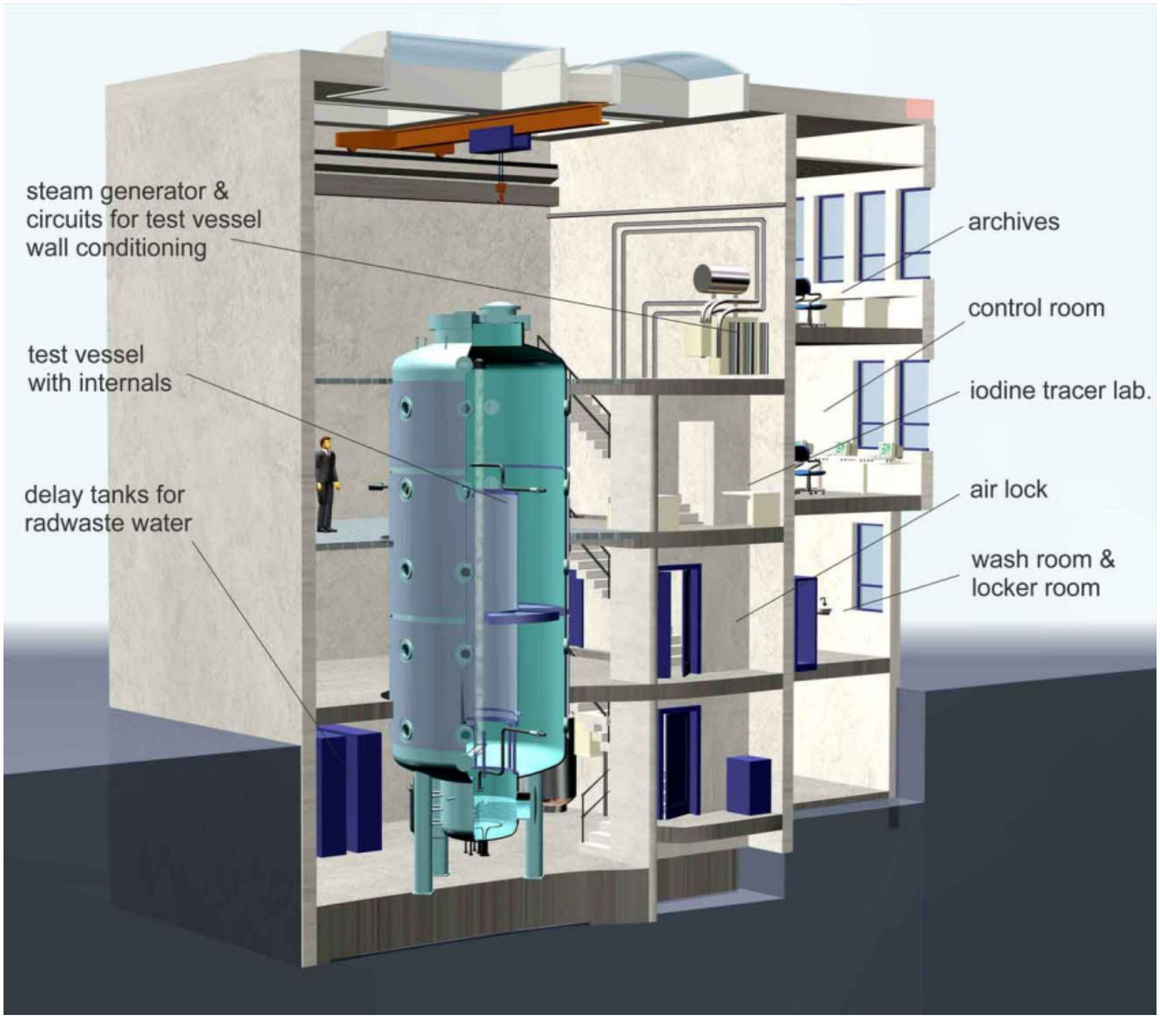


Figure 12. Rendition of the OECD-THAI Facility for investigating fission product and dust transport [Poss 2009].

Finally, high-accuracy differential nuclear data (nuclear cross section) libraries are required as input for all computational reactor physics tasks associated with NGNP design, licensing, and subsequent operation. Any simulation is only as accurate as the input data, and in the reactor physics field, the differential nuclear cross sections for the various materials used in the reactor constitute the most fundamental and crucial input information needed for the computational simulation process. For example, computational studies performed at INL show that for a reference prismatic HTGR fuel design, an uncertainty of as little as 10% in the Pu-240 capture cross section can lead to uncertainties in system reactivity of as much as 500 pcm absolute reactivity because of the propagated uncertainty in Pu-241 buildup. This is an indication of high sensitivity to this particular cross section. Furthermore, earlier integral experiment-based code validation studies performed and published by INL [Sterbentz, 2002 and Sterbentz and Wemple 1996] for low-enriched fuel with thermal or slightly hyperthermal neutron spectra representative of typical HTGR designs, show that computations of the inventories of the plutonium isotopes of interest here can vary by as much as 30% from corresponding measurements at burnups of less than one-third of what is contemplated in a baseline HTGR scenario. Such discrepancies can propagate in a manner that can have major effects on the uncertainty of computed safety-related reactor parameters such as reactivity, Doppler feedback, etc.

In FY 2004, ANL and INL examined information on several past and present experimental and prototypical facilities based on HTGR concepts that could potentially be used for the V&V basis of codes employed in the design and analysis of HTGR cores. A preliminary assessment of the applicability of the existing test data for benchmarking the pebble-bed and prismatic block-type cores was performed as part of that effort [Terry et al., 2004]. The experiments assessed included:

- *Pebble-Bed Type Cores*: ASTRA, AVR, CESAR II, GROG, HTR-10, HTR-PROTEUS, KAHTER, SAR, and THTR.
- *Prismatic Block-Type Cores*: CNPS, DRAGON, Fort St. Vrain, GGA HTGR Criticals, HITREX-1, HTLTR, HTTR, MARIUS-IV, Peach Bottom HTGR, Peach Bottom Criticals, SHE, NESTOR/HECTOR, and VHTRC.

Trends were observed in the experiments performed in the various facilities investigated. It was found that most of the experiments for block-type cores were performed in the United States, while those on pebble-bed cores were done predominantly in Europe. Most of the early U.S. experiments used highly enriched uranium. This was not typically the case for the European experiments. Additionally, experiments are currently being performed for both pebble-bed and block type cores in Asia (Japan and China) as well as in Russia. Under this NGNP program element, we will have there will be an opportunity to influence the direction of these experiments in a way that benefits the NGNP effort.

The 2004 assessment revealed that the HTGR systems under development in the Gen-IV program differ in significant ways from previous high-temperature reactors (e.g., thorium utilization, highly enriched fuel, BISO versus TRISO fuel, thermal efficiency, operating temperatures, etc). These differences limit the applicability and direct usefulness of some of the existing experimental data for NGNP core designs. Furthermore, it was acknowledged that for data produced on commercial basis or by foreign governments, availability of the data might be quite limited. An effort was made to identify experimental tests of the highest priority, recover the data for those cases, and then develop standard problems (benchmarks) that are of sufficient quality for use in the licensing of the HTGR analysis codes. A set of criteria employed to judge the relevance of the different tests included: purpose of the previous experiment, geometry of core, fuel forms, core materials, physics parameters measured, measurement state, availability of design and uncertainty data, and applicability of data to V&V. Based on these criteria, the experiments judged to be of the highest priorities for the pebble-bed cores are ASTRA, AVR, HTR-10, and HTR-PROTEUS, and for the prismatic block-type cores are HTTR, VHTRC, and CNPS.

Integral evaluations of HTTR and HTR-PROTEUS were performed from 2006 through 2010, and the results have been submitted for inclusion in the *International Reactor Physics Benchmark Evaluation Handbook*. Evaluations of the other facilities will be conducted from FY 2011 through FY 2013.

4.1.2.8 Engineering Scale Reactors

Two gas-cooled test reactors are presently operational for integral experiments: the HTR-10 located at the Institute of Nuclear Energy Technology (INET) in Beijing, China, and the HTTR at the Japan Atomic Energy Agency (JAEA) in Oarai, Japan. Since integral experiments are the only experimental sources that may be able to produce the complex interactions between dominant phenomena identified in the NGNP system-specific PIRT, they are essential important for systems analysis and CFD code validation studies. D data from both the HTTR and the HTR-10 will be important in the calculational matrix required for plant licensing by NRC.

Sketches of these two facilities are presented in Figure º13 and Figure °14 below. Validation studies are needed using the data generated at these facilities to date. In addition, arrangements will be made to enable the NGNP Program to collaborate with INET and JAEA such that specific experiments may be specified that can be linked directly to the NGNP preliminary and final design PIRTs.

HTR-10

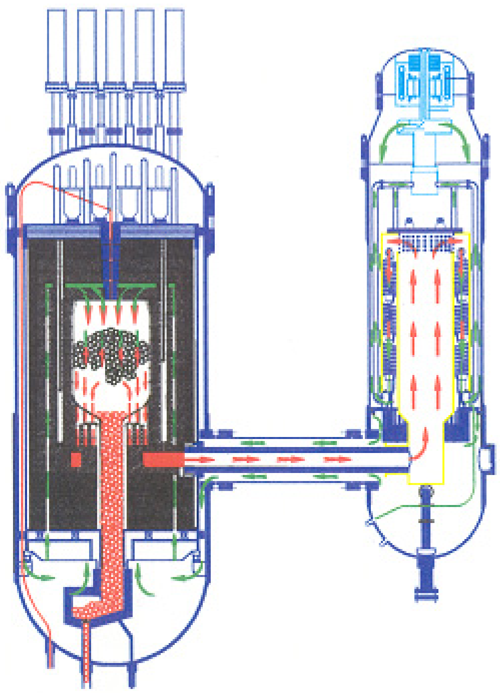


Figure 13: Schematic of HTR-10

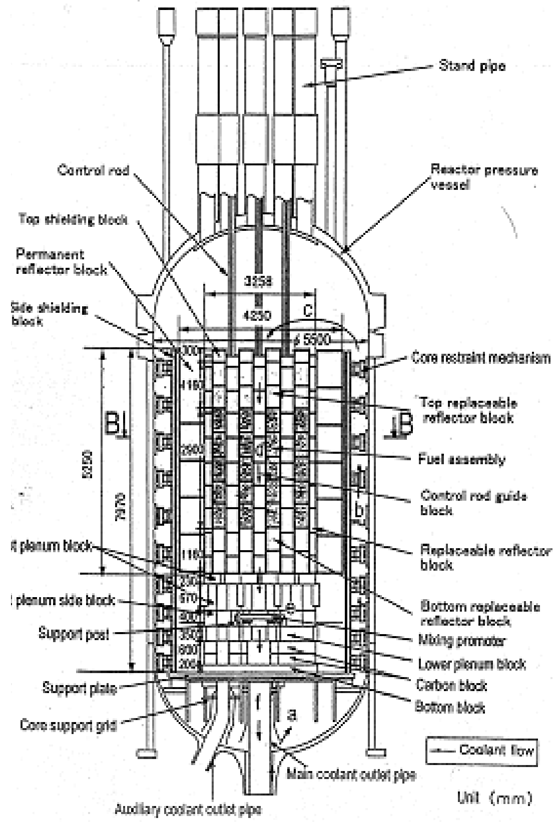


Figure 14: Schematic of HTTR.

HTR-10 is a 10 MW pebble bed HTGR that became operational in 2000. INET plans to perform a spectrum of experiments essential to the NGNP Project. Among the experiments may be a pressurized conduction cooldown experiment (PLOFC), a rod ejection experiment, and an anticipated transient without scram. Tritium permeation measurements have been taken and were acquired by the Project in 2010. This data will be used to validate the TPAC code recently developed at the INL to study tritium permeation in HTRs.

The HTR-10 reactor vessel (see Figure 33) is approximately 11.2 m high with a 1.8 m diameter core that is 1.97 m high with ~27,000 pebbles. The reactor is designed to operate at 10 MWth. The average power density is 2 MW/m^3, the core inlet temperature is 250 to 300°C, and the core outlet temperature ranges from 700 to 900°C. Benchmark experiments performed in HTR-10 are available via the International Atomic Energy Agency (IAEA).

HTTR

The HTTR Project is centered on the 30 MWth prismatic engineering test reactor shown in Figure 14. However, the HTTR Project also has a number of support projects that provide useful data (e.g., the vessel cooling system test series based on cooling panels inside a vessel containing heating elements and the heat transfer studies based on the hemispheres heated from below and cooled using natural convection). JAEA has planned a spectrum of HTTR experiments that include various reactivity transients and loss of cooling conditions. Some of these tests have been performed and still others are planned in 2010 through 2012. JAEA has proposed a collaboration on HTGR research and development that would provide data to the NGNP Project for the validation of codes.

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

Supporting experiments include a series of tests performed to simulate the heat transfer to the VCS cooling panels [see IAEA 2000]. The experiments are summarized in Table 2.

Table 2. VCS experiments: HTTR Project.

Experiment	I	II	III	IV	V	VIa	VIb
Gas	Vacuum	Helium	Nitrogen	Helium	Helium	Helium	Helium
Pressure MPa)	1.3 × 10-6	0.7	1.1	0.47	0.64	0.96	0.98
Power (kW)	13.1	28.8	93.9	77.5	29.7	2.6	8.0
Cooling panel	Water	Water	Water	Water	Air	Air	Air

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

A 50-day run at full power was completed early in 2010. Nine transient experiments are planned to study the reactor response to a loss of cooling under different circumstances (Figure 15). Data from these tests may be provided to the NGNP Project under a collaborative research and development program.

