High-Temperature Gas-Cooled Reactor (HTGR)
NRC Research Plan

Purpose

The purpose of the U.S. Nuclear Regulatory Commission (NRC) High-Temperature Gas-Cooled Reactor (HTGR) Research Plan is to aid the Office of Nuclear Regulatory Research (RES) technical staff in ensuring that the planned research activities for the Department of Energy proposed HTGR [also referred to as very high temperature reactor (VHTR) and Next Generation Nuclear Plant (NGNP)] directly support NRC's goal of developing the required models and acquiring the data to perform an independent confirmatory safety analysis of an HTGR design. Furthermore, the plan will help ensure that the NRC work complements research and development (R&D) activities conducted at other institutions nationally and internationally. In addition, the research plan ensures that the products can be developed in a timely manner to support the user offices (primarily, the Office of New Reactors or NRO) in conducting a licensing review for the proposed NGNP. The plan also provides RES and NRO management a planning and decision-making tool for prioritizing HTGR research activities and for formulating budgets. The plan replaces the previously developed Advanced Reactor Research Plan (ARRP)[1], which henceforth will serve as the technical background for the current document.

Scope

The R&D plan covers seven major technical disciplines unique to HTGR:

- Plant safety analysis including thermal-fluids and accident analysis
- Nuclear analysis
- Fuel performance and fission product behavior
- High temperature materials performance
- Graphite performance
- Safety issues related to process heat applications
- Structural analysis (with particular focus on high-temperature effects)

In addition, the plan covers several other technical disciplines where ongoing or planned generic R&D for light-water reactors (LWRs) will be applicable to HTGRs with appropriate modifications. These disciplines include:

- Instrumentation and control for high-temperature applications
- Human factors and human reliability analysis
- Probabilistic risk assessment (PRA) and risk-informed infrastructure development

The nature and scope of R&D in some of these technical disciplines will be better defined in the future as NRC identifies policy or key technical issues that may warrant R&D activities as part of their resolution. Possible examples include instrumentation and control for high-temperature applications and human factors and human reliability analysis.

[1] ADAMS accession number ML082590538.

The plan does not cover the following areas that are or will be addressed separately by NRO and/or other offices:

- Spent fuel storage and transportation
- Security and safeguards
- Regulatory guidance development
- Identification and resolution of policy issues

Assumptions

The following assumptions are made in the development of this plan:

- NRC will take full advantage of the HTGR (or VHTR) R&D sponsored by the U.S. Department of Energy (DOE), in particular those activities dealing with generation of experimental data.

- NRC will explore other international HTGR R&D programs (e.g., HTR-10, HTTR, RAPHAEL, EUROPAIRS, EU 7th Framework, etc.) again, particularly those activities dealing with generation of experimental data.

- NRC will rely on an applicant furnishing adequate data to make the safety case and to support the design.

- NRC will explore limited experimental programs on its own only in those situations where the applicant is not required to provide this information and it is not available through any other channel. An example would be plant response to a beyond licensing basis event not considered in the applicant's probabilistic risk assessment model.

- NRC will depend on international and national codes and standards bodies to develop approved codes and standards or to modify existing ones, as appropriate.

The outlet temperature (similar to core exit temperature in a light-water reactor) of an HTGR can vary from as low as 650^{o}C to 950^{o}C or above. The term very high temperature reactor (VHTR) is often applied to HTGR designs with outlet temperatures above 800^{o}C. The planned HTGR research described herein covers a wide range of outlet temperatures, including those corresponding to VHTR designs, an example of which is the NGNP. The high outlet temperature presents a number of challenges in the design and safety performance, particularly those associated with fuel and materials. The ultimate choice of outlet temperature can substantially affect safety margins and a broad range of safety related technical issues. Details of the planned research may thus be revisited and refined as outlet temperatures and other major design choices become more clearly defined by DOE or an applicant. The current trend in the NGNP conceptual designs is to focus on the outlet temperature around 750^{o}C so as to take advantage of existing code qualified material properties and performance databases. Accordingly, the scope of initial R&D activities described in the plan, primarily in the materials area, is predicated upon the choice of reactor outlet temperatures near the lower end (around 750^{o}C) of the outlet temperature range discussed in this paragraph.

Two basic designs of HTGR are considered in scoping the research plan—the pebble-bed reactor and the prismatic core reactor. Lacking the down selection to a single reactor type, the planned NRC safety analysis tools will cover both designs in as generic a manner as practicable. However, at some point in the future, the plan will need to be revisited when the selection is made.

The NRC's HTGR research plan