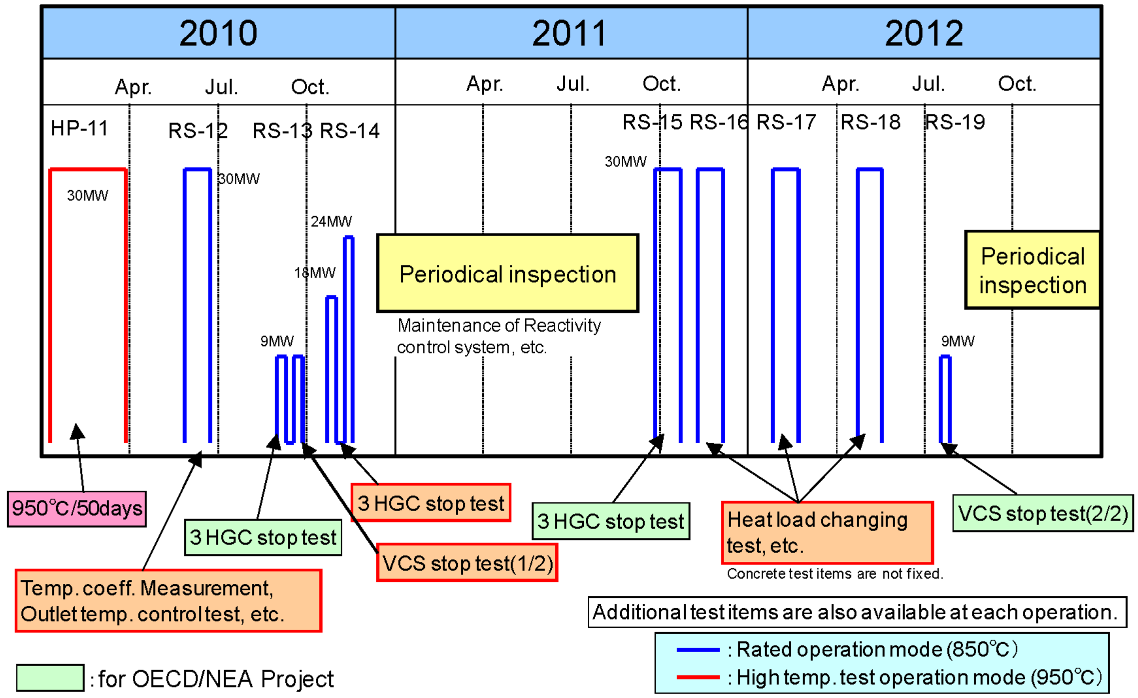


Figure 15. Planned experiment campaign in the HTTR.

4.1.2.9 Integral Experiments in the High Temperature Test Facility

An integral facility is one that is scaled directly to the reference reactor and which can be used to study the majority of the phenomena for the scenarios of interest including the phenomena interactions for each phase of the scenario. For example, during steady-state operations the core power distribution influences the exit helium temperature distribution entering the lower plenum and thus influences the potential for mixing in the lower plenum and the potential for having hot streaking that could translate to large temperature gradients in the gas velocity profiles leaving the reactor vessel and entering downstream heat exchangers or power conversion equipment. Phenomena in the core are thus related the phenomena in the lower plenum and downstream equipment. These phenomena interact with one another. Similarly, for depressurized conduction cooldown and pressurized conduction cooldown, the phenomena occurring in the lower plenum affect the heat transfer and phenomena occurring in the core and reflectors.

The reference prismatic reactor proposed for the NGNP is a modular high temperature gas-cooled reactor (MHTGR). Based on the MHTGR, an HTTF is being designed and will be constructed at a facility at Oregon State University in Corvallis, OR. The HTTF is scaled to one quarter of the size of the MHTGR and will have an electrically-heated core (see Figure 16). The first HTTF configuration is prismatic, however subsequent HTTF configurations may also be pebble-bed, depending on the need.

The HTTF will be operational for startup testing in FY 2012 and the formal prismatic test program will begin in FY 2013. It is anticipated that the HTTF will generate data for several years and the experimental test matrix will be tailored to match the experiments scheduled for inclusion in all of the other experimental facilities, including the reactor cavity cooling system, plenum experiments, core heat transfer experiments, air ingress experiments, and bypass flow experiments.

4.1.3 Accomplishments and Status – Experimental V&V and CFD Studies

The following list presents the accomplishments and status of experimental V&V and CFD studies:

- The preliminary validation matrix of experiments has been defined.
- Seven experiment types have been specified to produce data for the needed validation matrix.
- Lower plenum flow experiments under in the MIR Facility have been completed.
- Two experiments underway are the bypass flow experiments in the MIR facility and air ingress experiments.
- Preliminary designs of the integral experiments—the reactor cavity cooling experiment (at ANL) and the integral reactor vessel experiment (HTTF at Oregon State University)—have been completed and are being refined.
- The HTTF will initiate shakedown testing in FY 2012.
- Contract negotiations are underway to acquire HTTR test data.
- The NGNP is working closely with universities via NEUP to define and build experiments that complement the NGNP. Presently at least 15 universities are cooperating with NGNP via NEUP.
- The V&V30 standard committee has been established.
- Practices and procedures for reviewing and accepting thermal-fluids software are being formulated.

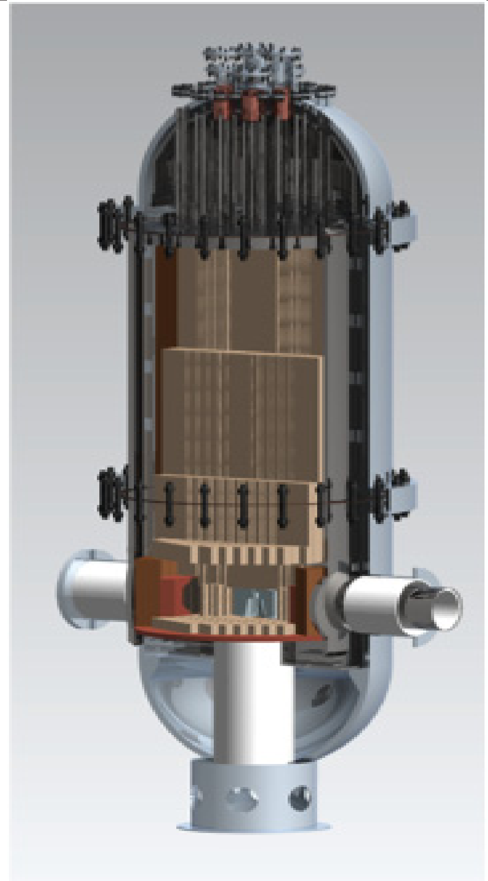


Figure 16. The HTTF integral experiment: a one-quarter-scale experiment based on the MHTGR.

4.2 Core and Plant Simulation

Simulations of the reactor core and plant under steady-state and transient core conditions form the basis licensing calculations. Confidence in the results of such analysis is required by the plant vendor, the regulator, and the plant operator. Early HTGR development and demonstration in the United States and Germany relied less on plant simulation and more on conservative analyses, post-construction testing, and improvements derived from operational experience. The NGNP will rely much more heavily upon simulations to ascertain plant behavior under all anticipated circumstances. The design and analysis codes developed under the early HTGR programs; however, have not been subjected to the continual improvement and operational validation enjoyed by LWR simulation codes. Hence, the codes used by vendors for licensing calculations, and the codes used by the NRC to evaluate those calculations, are characterized by numerous simplifying assumptions and large uncertainties. The main objectives of the NGNP Core Simulation task are to develop high fidelity models and benchmarks for investigating challenging HTGR phenomena and scenarios, and to increase the confidence or at least quantify the uncertainties in vendor calculations and NRC evaluations models.

4.2.1 Goals, Assumptions, and Requirements

Core and plant simulation **goals** are as follows:

1. Develop a 3-D core simulation (burnup and transient) capability that simulates anticipated normal and off-normal scenarios.
2. Ensure that analysis tools will generate results for HTGR benchmarks standard problems that meet all acceptance criteria for accuracy and uncertainty.
3. Develop a capability to quantify key core safety parameters and their sensitivities to specified material and boundary conditions.
4. Define and construct core steady-state and transient reference problems that reflect a broad range of normal and anticipated off-normal core events.
5. Perform simulations that estimate the range of conditions for fuels and materials to inform fuel and material testing.

Core and plant simulation **assumptions** are as follows:

1. HTGR core neutronics can be represented accurately using neutron diffusion theory and appropriately homogenized cross sections.
2. Uncertainties in key safety parameters generated by core analysis software can be adequately quantified using available techniques and software.

Core and plant simulation **requirements** are as follows:

1. The core simulation capability shall rely on a different set of tools than those being used by or developed for the NRC in order to provide independent confirmation of analysis results.
2. Neutronics models of the core shall account for local absorbers (burnable poisons, control rods) in order to accurately compute shutdown reactivity margins and temperature reactivity coefficients.
3. Neutronics models shall account for all levels of heterogeneity in the fuel and core.
4. Fuel management models shall account for anticipated block and pebble loading scenarios.
5. Uncertainty and sensitivity analyses shall conform to industry and regulator standards where defined (e.g., Code Sensitivity and Uncertainty (CSAU) Method).
6. Core steady-state and transient reference problems shall reflect a broad range of normal and anticipated off-normal core events.

4.2.2 Scope of Core and Plant Simulation

The neutronic design and operational analysis of the NGNP requires core analysis tools (codes and data) for (1) cross section preparation and fuel assembly lattice calculations to produce effective nuclear parameters for subsequent whole-core analysis, (2) static reactor analysis for core design and fuel management, (3) reactor kinetics and safety analysis, (4) evaluating the impacts of material-neutronics interactions on core design, (5) core heating and shielding calculations, and (6) decay heat calculations. The codes must also be qualified for use in safety evaluations. A recent review of existing state-of-the-art diffusion and transport whole-core analysis capabilities indicated that those codes containing R-Z (R-θ-Z) and Hexagonal-Z geometry options could be used for HTGR modeling. However, there are certain features of the NGNP design that require modifications to the available capabilities. Nearly all the advanced reactor physics tools in the U.S. nuclear industry have been developed for the analysis of LWRs, the dominant reactor types in the United States. There are however differences between the LWR and the NGNP designs that make the direct applications of the LWR tools inadequate for NGNP analysis. The physics characteristics of the NGNP are quite different from those of the commercial LWRs because of the (1) uniquely heterogeneous composition of the fuel and core (Figure 17) (2) solid graphite moderator and reflector, (3) long mean free path of neutrons relative to the size of the block or pebble spectral zone, (4) use and location of control absorbers (5) large temperature rise across the core, and (6) large holes for guiding control rods in fuel and radial reflector positions in prismatic HTGR designs.

The effect of the geometry of TRISO fuel particles embedded in compacts or spheres (termed the double heterogeneity effect) is felt in fuel utilization and reactivity feedback. It must be adequately represented in the lattice physics code used for the NGNP analysis in order to obtain accurate results in the core simulations over the entire fuel cycle. This effect has been found to be about 2 to 4% Δk/k (reactivity) in NGNP assemblies/cores using enriched uranium fuels, and about 10 to 15% Δk/k for those using transuranics fuels as in the deep-burn concepts. The NGNP blocks and core are also neutronically thin (as measured by the mean free path) compared to LWR assemblies and core, which poses a challenge to the neutronic codes based on homogenized few group constants. Consequently, more energy groups have to be used than for conventional LWR analyses, and multi-assembly or partial-core calculations (Figure 18) are needed for generating accurate homogenized cross sections and correction factors.

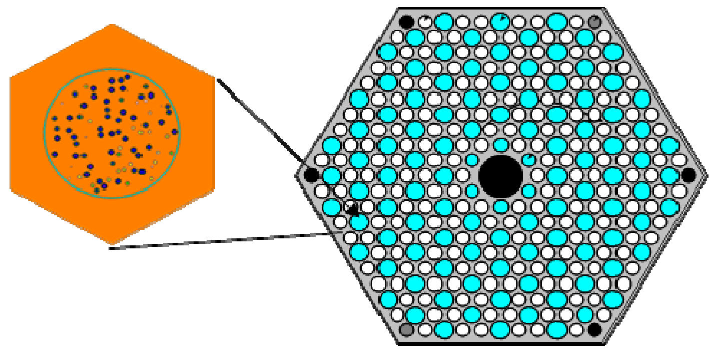


Figure 17. Heterogeneity within a single prismatic fuel block.

Figure 19 shows the results of a sensitivity study in which one-group diffusion parameters were generated for a single prismatic block as a function of the extent of the surrounding core that was included in the model. The plots indicate that one must include at least 50 cm (almost two blocks in any direction) of the surrounding core to adequately capture leakage effects within the block. Because the core annulus for the current NGNP designs contains only three rings of blocks, a lattice core calculation must include all of the surrounding fuel blocks and at least one reflector block. The traditional color-set approach used in LWR assembly calculations, which includes only one or part of one neighboring assembly, would yield inaccurate results.

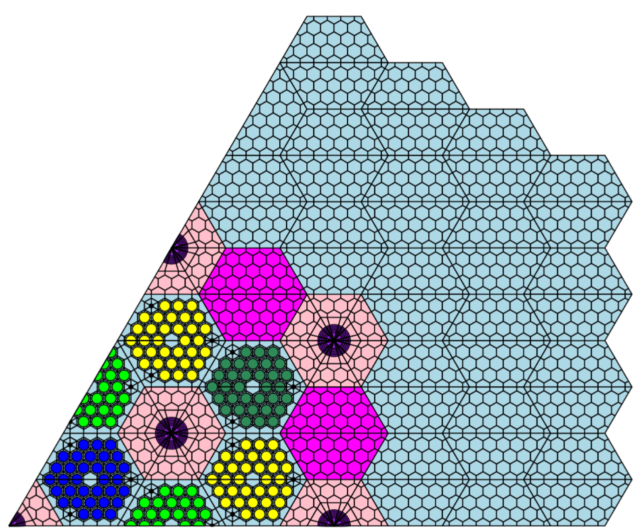


Figure 18. Partial core model of a prismatic reactor used for cross section generation.

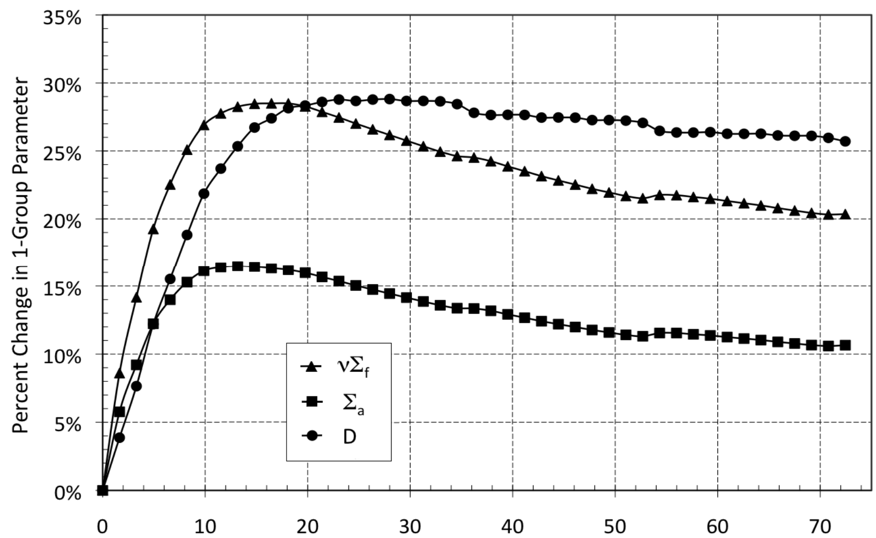


Figure 19. Extent of domain beyond a single block needed to capture leakage spectra.

The lattice physics and whole core diffusion codes should be able to treat all of these effects in order to accurately compute the temperature coefficients, core reactivity, flux, and power distributions in the annular core. Local neutron streaming effects arising from the large control rod holes in prismatic HTGR designs should also be accurately represented. The core depletion state, including the nuclide number densities and core burnup distribution, should also be accurately predicted. These core physics parameters have direct impact on thermal-fluids/safety analysis, fuels and materials designs, and plant economics. Additionally, the code suite should be computationally efficient in order to perform the large number of calculations required to support core scoping analysis and detailed designs in reasonable time.

The pebble bed neutronics problem is similar but somewhat less severe. Burnable poisons are not used in the leading pebble bed core designs and the important heterogeneities that must be treated in the cross section generation process are largely 1-D in nature. Like the prismatic reactor, however, the mean free path of neutron is relative to the size of the spectral zones—the equivalent of blocks in the prismatic core and the region for which cross sections are generated. Leakage between these spectral zones is significant, particularly in the radial direction, and therefore partial core calculations are also required. Like the prismatic reactor, these partial core calculations are also required to obtain cross sections for the reflector regions in which there is no internal source of neutrons.

4.2.2.1 Prismatic Reactor Core Simulation

In addition to the challenges of HTGR analysis described above, burnable poison and control rods pose further challenges for prismatic core modelers. In current NGNP designs, burnable poisons are placed at some of the vertices of the hexagonal blocks (Figure 20). In the traditional lattice physics approach, the effect of these poisons is smeared across the block. Even though the effect is limited to just part of the block, it extends into the neighboring blocks, further necessitating the need for partial core lattice calculations rather than simple block models. Burnable poisons also pose a challenge for the diffusion-based core simulators. Legacy hexagonal geometry codes, such as DIF3D that assume a homogeneous block composition, do not treat local absorbers adequately. Under NGNP, INL has developed a modification to the analytical Nodal Green's Function Method (NGFM) for nodal diffusion codes that explicitly treats local absorption in the nodal balance equation. The new method is being implemented in the HEXPEDITE code.

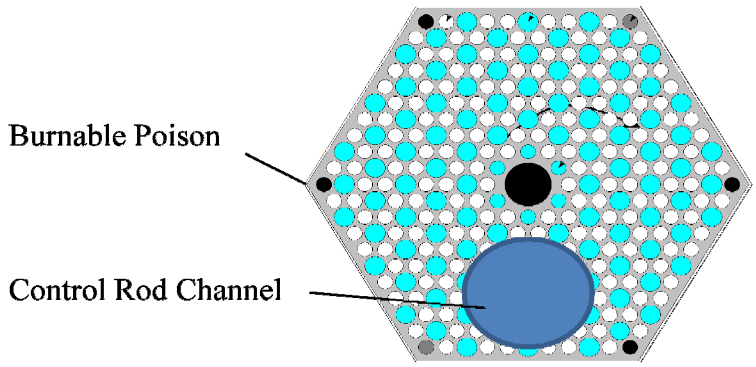


Figure 20. Absorber locations in prismatic blocks.

Some NGNP designs contain large channels for control rods in some of the fuel or reflector blocks. When filled with an absorber, this region exhibits considerable anisotropic scattering that is difficult to treat with traditional homogenization techniques. When empty, the channel provides a large hole for neutron streaming that is not well-captured, even by sophisticated transport codes. Computing compact power peaking in the presence of either of these features is a considerable challenge for reactor physicists.

Advanced nodal diffusion methods typically employ nodal equivalence parameters (discontinuity factors) to reduce homogenization errors arising from core heterogeneity (different rodded and unrodded fuel and reflector regions, and interfaces between the regions). Surface-dependent discontinuity factors are particularly very useful to take into account geometric asymmetry in the nodal approach, and thus must be provided. The need for surface-dependent discontinuity factors in nodal calculations necessitated the modification of several routines in the DIF3D-nodal Hex-Z version of the code (DIF3D-nodal). It was originally thought that using surface-dependent discontinuity factors in the DIF3D-nodal option would give good accuracy for all core configurations. This has not been the case for rodded configurations because of the relatively poor transverse leakage approximation made for the nodal option (particularly when a large hexagonal pitch is used in the code). Generally, REBUS-3/DIF3D results for the core multiplication factor and power distribution were found to be in good agreement with MCNP results, particularly when discontinuity factors are applied. It was also shown that the DIF3D-VARIANT option provides a better spatial solution in its diffusion approximation. In addition, it was observed that control rod worths could be estimated within an acceptable range compared to MCNP results. However, the core power tilt (particularly in the rodded zones) was not accurately modeled but perhaps could be improved with modifcation to the existing methods.

For transient calculations, one also requires an accurate yet fast neutronic solver. As mentioned above, the DIF3D-nodal solver, which uses the nodal expansion method to treat transverse leakage terms, in not sufficiently accurate, but it does run fast enough to be included in a transient core simulator. The more accurate DIF3D-VARIANT nodal transport solver, on the other hand, is too slow to be practical for many HTGR transients. The NGFM solver in HEXPEDITE retains the accuracy of DIF3D-VARIANT and the speed of DIF3D-nodal.

Table 3 shows the results of a comparison between HEXPEDITE and the different solvers available in DIF3D (nodal diffusion, fine mesh finite different diffusion, and nodal transport) for a simplified HTTR benchmark with all control rods inserted.

Table 3. Comparison of k_{eff} and runtimes for HEXPEDITE and various DIF3D solvers.

Code	Method	k_{eff}	Deviation from Reference (%Δk/k)	Runtime [sec]
HEXPEDITE	NGFM	0.87130	0.11	0.8
DIF3D	NEM	0.86384	-0.63	1.8
DIF3D	Fine Mesh FD	0.87139	0.12	1250
DIF3D	Nodal Transport	0.87017	—	380

HEXPEDITE yielded a core eigenvalue quite similar to the fine mesh diffusion and nodal transport solutions of DIF3D but with runtime comparable to DIF3D-nodal. For this reason, the time-dependent HEXPEDITE solver is undergoing testing for use in transient analysis. A matching depletion solver is being developed that would use the hexagonal nodal flux solution generated by HEXPEDITE to accurately predict compact powers during burnup analyses.

Transient Analysis

Both normal (load follow) and off-normal transients such as recuperator bypass (cold He injection) or inadvertent rod withdrawal can place considerable stresses on fuel and structural materials. More importantly, these reactors rely on Doppler feedback rather than control rods to shut down the fission reaction in the event of a significant reactivity insertion. An accident analysis must involve coupled, time-dependent diffusion and thermal fluid analysis in order to capture and bound all anticipated transient phenomena. For rod adjustments and some thermal initiators, simple temperature feedback coefficients can be estimated from stand-alone neutronics analyses and used in system codes with simple point kinetics solvers. These are relatively fast and reasonably accurate if the reactivity insertion is minor.

For the larger reactivity events listed above, spatial effects can be considerable and result in unacceptable local fuel stresses or xenon oscillations. In these cases, fully coupled 3-D neutronics/thermal fluid analysis is required. These are computationally intensive and can take days to complete, even a simple reactivity simulation. Coarse mesh nodal solvers, such as HEXPEDITE, relieve the computational burden by solving the diffusion equation in a higher fidelity form but over a much coarser mesh. Nonetheless, the solver must be very efficient as the system of equations must be solved at each time step. Time steps may be a short, at milliseconds, earlier in the transient but lengthen to minutes later on.

Likewise the thermal fluid solver must be similarly efficient and detailed in order to capture the thermal feedback effects that dominate these transients. Traditional coupling of the neutronic and thermal fluid solvers involves solving each separately (split operator) and passing parameters (power density, temperatures) at orchestrated intervals. Care must be taken in choosing the time step to avoid stability and accuracy problems in the numerics.

INL is taking two complementary approaches to prismatic transient analysis. The HEXPEDITE code contains a time-dependent neutronics solver but it has yet to be adapted and tested on HTGR problems. This will be accomplished in 2011. The new solver will be implemented into the RELAP5-3D code for simulating minor reactivity and thermal transients. Severe reactivity insertions will require a faster, more tightly coupled, high fidelity thermal fluid solver. It is planned to incorporate the HEXPEDITE solver into the MOOSE multiphysics framework to be discussed in Section 4.2.2.3. Research is also just underway to develop a coarse mesh CFD solver than can be used to provide high fidelity temperature profiles but with the speed closer to that of a system code.

Burnup Analysis

The diffusion solver in HEXPEDITE (or its functional equivalent) can be used to drive fuel management calculations by coupling the neutronic solver to nodal depletion and fuel shuffling solvers. The power, temperature, and feedback properties of a HTGR change with burnup and the regulator will require that transient simulations be performed at different stages in the burnup cycle. A nodal depletion solver uses the spatial flux profile generated from the nodal diffusion solution to simulate the depletion of fuel across the blocks. As with transient analyses, high fidelity Monte Carlo/ORIGEN based depletion is computationally too demanding for routine fuel management and sensitivity studies. Work began in 2010 to convert the Cartesian-geometry nodal depletion solver NOMAD to work with the hexagonal HTGR geometry. Testing will commence later in FY 2011.

4.2.2.2 Pebble Bed Reactor Core Simulation

For steady state and depletion problems (core design and fuel management), the INL has developed the PEBBED code. PEBBED solves the multigroup diffusion and burnup equations for recirculating pebble bed cores in which the fuel is continuously loaded and moving downward through the core during operation.

Under NGNP, the COMBINE code has been modified to generate accurate cross sections for PEBBED analysis. COMBINE originally provided the functionality of traditional pin cell and pebble codes such as the GAM-ZUT-THERMOS sequence in the German PBR simulator VSOP. It possessed a fast spectrum slowing-down module with Bondarenko and Nordheim Integral resonance treatments. The fast module generated an s-scattering source for the thermal module which supported upscattering and a crude treatment of self-shielding formulated for LWR fuel pellets.

COMBINE has since been modified to support PBR core simulation. The B-3 transport equation is now solved over the entire spectrum with simultaneous upscattering and resonance treatments in 167 energy groups. User-supplied buckling terms can reflect net inward or outward leakage. COMBINE accepts separate kernel-to-kernel (intrapebble) and pebble-to-pebble (interpebble) Dancoff factors generated by the PEBDAN code to account for shadowing. A 1-D discrete ordinates transport (ANISN) solver has been embedded in the code to capture spatial effects. With pebble bed geometry in mind, COMBINE employs a multistage homogenization process that minimizes the error inherent in the multigroup approximation. Explicit transport models of the TRISO particle, pebble, and radial core wedges are solved in 167 groups before being coalesced in energy and space to generate few-group cross sections for PEBBED (see Figure 21). Axial and azimuthal leakages are still treated with transverse buckling terms. Radial leakage, which is the dominant contribution to leakage from the spectral zones, is captured explicitly in the core transport stage and in 167 groups. The source for the radial reflector regions (including control rods) is the true current emanating radially from the core region with adjustments for axial and azimuthal effects.

The implementation of the multistage approach to homogenization in COMBINE enables the explicit modeling of the different structures that are present. For the pebble bed itself, the transport equation is solved in spherical geometry first for the TRISO particles and then for the pebbles and surrounding coolant. Explicit geometrical models are not required for the bulk of the reflector and core barrels as these regions are largely homogeneous. A 1-D cylindrical transport calculation is used, however, in the homogenization of control rod regions. Shows the agreement that can be obtained in the few-group flux profiles between the PEBBED diffusion calculation and the ANISN transport calculation.

Figure 22 is a simplified reactor model in axial variations and leakage have been neglected to facilitate a direct comparison to the radial transport solution. The solid lines are the transport flux profiles and the points are generated by PEBBED, The agreement is good even through the control rod region (green).

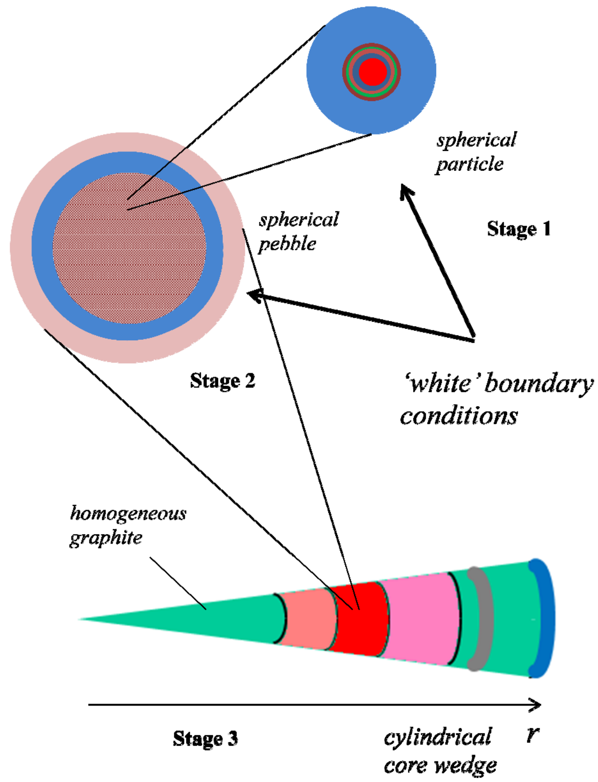


Figure 21. 3 Stage homogenization process in COMBINE.

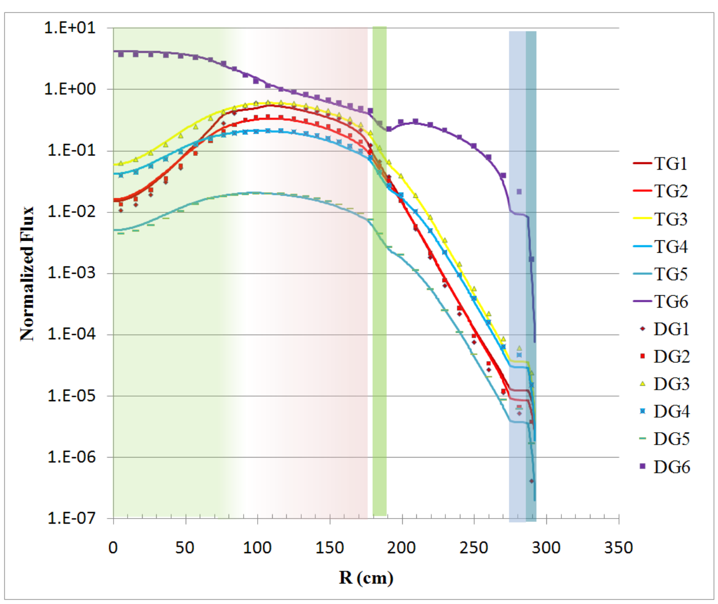


Figure 22. Radial flux profiles in a pebble bed reactor with a dynamic inner reflector.

Transient Analysis

Concurrently with PEBBED development, the CYNOD kinetic solver was developed to support PBR transient analysis. Using the same cylindrical diffusion solver as in PEBBED, CYNOD has been recently upgraded to support 3-D analyses. The treatment of control rods in transient simulations is more complicated in transient analyses because the simple treatment of the control absorber in a computational node leads to nonphysical cusping of power ramps. For this reason, a more sophisticated approach that uses 3-D response functions is being developed in conjunction with Georgia Tech. These response functions are computed using high fidelity transport codes but are integrated seamlessly in the nodal diffusion core simulator for fast yet accurate reactivity simulation. The modifications were completed in FY 2010 and testing is to be completed in FY 2011.

Temperature feedback in these calculations requires coupling of the CYNOD solver with an appropriate thermal fluid solver such as RELAP or THERMIX-KONVEK. RELAP has the advantage of being a world standard for systems analyses codes. The coupling of CYNOD to RELAP was completed in early Fyn 2010 and demonstrated on a PBMR400 Transient Benchmark problem. The coupling was observed to be quite inefficient, and more optimization is required to make it suitable for NGNP analyses. The THERMIX-KONVEK code is a legacy pebble bed reactor thermal fluid solver developed in Germany. It has been coupled to both PEBBED for steady state calculations (see Figure 23) and to CYNOD for transient analysis. THERMIX supports 2-D (RZ) analyses only, and has other computational limitations that prevent it from being used in many HTGR investigations. Nonetheless, it has been used for to investigate a range of fuel cycle and reactivity problems in support of NGNP goals.

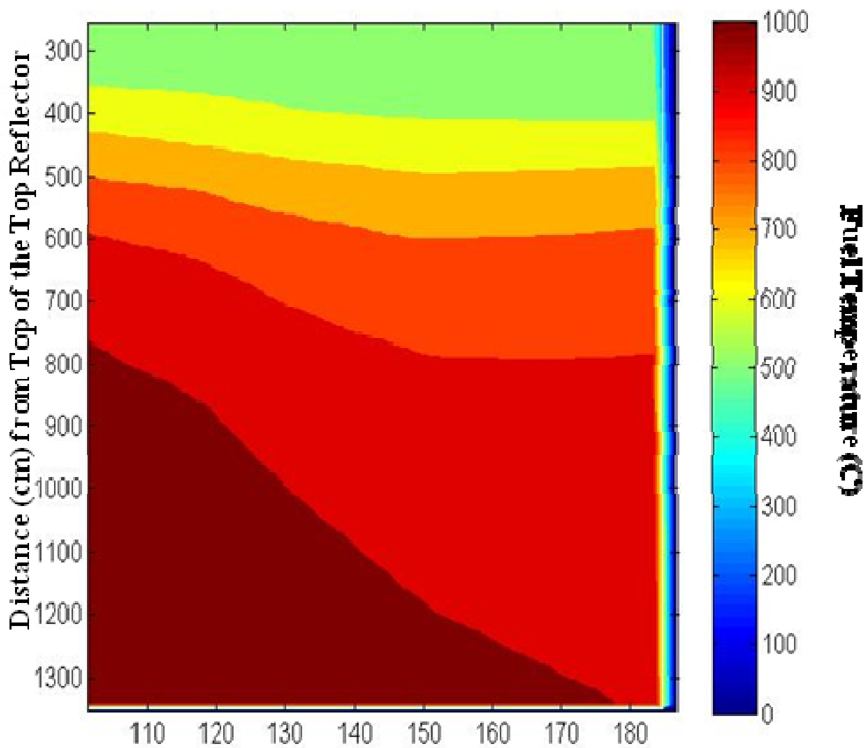


Figure 23. Colormap of the fuel temperature profile in the PBMR400 using PEBBBED-THERMIX.

The ability of the HTGR to shut itself down after a reactivity event is a consequence of Doppler broadening of neutron capture resonances in U-238 and is strongly dependent upon the geometry and temperature of the fuel. Low-order thermal fluid analyses contain pebble models in which the heavy metal is homogeneously mixed with the graphite rather than lumped into kernels. The temperature of the mixture near the center of the pebble is taken as the temperature of the fuel. This is a reasonable assumption for steady-state analysis but leads to considerable error in transient calculations. In FY 2009 and FY 2010, the fuel temperature model in the THERMIX code was modified to compute the kernel temperature during a reactivity transient. The effect on the predicted trajectory of a rod ejection event is shown in Figure 24. Three fuel temperature models were tested with the time-dependent heterogeneous fuel model (yielding a better prediction of the physical behavior of the core as compared to quasi-static homogeneous and heterogeneous fuel models. This model will be incorporated into other thermal fluid solvers being used for PBR analysis and be adapted for use in prismatic transient analysis in FY 2012.

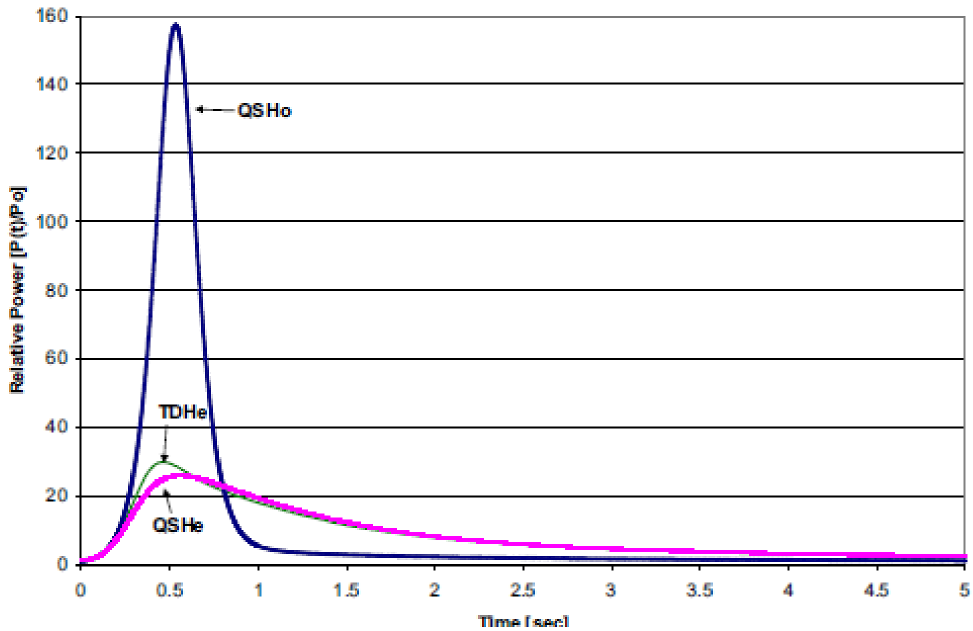


Figure 24. Comparison of core power (relative to nominal) assuming three different fuel temepratures models in CYNOD-THERMIX.

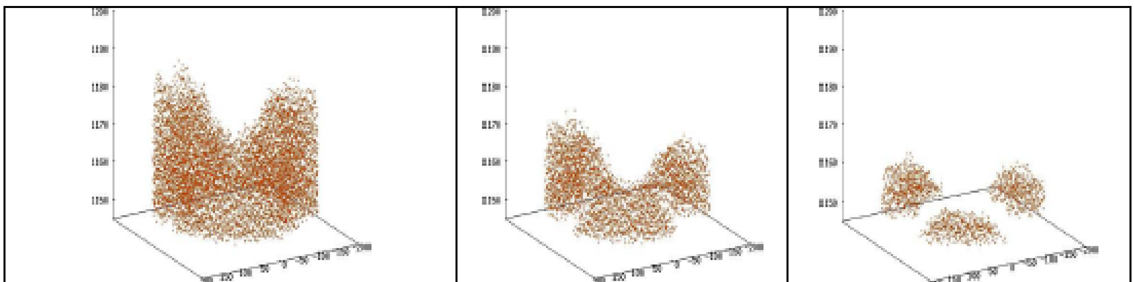
Non-equilibrium Core Burnup States

Unlike VSOP, PEBBED converges directly upon the equilibrium core state, i.e., the asymptotic flux and burnup profile, that a PBR achieves with continuous loading and discharge of fuel. This state is achieved only after the reactor has been operating at steady-state for a considerable period of time (6 months to a few years). Because the preasymptotic operation phase will contain a significant amount of fresh fuel, there may be periods during this interim state in which the reactor attains a reactivity in excess of the equilibrium core. A regulator will likely require that this peak reactivity state be determined and used as a starting point for transient analyses. The PEBBED burnup solver must be modified to treat the transition period leading up to this asymptotic state in order to provide the initial conditions for safety simulations. This effort will require about 1 person-year of development time but it is of lower priority than the other tasks underway.

Pebble Motion

The accurate computation of burnup in the pebble bed requires knowledge of the direction and speed of the pebble flow. INL has developed the PEBBLES code, which simulates the mechanics of flowing pebbles. PEBBLES has been used to simulate the loading and pebble flow in the PBMR400, but the simulation takes too long to support practical core design calculations. Progress in parallelizing the code was made in FY 2010 to support future burnup analyses using PEBBED. PEBBED currently assumes pebble flow profiles obtained empirically from literature sources.

The PEBBLES code has other uses as well. It has been used to simulate the densification of the pebble bed during an earthquake. The result was fed into a CYNOD-THERMIX simulation to estimate the reactivity effect of shaking the vessel. The effect was shown to be measurable but not significant in terms of core safety [Ortensi-2]. Figure 25 shows the settling of the top loading cones in the PBMR 400 during an earthquake, while the plot in Figure 26 shows the power response of the core.



Time [sec]	0.0	5.3	33.7
Pack. Fraction	0.6122	0.6145	0.6224
Core Height [cm]	1184.38	1171.31	1160.98

Figure 25. Snapshots of PBR loading cones during an earthquake as simulated using PEBBLES.

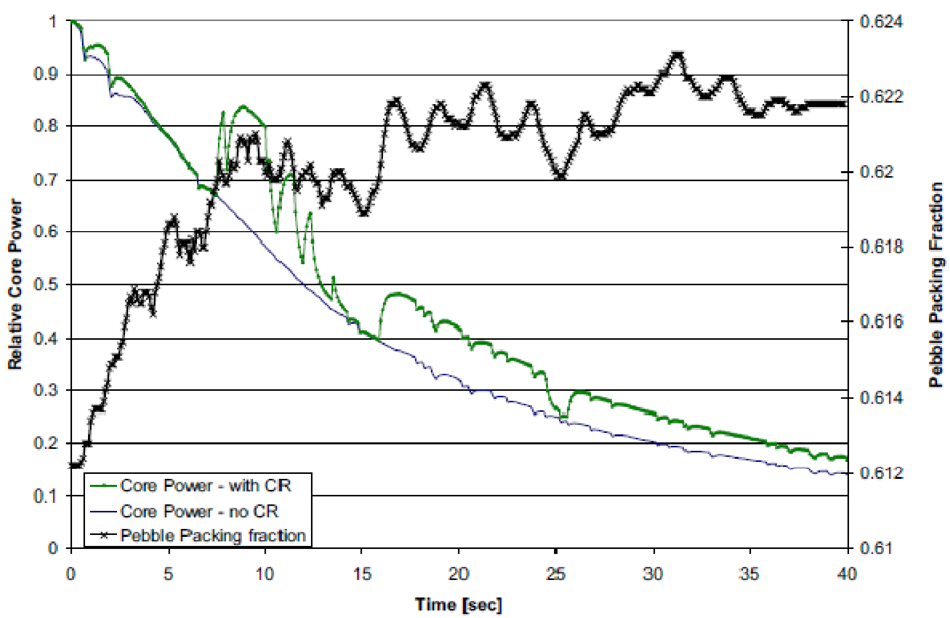


Figure 26. Core power trajectory and pebble packing fraction during an earthquake.

Modifications to the PEBBLES code commenced in FY 2010 to support analysis of dust generation. Friction and wear correlations were being added to the governing equations to predict the creation of dust in the primary. Validation of the code is pending the generation of graphite tribology and friction factor data.

4.2.2.3 Core Thermal Fluids

Core and system thermal simulation of both the pebble bed and prismatic HTGRs can be performed accurately to first order with low order fluid dynamics and heat transfer equations that assume either a uniform porous medium (pebble bed) or a network of individual pipes (coolant channels in prismatic reactors) with a common pressure drop. The density is also assumed to be constant (Boussinesq approximation). These codes rely on experimentally determined friction factor and heat transfer correlations appropriate to the geometry and materials used.

Pebble Bed Thermal Fluids

As mentioned in the previous section, the THERMIX-KONVEK or THERMIX-DIREKT codes (or its governing equations) are used in many PBR core simulators, including VSOP (Very Superior Old Programs), PEBBED, and PANTHERMIX. Convective heat transfer and pressure drop correlations for pebble beds of uniform packing fraction have been validated in experiments such as SANA [IAEA 2001]. Data from the SANA experiment is often used to validate the basic thermal fluid solvers for new pebble bed simulators such as the PRONGHORN code under development at INL (see Figure 27).

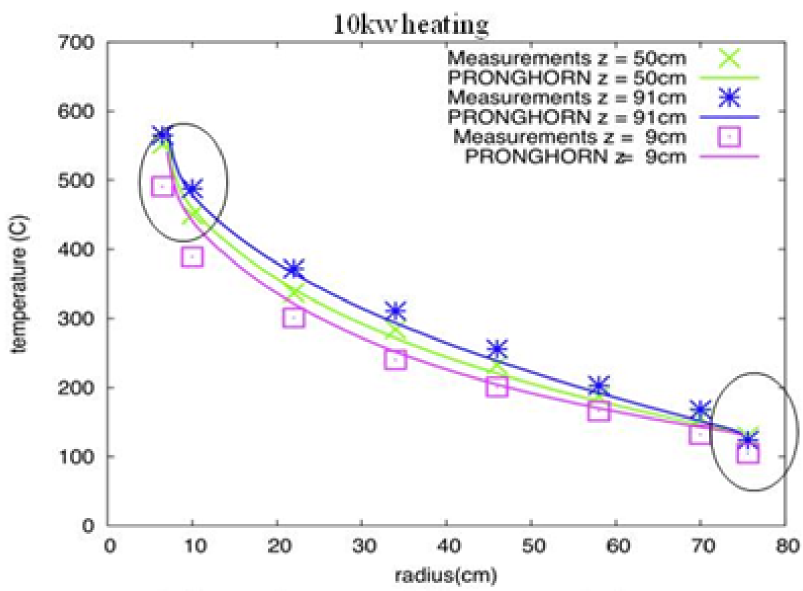


Figure 27. Measured temperature profile in the SANA experiment vs. profile predicted by PRONGHORN.

Codes based upon the THERMIX equation set yield good agreement with data in the middle of the pebble bed but the correlation fails somewhat near the walls where the bed porosity approaches unity. For system analysis, this error is assumed to be negligible or empirical corrections to the heat transfer correlations are applied. CFD analyses in which pebbles are modeled explicitly are computationally expensive and thus are limited to small (<50) numbers of pebbles. These may be useful for detailed investigation of turbulence around a pebble, but cannot yet be used for core-wide analyses.

Under low flow conditions, radiation and conduction dominate heat transfer within pebble beds with good agreement with experiments given by the correlations of Zehner and Schlünder [1972] and Robold [Robold 1982]. Such correlations are widely used for PBR accident analysis.

Bypass flow is only approximately modeled with codes like THERMIX and RELAP. Flow between the reflector blocks is modeled as one or more vertical flow channels with the flow streams added to the pebble bed exit flow stream. The equations for a porous medium are also used here with a friction factor arbitrarily chosen to obtain a prescribed bypass flow fraction. Given almost no experimental data on which to support bypass flow estimates, these flow fractions are often guessed or estimated from complex CFD models of the reflector regions.

Prismatic Thermal Fluids

The engineered coolant channels drilled into prismatic fuel blocks are long circular tubes for which heat transfer and pressure drop characteristics have been determined experimentally. Simple unit cell models of a fuel pin surrounded by a ring of these channels can provide first order estimates of fuel, graphite, and coolant temperature. This modeling approach, used in RELAP, is fast and can yield overall core behavior but it assumes that each block has a uniform temperature. Each block is a RELAP heat structure that can exchange heat with its neighbors. Bypass flow is often not modeled, but can be roughly approximated as in THERMIX. A better approximation often used in core simulations model bypass flow using 1-D flow channels with the distribution of flow solved using a common pressure drop and assumed geometry and friction factors. Such models neglect the horizontal flow that may form along the top and bottom block surfaces, but can still yield reasonable 2-D or 3-D temperature profiles.

Detailed effects of multidimensional bypass flow on block temperature profiles is being investigated experimentally and computationally (see Section 4.1.2.1). Most modeling is steady-state, and thus far limited to one-twelfth sections of one or more vertically stacked blocks (Sato 2010). Under the SHARP project, a one-twelfth model of an entire HTGR core is being modeled (steady-state) with CFD and neutron transport using the massively parallel computing cluster at Argonne (Pointer 2010). Such calculations can provide computational verification of lower-order system models that use approximate approaches.

One approach just underway in NGNP is to use a two-step homogenization process, which is done in neutronics. A high fidelity CFD model of a block with surrounding bypass channel is used to obtain homogenized or average thermal fluid parameters that are inserted into a coarse mesh model of the entire core. As in neutronics, each block is homogenized with the coupling of heat flux and fluid flow between blocks obtained from the whole core solution. Though still in the theoretical stage, this coarse mesh CFD technique promises to capture the complex physics of HTGR thermal fluids with the accuracy approaching that of CFD but with a computational efficiency that would support 3-D coupled transient analysis.

4.2.2.4 Plant Simulation and Process Heat Plant Coupling

RELAP5-3D is widely used for system-wide analysis of nuclear plants and will be so used for the NGNP. The core model described above can be coupled to various balance-of-plant components to investigate system-level interactions and behavior, and to support economic analysis of process heat plant applications. Improvements in core modeling techniques, either through refinement of the RELAP core heat structures or through the development and implementation of the coarse mesh CFD approach described in the previous section, will be exploited as they become available.

The commercial software packages ASPEN and HYSIS have been acquired by the Project and used to simulate the behavior of candidate process heat and hydrogen production plants. The models developed using these codes will be used to define the load transients anticipated for the NGNP. These transients will be incorporated into the design technical and functional requirements. RELAP5 models of the primary loop (core, steam generator, and.intermediate heat exchanger (IHX) may be coupled ASPEN or use the output data as boundary conditions. As the IHX and steam generators will be most sensitive to core and load transients, the effective coupling of RELAP and ASPEN will be an important challenge for assessing performance. Data from this modeling can also inform in-service inspections and characterizing pressure barriers.

4.2.2.5 Fission Product and Dust Transport

Complimentary simulations of fission product transport through and out of the primary, including the interactions with dust, will be conducted using a system code such as MELCOR. As such codes were not originally designed to model dust, some modifications and experimental validation will have to be performed. A plan is being developed in FY 2010 to identify gaps in the ability to model HTGR fission product and dust transport and will address such issues as liftoff and deposition, dust and fission product chemistry, and multicomponent fission product transport within the reactor buildings. The ability to accurately model fission product transport with MELCOR is an open question and so complementary analysis and evaluations will be needed to assess the value of this activity.

4.2.2.6 Multiphysics Applications

The core simulators described above adhere to the traditional reactor analysis approach whereby high order transport codes are used on subsets of the core (blocks or pebbles) to obtain average diffusion theory parameters for use in fast, lower order whole core simulations. These methods have been deployed very successfully in LWR design and accident analysis but, as described in the previous section, significant enhancements and modifications must be made in order to perform HTGR analysis with the same level of accuracy. The resulting codes will be used for plant simulation for design and safety analysis where time-dependent, whole core or plant behavior is desired.

Modern computational algorithms and massively parallel computing platforms are now beginning to enable high-fidelity transport simulations, not just on subdomains but on substantial sections of the reactor. While these models and codes are still too slow and unnecessarily detailed for many basic design and safety applications, they can be useful for investigating particularly complex scenarios and phenomena. They can also be used to verify the lower order coupled simulations used for most applications. The following paragraphs describe the efforts (underway or planned) to apply so-called multiphysics/multiscale codes and techniques to the HTGR.

High Fidelity Coupled Neutron and Thermal Fluid Transport

A heterogeneous whole-core transport capability using stochastic or deterministic transport theory solution method would be desirable for benchmarking lower order (but faster) core modeling techniques and codes. Such approaches reduce or eliminate the need for cumbersome and complex tasks of lattice cross section generation, condensation, functionalization, local information recovery, etc. On the other hand, such codes require tremendous computing power only available at a few national laboratories. ANL is developing the SHARP computational framework for implicit coupling of high resolution transport codes. Recently, ANL successfully simulated a one-twelfth core of a fresh prismatic HTGR core at steady state using a coupled neutron transport (DECART) and computational fluid dynamics code (STAR-

CCM). While not yet practical for core design, fuel management, and transient analyses, this effort demonstrates that high fidelity coupled transport analysis is possible [Pointer 2010].

An alternative approach being pursued under NGNP is to build a core simulator with MOOSE. MOOSE is a computational platform specifically designed for solving arbitrary and complex systems of partial differential equations. It exploits the computationally efficient Jacobian Free Newton Krylov method for implicitly coupling the physics coded by the user. Because the basic meshing and solver tools are embedded within MOOSE, the code developer need only provide the governing equations that describe the physics of the system. [Gaston 2009].

Fuel Performance and Core Dynamics

The integrity of the TRISO pressure boundary is assaulted during the reactivity and thermal transients of normal power ramps and accident trajectories. INL has developed the PASTA (PArticle STress Analysis) code for computing the mechanical stress on the TRISO boundaries as a function of temperature, gas pressure, and other factors [Boer 2010]. PASTA was coupled to PEBBED to compute the fuel durability of particles in a typical PBMR 400 pebble during its life in the core and in transmutation particles in a deep burn pebble (see Figure 28). It is planned for PASTA to be coupled to the CYNOD-THERMIX core simulation code to simulate the stresses on pebble bed fuel during various power ramps under normal and accident conditions. PASTA will also be coupled to the time-dependent prismatic core simulator (HEXPEDITE-K). Testing will be completed on that code in FY 2011.

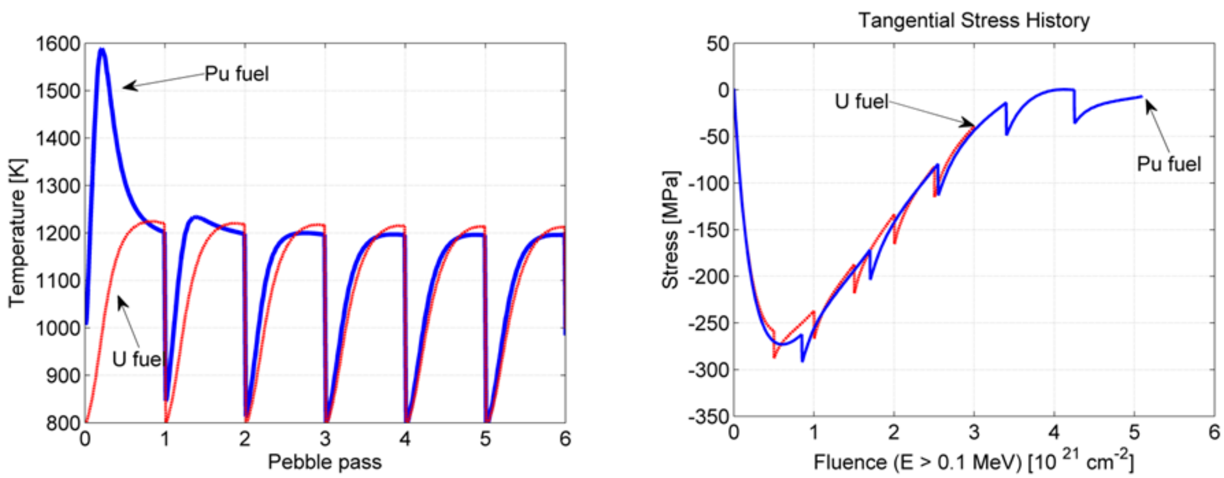


Figure 28. Comparison of the temperature and stress histories of UO_2 and deep burn pebbles.

In late 2008, development of a MOOSE-based pebble bed reactor core simulator commenced with the goal of executing all of the steady-state and transient problems in the PBMR400 Coupled Core Transient Benchmark sponsored by the OECD [OECD 2005]. The simulator, PRONGHORN, would initially model neutronics with finite element diffusion and thermal fluid behavior using the THERMIX-KONVEK equations set [Park 2010]. 3-D results for the steady state exercises were produced in mid-2009 and the remainder of the transient exercises are to be completed in 2010. Figure 29 compares the results from PRONGHORN and other pebble bed transient simulator codes.

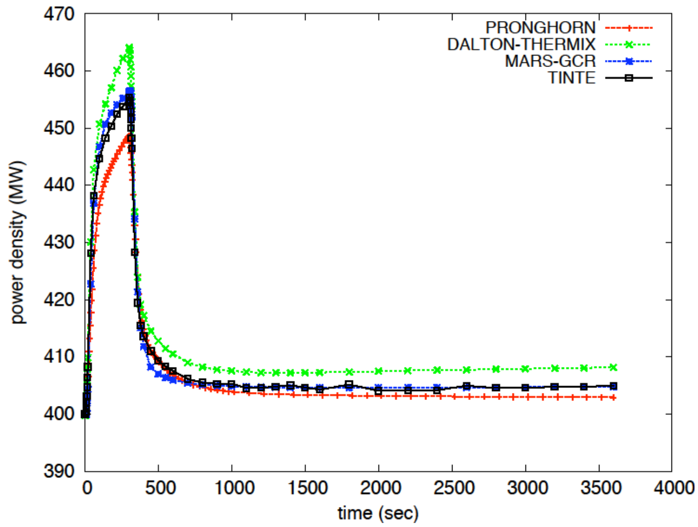


Figure 29. Results of a cold helium injection transient as generated by PRONGHORN.

In addition to the short development time, the MOOSE framework allows the implementation of additional and higher fidelity physics to replace the existing physics as the need arises. As PRONGHORN is still a developmental code, it will be used to explore particular phenomena and scenarios not well captured by the more established core simulators. Near and medium term applications of PRONGHORN include:

1. *Coupled neutronic and thermal fluid behavior near the pebble bed core-reflector boundary*. As mentioned previously, the equation set used in most thermal fluid codes does not treat the variable porosity and heat transfer at the boundary between the pebble bed and solid reflector. Neutron diffusion theory is also known to be in error because of this change in composition and currently no diffusion code automatically treats the variation in the pebble density.

2. *Simulation of loss of forced cooling events with better representation of plant boundary conditions*. In order to accommodate the limited capabilities of existing core simulator codes, the PBMR400 Coupled Core Benchmark specified a simple constant pressure boundary condition at the core inlet for the pressurized loss of forced cooling exercise. This is an unrealistic assumption that may lead to significant errors in the trajectory of the predicted accident sequence and temperatures of the fuel and metallic vessel components. PRONGHORN does not share this limitation.

3. *Fuel performance and core dynamics*. The physics of fuel stress and core dynamics investigated using PASTA and CYNOD (Section 4.2.2.2) will be incorporated into the PRONGHORN code to yield high fidelity simulations of reactivity transients and their effects on TRISO durability.

4. *Water ingress*. Steam entering the core from a leak in the steam generator or shutdown cooler will cause a power spike (water is a moderator) in addition to transporting fission products through and perhaps out of the primary loop. The physics of this event involve neutronics, fluid flow and heat transfer, and fission product transport.

5. *Graphite dimensional changes*. Graphite shrinks and swells as a complex function of irradiation and temperature. These dimensional changes cause the bypass flow profile to change over time and with the attendant effects on core temperature. Coupling neutronics, thermal fluids and mechanics poses a particularly demanding computational problem and is one of the grand challenges of HTGR core simulation.

4.2.2.7 Uncertainty and Sensitivity Analysis

Quantification of the uncertainties in computed core physics parameters that result from propagation of uncertainties in the underlying nuclear data and other input parameters used in the various modeling codes is a key component of the quality assurance process for reactor physics modeling and simulation. It is also an important mechanism for quantifying the need for additional nuclear cross-section measurements and/or integral evaluations for HTGRs and as a guide in planning of future integral measurements and evaluations. Mathematically rigorous sensitivity and uncertainty analysis based on perturbation theory can, for example, be used to identify nuclides that contribute to calculational uncertainties and to quantify the propagated uncertainties in the context of the currently anticipated NGNP core designs. Sensitivity coefficients are calculated by generalized perturbation theory codes and folded with multigroup covariance data (where available) to derive propagated uncertainties in computed integral reactor parameters arising from the nuclear data. Integral parameters to evaluated include reactivity, peak power, reaction rate ratios, nuclide inventory, safety coefficients, etc. The impact of cross-section data uncertainty on the accuracy of each parameter is evaluated, along with the identification of nuclides, cross-section types, and energy ranges that have the greatest impacts on the accuracy of integral parameters. The process also can be used to rigorously quantify whether a given existing integral benchmark experiment is sufficiently similar to a contemplated NGNP system design to be of significant utility for validation of computations for the system being designed.

These rigorous perturbation techniques can be applied to neutronics calculations because the adjoint of the governing transport equation is available and can be manipulated numerically on a computer. The corresponding adjoint for the thermal fluid system of equations has yet to be derived for practical reactor thermal fluid problems. Variational theory therefore cannot yet be applied to the coupled core simulation problem, causing reliance on forward sensitivity techniques. In this approach input parameters are manipulated in a stochastic manner over their known and estimated range of variability. The effect on output parameters is computed and statistically correlated to the inputs to obtain sensitivity coefficients. Uncertainty bounds and confidence intervals can be obtained for key safety and performance parameters. The NRC has recognized a process of obtaining such parameters (the Code Scaling, Applicability, and Uncertainty or CSAU process) and has accepted some boiling water reactor (BWR) LOCA uncertainty analyses. Although considerable R&D is being expended in this area, the methods are still considered developmental. A very limited number of computer codes are available for performing such analyses, one of which is the Software for Sensitivity and Uncertainty Analysis code developed by the GRS (Gesellschaft für Anlagen und Reaktorsicherheit) company of Germany. SUSA envelopes the user’s analysis code, manipulates the user-supplied input parameters over the specified ranges, executes the minimum number of runs to generate a certain confidence value in the output uncertainty, and computes and ranks the output sensitivities. This software has been acquired by the Project and is being evaluated. Applications to core simulation will commence in FY 2011.

So far the SUSA software has been applied mainly to core thermal fluid simulations such as BWR large break LOCA analysis and, in a preliminary investigation, to the calculation of uncertainty in peak fuel temperature in the PBMR400 after a large pipe break (Strydom 2010). Figure 30 shows the assumed distribution in the core thermal power and the inlet temperature (top) and the effects of these on the peak fuel temperature attained while the core cools off.

The nature of the tool, however, allows it to be applied to the analysis of any parameters for which a computational model can be constructed. SUSA can be applied to fission product transport models (e.g., MELCOR), fuel performance models (e.g., PARFUME), tritium migration (TPAC), etc.

The ultimate value of the uncertainty analysis capability may be in influencing design decisions early in the process. There are numerous flow and heat transfer unknowns, material properties, behavior under irradiation, manufacturing tolerances, and stochastic phenomena that feed into the overall uncertainty in various safety and performance parameters. A comprehensive effort to reduce all possible uncertainties in an effort to minimize margins would consume more resources than are available to the project. It would be, in effect, a strategy for full employment at the national labs while delaying indefinitely the final design process.

Instead, the SUSA tool can be applied to the evaluation model to compute and rank the sensitivity of a plant safety or performance parameter to any of a number of design and material inputs as shown in Figure 31. Those parameters found to have a significant impact on margin could become the focus of further experimental investigation, or provoke design changes by the vendor in the preliminary design phase. The remaining inputs could be neglected and their uncertainties safely subsumed into to overall safety margin.

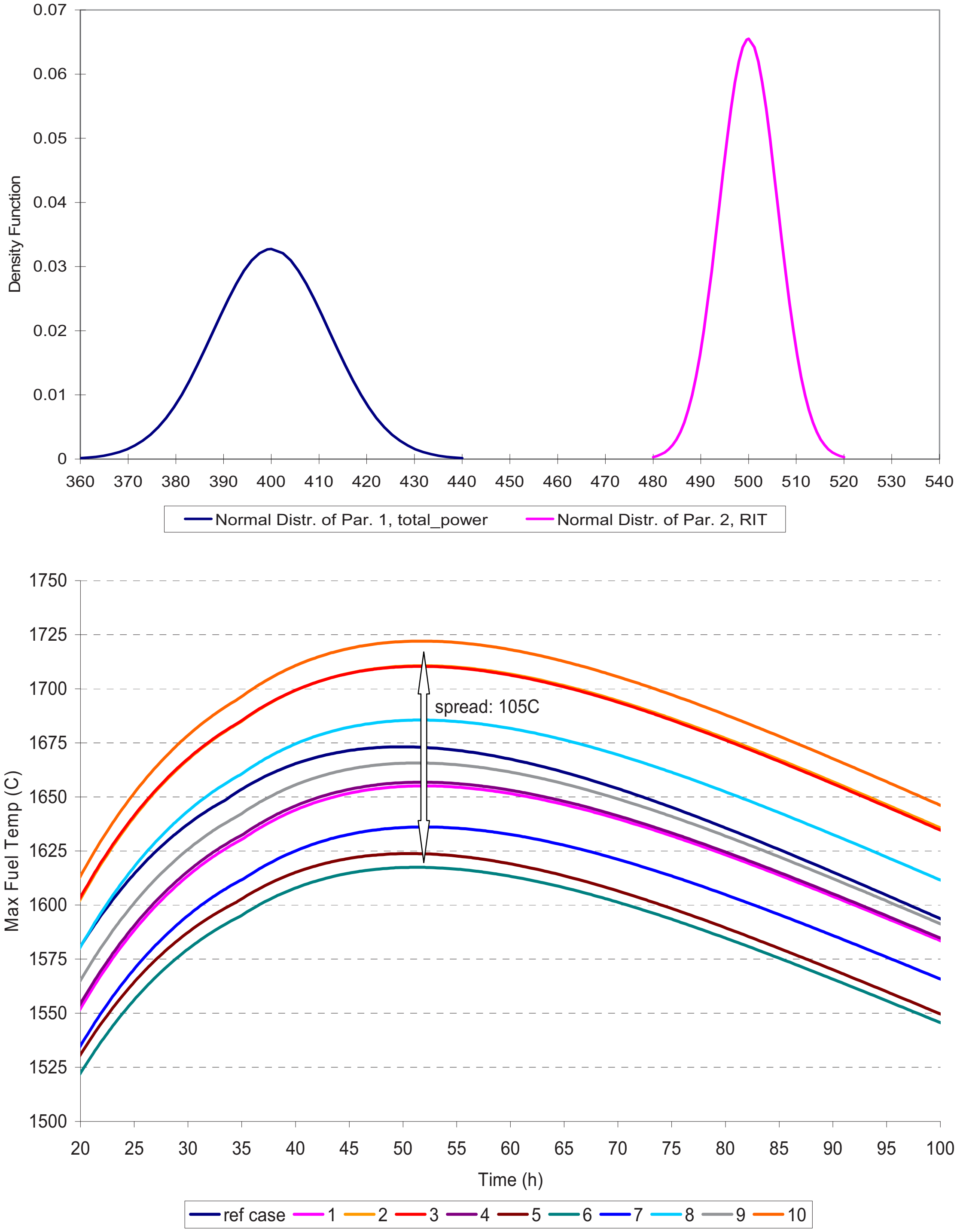


Figure 30. Input and results from a PEBBED analysis of Oeak temperature during a PBMR400 depressurized conduction cooldown: distributions of core power and inlet temperature (top); distribution in peak fuel temperature (bottom).

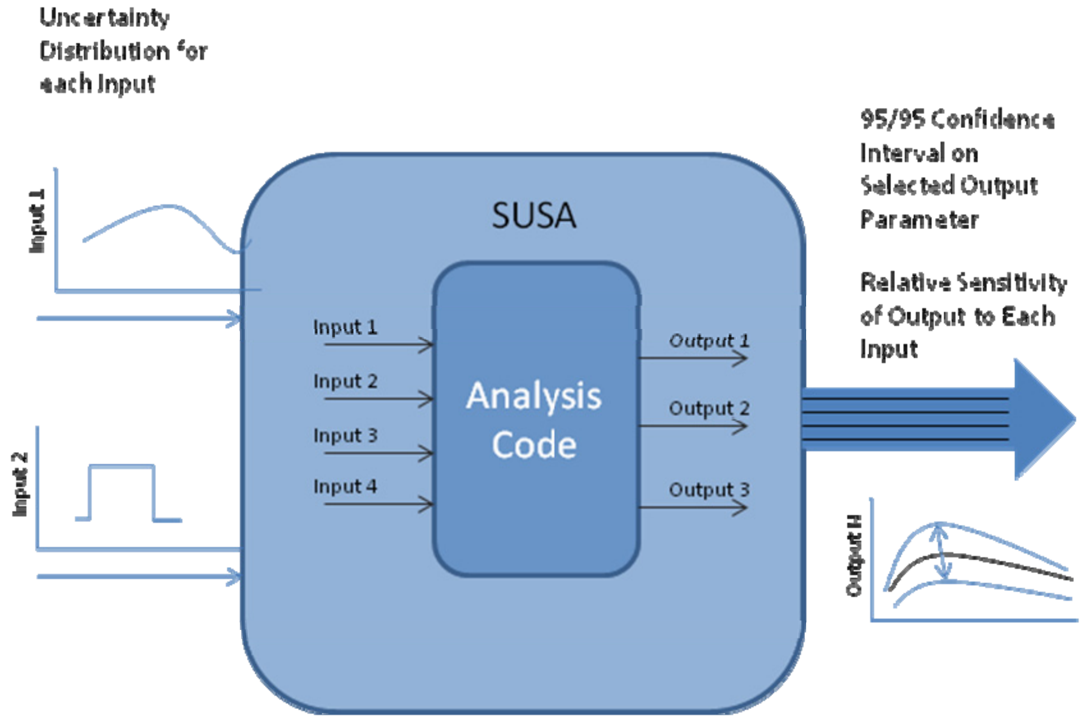


Figure 31. Information flow in a SUSA analysis.

4.2.3 Accomplishments and Status

By the end of FY 2010, the following tasks will have been accomplished.

- Upgrade the COMBINE code to perform multiscale homogenization and generation of cross sections for pebble bed core simulation and integrated the code into the PEBBED-THERMIX PBR fuel cycle analysis code.
- Develop and code a 3-D transport treatment of control rods in the pebble bed reactor to support CYNOD simulations of PBR transients.
- Complete and test a pebble dynamics simulation tool to support PBR burnup analysis, earthquake simulation, and dust production studies.
- Complete and test the integration of the CYNOD PBR nodal kinetics solver into the RELAP system analysis code.
- Perform simulations of earthquake-induced PBR transients using a high-fidelity time-dependent fuel temperature model and a discrete element pebble dynamics.
- Complete the CRP-5 Benchmark Evaluation of the PBMR400 Equilibrium Cycle using the PEBBED-COMBINE-THERMIX code package.
- Complete evaluations of the HTTR and Proteus Reactor Physics Benchmarks using MCNP.
- Complete development of a analytical treatment of burnable poisons in prismatic fuel blocks and integrated the treatment into a nodal diffusion solver

- Complete the 3D analysis of the HTTR critical core using deterministic core analysis tools as a validation and verification of methods and a step toward full core burnup and transient analysis.
- Complete most of the PBMR 400 Transient Benchmarks using the PRONGHORN multiphysics core simulator developed on the MOOSE platform.
- Acquire and tested the SUSA software for uncertainty and sensitivity analyses.

The following task are planned for FY 2011 through FY 2013:

- Develop, code, and verify a nodal depletion algorithm to support prismatic core burnup and fuel management analyses.
- Develop a fast an accurate online cross section generation capability to support space-dependent HTGR transient simulations.
- Integrate the nodal diffusion solvers (CYNOD and HEXPEDITE) with a 3-D thermal fluid core and plant simulators to support 3-D transient analysis of pebble bed and prismatic reactors.
- Conduct investigations of PBR wall heat transfer and other complex neutronic/thermal-fluid phenomena using the multiscale, multiphysics capabilities of PRONGHORN.
- Verify the pebble motion and dust generation simulation capabilities of PEBBLES and validate against AVR or other experimental data.
- Finalize the HTTR and Proteus reactor physics benchmark evaluations.
- Analyze fuel behavior during HTGR transients using a coupled fuel performance code (PASTA) and core simulator (CYNOD and HEXPEDITE).
- Lead the specification for a prismatic core transient benchmark and complete the evaluation.
- Complete and test upgrades to the MELCOR code in support of dust and fission product transport studies.
- Compute the uncertainties in peak fuel temperatures and shutdown margin because of uncertainties in core parameters.
- Support experimental campaigns in thermal validation and verification and fuel and graphite validation and verification.

5. PROGRAM SCHEDULE AND COST

A detailed resource-loaded activity-based schedule for the activities presented in the technical program plan for experimental V&V and core simulation activities has been developed and is used to guide and prioritize activities year by year. A higher-level summary of that schedule is shown in FIGURE 32. The critical path for experimental validation of system and CFD codes is through the construction and operation of the High temperature Test Facility and Natural Circulation Shutdown Heat Removal Test Facility, with simultaneous experiments in air ingress and bypass flow. Activity then shifts to core heat transfer, plenum-to-plenum, and the other experiments defined in the plan. HTTF and NSTF are the more complex integral facilities that will require considerable construction and shakedown testing before data is generated. Activities in Core Simulation initially focus on basic neutronics modeling (development of acceptable cross section generation and neutron transport techniques) along with the establishment of a bounding range of core transient exercises to be used in benchmark studies.

Based on the schedule, the high priority experiments for the prismatic reactor will be completed by 2013 with the remainder to be completed by 2016. Likewise the core simulation tasks will follow a similar high level schedule.

A itemized cost breakdown is shown by task below with the experimental V&V tasks followed by the core simulation tasks. The total program cost is estimated to be ~ $71 M.

- Integral tests in HTTF **($15M)**
  - NRC Prismatic ($8M)
    DOE Prismatic ($2M)
    DOE Pebble Bed ($5M)
- Ex-Core Heat Transfer in NSTF **($2.5M)**
- Air Ingress **($5M)**
- Fission Product Transport **($5M)**
- Water Ingress **($10M)**
- Bypass in MIR **($4.5M)**
- Core Heat Transfer **($2M)**
- Plenum-to-Plenum **($2M)**
- Lower Plenum **($10M)**
- HTTR/HTR-10 Data Acquisition **($3M)**
- Differential Cross Section Measurements **($4M)**

TOTAL **$71M**

The schedule of high level experimental thermal fluid V&V tasks to support at 2013 license submittal is presented in Figure 32.

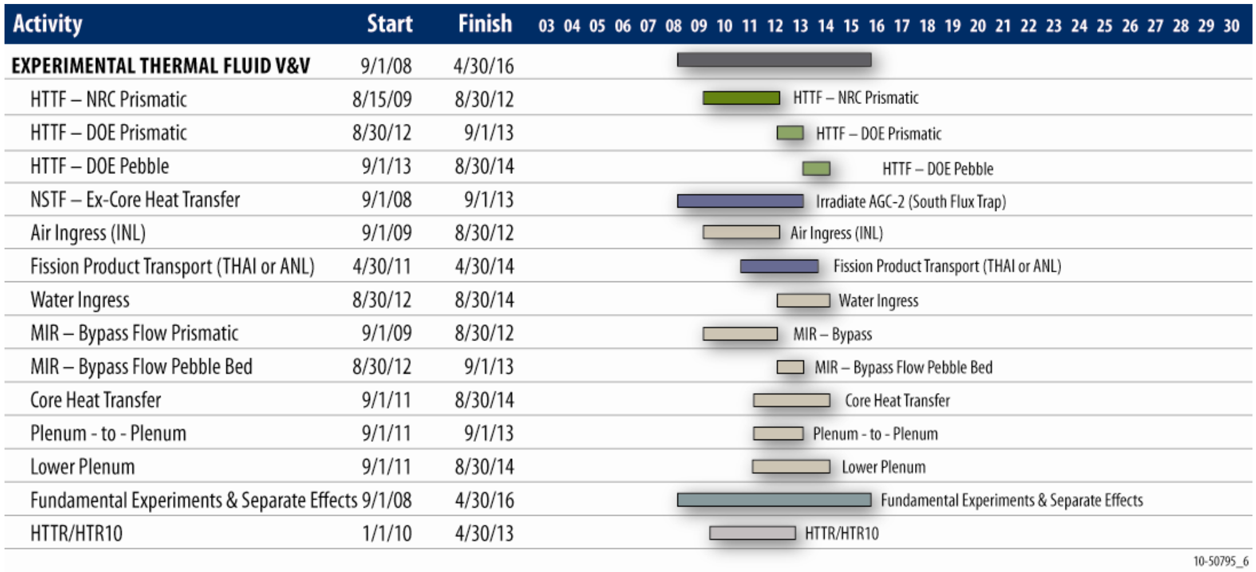


Figure 32. High level schedule of experimental thermal fluid V&V tasks to support at 2013 license submittal.

Estimates of cost and schedule for the core simulation development and analysis campaign are as follows:

- Analysis of Pebble Bed Coupled Code Transient Benchmarks **($1M)**
- Transient-Driven Fuel Performance Simulation of a PBR **(1M)**
- Uncertainty Analysis in Fuel Temperatures and Shutdown Margin in the PBR **($0.5M)**
- Development of PRONGHORN and Analysis of Pebble Bed Transient Benchmarks **($2M)**
- PRONGHORN Analysis of Wall Heat Transfer during a PLOFC **($1M)**
- Multiphysics Analysis of a PBR and Prismatic PLOFC with Constant Coolant Inventory
- Specification and Execution of a Prismatic Transient Core Benchmark **($3.0M)**
- Analysis of Effects of Reactivity Effects on Fuel Performance in a Prismatic Core **($1M)**
- Development of BIGHORN for Analysis of Prismatic Transient Benchmarks **($2M)**
- Demonstration of Fuel Cycle Analysis Capability with Depletion Benchmarks **($2M)**
- Validation of Pebble Motion and Dust Production Model **($1M)**
- Validation of and Analysis with a Dust/Fission Product Transport Model **($3M)**
- Uncertainty Analysis in Fuel Temperatures/Shutdown Margin in the Prismatic Core **($0.5M)**
- Multiphysics Analysis of Graphite Mechanics under Irradiation and Fluid Flow **(1M)**

TOTAL **$19M**

Activity	Start	Finish
CORE SIMULATION ACTIVITIES	**10/1/07**	**9/30/13**
PBR Core Simulation	10/1/07	9/30/11
Analysis of PBMR400 Coupled Code Transient Benchmark	10/1/07	4/29/11
Transient-Driven Fuel Performance Simulation of a PBR	10/1/08	4/29/11
Analysis of Uncertainty in Peak Fuel Temperature in the PBR	10/1/09	9/30/11
Multiphysics Analysis	10/1/08	9/30/13
PRONGHORN Execution of PBMR400 Coupled Code Transient Exercises	10/1/08	9/30/10
PRONGHORN Analysis of Wall Heat Transfer in a PBR	10/1/10	9/30/11
Multiphysics Analysis of a PBR and Prismatic PLOFC with Constant Coolant Inventory	10/1/10	9/28/12
Multiphysics Analysis of Graphite Mechanics under Irradiation and Fluid Flow	10/3/11	9/30/13
Prismatic Core Simulation	10/1/09	4/30/13
Specification of Set of 3D Coupled Core Transient Exercises for a Reference Prismatic Core	10/1/09	4/29/11
Execution of the 3D Coupled Core Transient Exercises for Prismatic Core at BOC	10/1/09	4/30/12
Transient-Driven Fuel Performance Simulation of a Prismatic Reactor	4/1/11	9/28/12
Completion of Analysis of Uncertainty in Peak Fuel Temperature Exercises for Prismatic Core at EOC	10/1/09	4/30/13
Development and Testing of the Nodal Depletion Solver	10/1/09	10/3/11
Execution of the 3D Coupled Core Transient Exercises for Prismatic Core at EOC	10/4/10	4/30/13
Other	10/1/07	9/30/11
Execution of a Primary Coolant Break Model with Dust Dispersion	10/1/09	9/30/11
Validation of Pebble Motion and Dust Generation Model of the Reference PB-HTR	10/1/07	9/30/11

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Figure 33. High level schedule of core simulation activities to support a 2013 license submittal.

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Appendix A
Qualification, Selection, Validation, and Verification of Codes and Models

Practices and procedures are divided into several categories to indicate the goal and intent of each. These categories include Scenario Identification, Code Verification, Code and Calculation Documentation, Reduction of Numerical Error, Quantification of Numerical Uncertainty, and Calculation Validation. Quantification of numerical uncertainty is discussed in some detail in Johnson et al 2006. A more detailed explanation of each of these concepts is provided in the Appendix.

A-1. SCENARIO IDENTIFICATION

Practices and procedures are divided into several categories to indicate the goal and intent of each. These categories include Code Verification, Code and Calculation Documentation, Reduction of Numerical Error, Quantification of Numerical Uncertainty and Calculation Validation. Quantification of numerical uncertainty is discussed in some detail in Johnson et al 2006. Validation, including calculation validation, is discussed in Section A-3.4.

A-2. CODE VERIFICATION

Code Verification involves the determination of coding correctness, Roache [1998], a process separate from Calculation Verification (the Quantification of Numerical Uncertainty). Idaho National Laboratory (INL) recognizes that the analysis software that will be applied to reactor safety analysis will already have been subjected to a variety of Code Verification tests. What will be required is documentation of these tests.

Only those tests that exercise the options used in the particular computations need to be documented. The tests as a suite must be designed to exercise all the terms in the governing partial differential equations. For example, it is not adequate to only test the code on linearly varying solutions such as planar Couette flow, since this solution does not exercise vertical convection terms and others. The most complete and convincing type of Code Verification test uses the Method of Manufactured Solutions [Roache 1998], but this will not be required. If Method of Manufactured Solutions not used, it will probably be necessary to use a suite of test problems to demonstrate code correctness. For all of these problems, the observed rate of discretization error convergence should be documented and compared to a theoretical value for the discretization algorithms employed. If it is not, then more stringent requirements will be enforced during Calculation Verification (Johnson et al 2006). The Code Verification must also include some data on the effect of iterative convergence criteria on numerical results. (See “Reduction of Numerical Error” below for details.)

A-2.1 Code and Calculation Documentation

Software that is used for nuclear reactor safety analysis must be described in detail in code documentation. Such documentation should include describing equations used and their discretization as well as the basics methods used to obtain a solution. The truncation error and its formal order or accuracy should be given. The code documentation must include all details of implementation of the turbulence models used in calculations, e.g., turbulence models for computational fluid dynamics (CFD) software.

The code documentation should be available for reference to reviewers who must review the associated calculational results.

For each calculation performed and submitted as a safety analysis, assumptions must be listed along with the details of the methods and models used. Other details including, but not limited to boundary and initial conditions, model constants (parameters) and other relevant information must also be provided. Options not used in the calculation need not be documented.

A-2.2 Reduction of Numerical Error

The reduction of numerical error is clearly a desirable objective for numerical calculations. Lessons have clearly been learned about what not to do when using computational techniques for numerical analysis. These have been canonized in the requirements for manuscripts submitted to well-known journals, such as the American Society of Mechanical Engineers (ASME) Journal of Fluids Engineering. It therefore seems prudent to apply them to the application of relevant software to reactor safety analysis. Examples of such requirements are those given in the ASME Journal of Fluids Engineering "Statement on Numerical Accuracy." Details regarding the philosophy and meaning of the various key points are discussed in Johnson et al 2006. Examples of the content include requirements that: (a) methods must be at least second order accurate in space, (b) grid independence or convergence must be established, and (c) in transient calculations phase error must be assessed and minimized.

Grid independence is the process of refining the grid from some the starting point until numerical results stop changing or change by negligible amounts. Theoretically, the results will continue to change until the grid spacing approach zero. The precision of the machine, however, will halt this process at a finite grid spacing. This is sometimes referred to as achieving machine zero (of the residuals). Not only is the process of obtaining grid independence important to reducing numerical errors, it is also a good way to obtain estimated of numerical uncertainty (see "Quantification of Numerical Uncertainty" in Johnson et al 2006).

Iterative convergence relates to the number of iterations required to obtain residuals that are sufficiently close to zero either for a steady-state problem, or for each time step in an unsteady problem. This error is in addition to the numerical error associated with the truncation error terms. Because of the well-known and unacceptable sensitivity of some commercial codes to the iteration tolerance, and the too lax default tolerance, the final calculations must determine this effect. At least two levels of iteration tolerance must be shown and the sensitivity presented. For example, if results for a solution functional f are presented using a default iteration tolerance of (say) 10-3 reduction in residual from the initial condition, as required in Freitas et al [2003], then another calculation with 10-4 will be required, and the sensitivity f'10 will be stated as the normalized % change in f per decade of change in iteration tolerance.

$$f_{10}^{'} = \Delta f = [f(10^{-4}) - f(10^{-3})] / f_{norm} \times 100\%$$

The normalization can be based on f norm = f(10-3) when divides by near zero are not a problem, otherwise by another appropriate normalization. The final test of sufficient tightness of the iterative tolerance will be the acceptability of the final results based on estimation of numerical uncertainty and validation metrics.

For transient calculations, the same convergence criterion should apply as for spatial convergence (grid independence). The time step should be refined until negligible change is obtained. Also, though not required by the Journal of Fluids Engineering, it is recommended that the time-wise discretization scheme should be second-order accurate or better. While there are other practices to reduce numerical error, the above will constitute the required practices for rector safety analysis at the present time. Certainly, other practices that reduce numerical error are allowed and even encouraged.

A-3. VALIDATION

Whether or not software is adequate for performing best-estimate very high temperature reactor (VHTR) analyses is determined using both “top-down” and “bottom-up” evaluations, as summarized in Figure A-1 and described in the following sections.

A-3.1 “Bottom-Up” Code Adequacy

Bottom-up evaluation of code adequacy entails examination of four features: the pedigree, applicability, fidelity, and scalability of the code under consideration.

The pedigree of a systems code consists of its history, its development procedures, and the basis for each correlation that is used in the code. Any correlations, data sources, and approximations used in the code must be documented, e.g., in textbooks, laboratory reports, papers, etc. The uncertainty data used to bound the correlation(s), data, and approximations must be included in the documentation, e.g., instrumentation uncertainty, data system uncertainties, etc. The basis for the uncertainties should be traceable and reproducible. The assumptions and limitations of the models must be known and documented.

The applicability of a systems code depends on the range of use of each of its correlations, data, and approximations. Those ranges must be documented and referenced. Finally, the range of applicability claimed in the code manual should be consistent with the pedigree—or if a greater range is claimed then the justification for the increase in range must be reported.

The fidelity of a systems code means the degree to which the code’s predictions agree with physical reality. High fidelity requires that the mathematical models and correlations used in the code are not altered in an ad-hoc manner from their documented formulation. A code is validated when it is shown that the code’s predictions of key parameters agree within allowable tolerances with experimental data. The validation effort should be complete for all the key phenomena in the events of interest. Finally, benchmarking studies may either supplement the validation effort or make up the validation effort if appropriate standards are available, e.g., comparison of code calculation with a closed form solution.

“Bottom-up” scaling stems from the need:

- To build experimental facilities that model the desired full-scale system
- To closely match the expected behavior of the most important transient phenomena in the scenarios of interest
- To demonstrate the applicability of data from a scaled facility to a full-scale system and to defend the use of data from a scaled facility in a code used to calculate the behavior of a full-scale system
- To relate a calculation of a scaled facility to a calculation of a full-scale system.

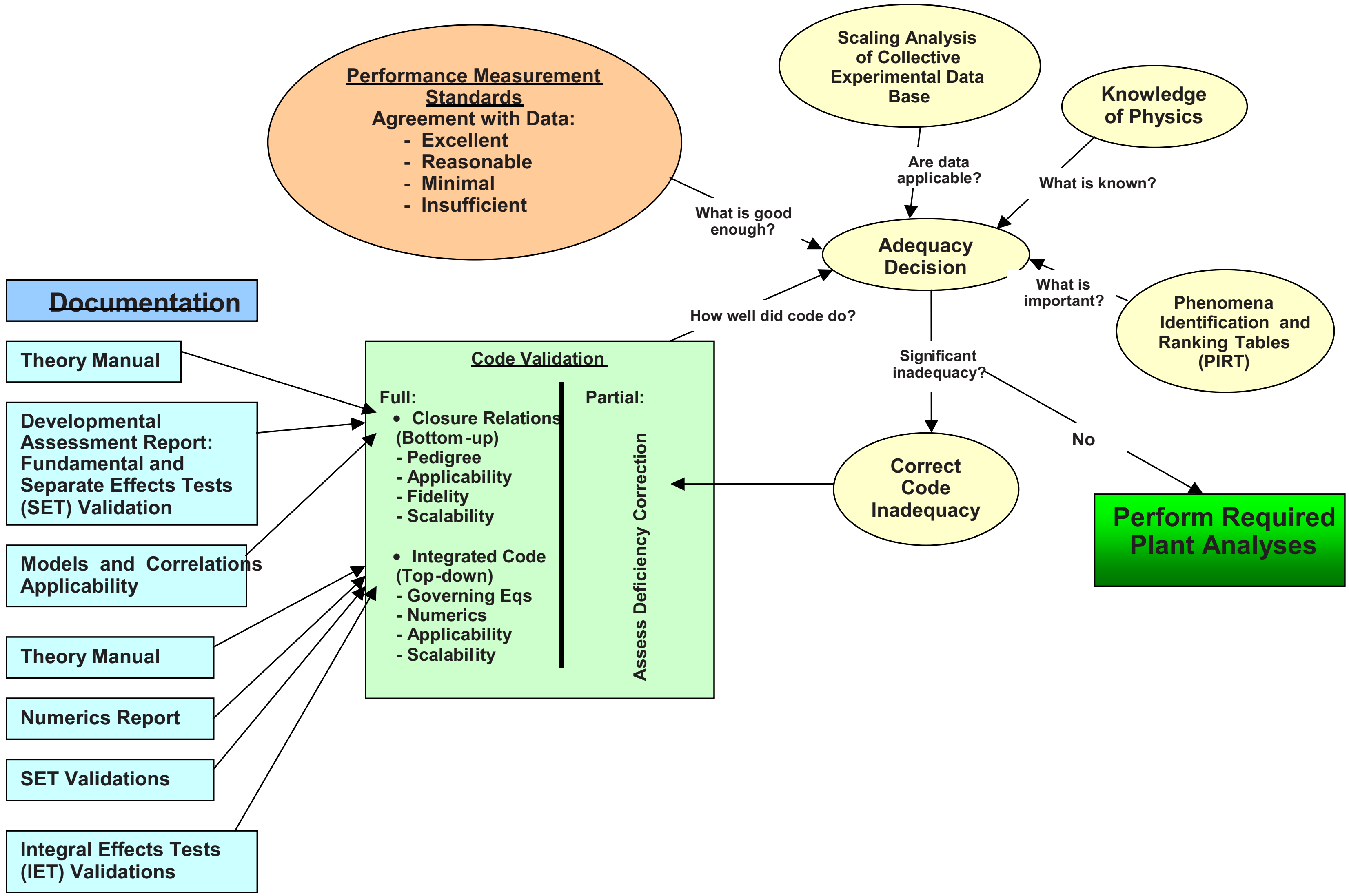

Figure A-1. VHTR system design software: elements of adequacy evaluation and acceptance testing practices.

Usually, scalability studies are performed to scale key parameters for a portion of the system behavior—not to correlate the global system behavior. Therefore, scalability analyses consist of four steps: (1) isolate the "first-order" phenomena, (2) characterize the "first-order" phenomena, (3) convert the defining equations into nondimensional form, and (4) adjust the experimental facility conditions to give equivalent behavior with the full-scale system within the limitation of the facility (or nearly equivalent, i.e., based on non-dimensional numbers that follow from step 3).

As implied in the above discussion, "bottom-up" code adequacy techniques focus principally on closure relationships. Thus, the field equations used in the code must be correctly formulated and programmed. In addition, the field equations must be reviewed by the scientific community, and its agreement on the correct formulation and insertion of the governing equations in the code must be obtained.

A-3.2 "Top-Down" Code Adequacy

The "top-down" approach for ensuring code adequacy focuses on the capabilities and performance of the integrated code. The top-down approach consists of four parts: numerics, fidelity, applicability, and scalability.

- *Numerics*. Evaluation of the numerical solution considers (i) convergence, (ii) stability, and (iii) property conservation[f]. Again, agreement by scientific community on acceptable convergence, stability, and property conservation must be obtained.
- *Fidelity*. The fidelity of the code is demonstrated by performing thorough code assessments based on applicable integral-effects and separate-effects data. The data are part of an agreed-upon code assessment matrix constructed based on the transients of importance and the key phenomena for each phase of the transients.
- *Applicability*. The code must be shown capable of modeling the key phenomena in the system components and subsystems by conducting thorough validation studies. The key phenomena are identified in the Phenomena Identification and Ranking Table (PIRT).

  The method to determine whether the code is capable of modeling key phenomena is to compare the calculation produced by the code to data that have known uncertainties. For example, "excellent" agreement between the code calculation and data is achieved if the calculated value is at all times within the data uncertainty band.

  The degree of agreement between the code calculation and the data is generally divided into four categories as given in Table A-1. A more rigorous definition is given in Schultz 1993. A code is considered adequate in applicability when it shows either excellent or reasonable agreement with the highly ranked phenomena (sometimes identified as the dominant phenomena) for a transient of interest. If the code gives minimal or unacceptable agreement, then additional work must be performed; the work may range from additional code development to additional analysis to understand the phenomena.

f. Property conservation issues arise when two calculations of the same property are performed by a systems code using two different algorithms or methods. This practice may follow in an effort to enhance the accuracy of the code result. Because the two methods are likely to calculate slightly different values of the same property, e.g., pressure, property conservation must be considered.

Table A-1. Code adequacy identifiers

Classifier	Description
Excellent	The calculation lies within or near the data uncertainty band at all times during phase of interest.
Reasonable	The calculation sometimes lies within the data uncertainty band and shows the same trends as the data. Code deficiencies are minor.
Minimal	Significant code deficiencies exist. Some major trends and phenomena are not predicted. Incorrect conclusions may be drawn based on the calculation without benefit of data.
Unacceptable	A significant difference between the calculation and the data is present—and the difference is not understood. Such a difference could follow from errors in either the calculation or the portrayal of the data—or an inadequate code model of the phenomenon.

- Scalability. Experimental scaling distortions are identified and isolated, e.g., inappropriate environmental heat losses that stem from the larger surface-to-volume ratios that are inherent to scaled facilities. Finally, an effort to isolate all code scaling distortions is performed through the code assessment calculations. Scaling distortions may arise from an inappropriate use of a correlation developed in a small-scale system when applied to a full-scaled system.

A-3.3 Validation Process

Validation of the analysis tools, for example the systems analysis and CFD software, will proceed using a process designed to include the expertise in not only the nuclear industry but the expertise external to the nuclear industry when required. Participation by experts at the national laboratories together with university experts and industry experts will ensure the software tools achieve the defined objectives.

The process is centered on defining a validation matrix which serves as the foundation for a set of standard problems for both systems analysis and CFD software. The validation matrix is assembled by correlating the key phenomena identified in the PIRT for the most challenging scenarios with the available data sets. If data sets are needed, but not available, then experiments will be designed and performed to provide the needed data. The experiments will be specified to meet the standard required for software validation, that is, with a reasonable uncertainty band and with a data range that either includes the required validation range or can be scaled include the required validation range. Subsequently, the data sets become the basis for standard problems that will be used by the VHTR validation community. The VHTR validation community consists of users in Boxes 2 through 5 in Figure 3, i.e., the national laboratory users, the university community, the Generation IV International Forum (GIF) community (including the vendor community), and the community of users (who may be outside the nuclear community) for the software being validated. This process is shown in Figure A-2.

The process for specifying standard problems begins with the formation of a “Standard Problem Committee” (presently being formed through the auspices of the GIF Methods Project Management Board). The Standards Problem Committee will consist of members from selected universities, the VHTR Program Group, GIF organizations, and the vendors. The standard problems will be defined to meet a prescribed standard.

The validations performed using standard problems will be assigned to those who will perform validation exercises by the "Standard Problem Oversight Committee" who will also formulate the practices and procedures that will be used for performing the validation calculations. This committee will be composed of experts in the use and validation of the software. For example, one of the committee members for the CFD Standard Problem Oversight Committee will be an expert selected by the ASME CFD Technical Committee, which was previously responsible for the well-known CFD Triathlon in the 90s, who will not necessarily be a member of the nuclear community. Other members will be from the VHTR Program Group, universities, and perhaps the vendors. Following assignment of the standard problems to the participants, the committee will also oversee the final review and publication of the validation studies in the literature as shown in Figure A-2.

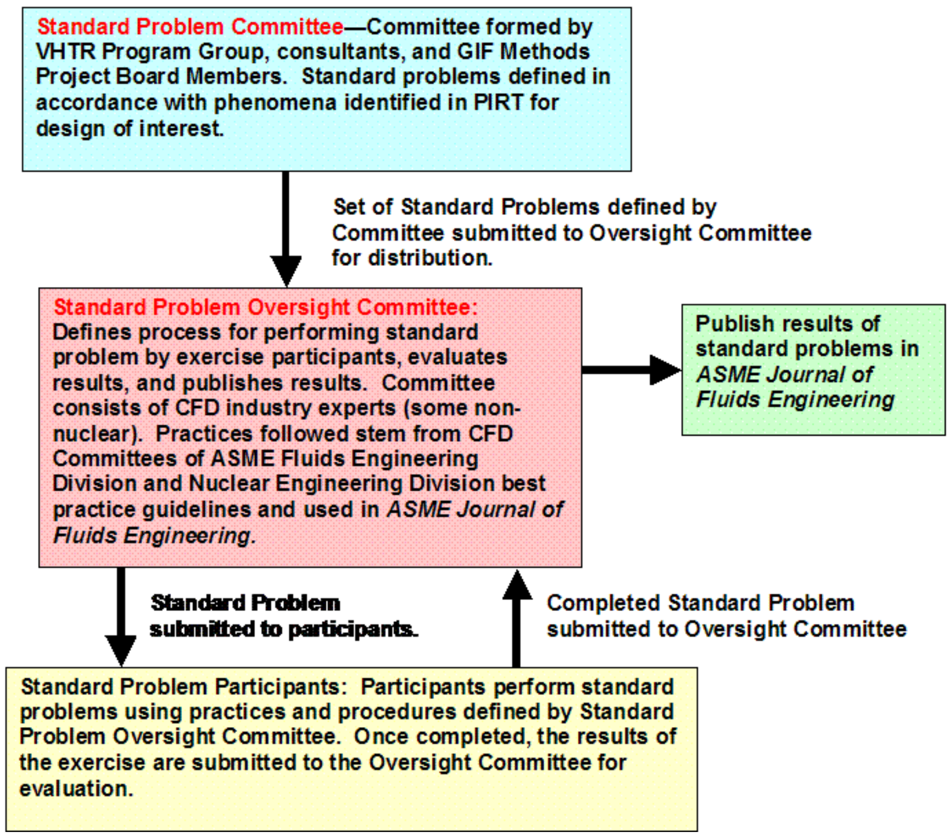

Figure A-2. Validation process—including participation by experts from the national laboratories, universities, vendors, and the community specific to software undergoing validation.

A-4. SOFTWARE TOOL SELECTION AND SOFTWARE DEVELOPMENT

A-4.1 Software tool selection

When confronted with the need to calculate some of the phenomena that will be encountered in the VHTR scenarios, it is inevitable that analysts will be required to choose one software tool over another. This will be particularly true of systems analysis software (for example, GRSAC, MELCOR, and RELAP5—see Figure 8). To assist the analyst in formally choosing software, a methodology is given in Figure 11 where a flow chart summarizes key factors and questions such as:

1. The phenomena or scenario that requires analysis, as identified in the PIRT. Has the software ever been used to analyze the phenomena or scenario? By answering this question the analyst may be introduced to references and other experts who have applied the software to similar phenomena or scenarios. Hence a body of useful information may be available.

6. Are the phenomena modeled properly? And does the model region of applicability correspond to the system phenomena or scenario envelope? These questions may be most easily answered by using the required manuals and documentation identified in Figure 9, e.g., models and correlations, theory manual, scaling relationships and applications, developmental assessment reports, validations, etc.

7. Have validation studies been completed for the phenomena or scenario? Were the validation results reasonable or excellent (as defined in Table A-1)—or were the results minimal or unacceptable? If a body of validation results are not available, or if the validation results were not "reasonable" as a minimum, then either the software should not be used or it should be validated to ensure that the calculated results are reliable rather than misleading.

Only when acceptable answers are obtained for the questions listed above, can the software under consideration be used for the required analysis with confidence.

A-4.2 Software Development

The Methods Research and Development Program is geared to principally make use of existing software unless it can be shown that the present capability is inadequate for designing and licensing the VHTR. Presently, for the thermal-fluids analysis needs, it is projected that only a few additional CFD and systems analysis code development efforts will be required. For reactor physics, some developmental efforts are required to accommodate the specific fuel and material types. However, every effort is being made to complete the required reactor physics software development as early as possible in the program.

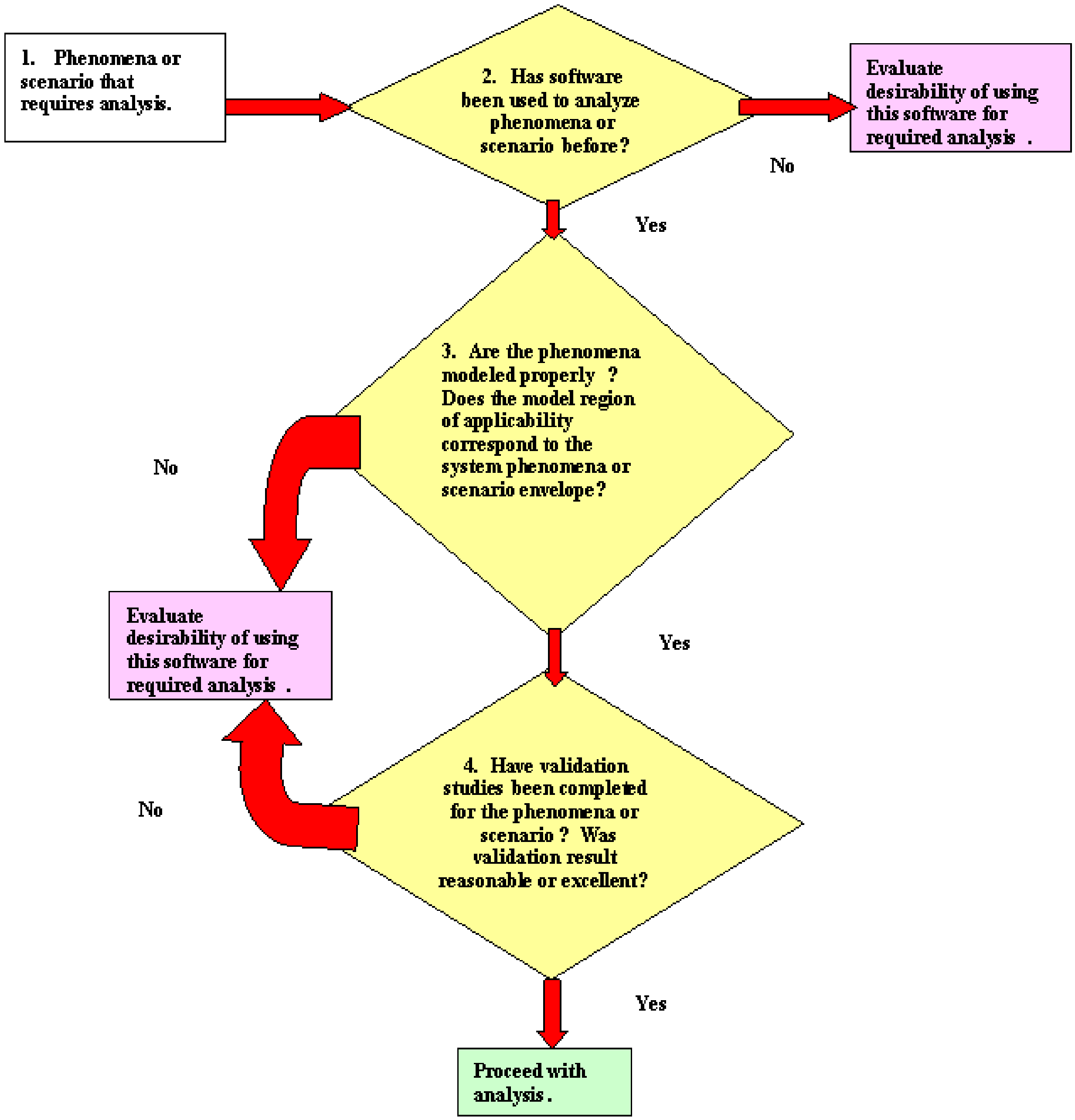

Figure A-3. Flow chart to evaluate applicability of analysis software.

A-5. REFERENCES

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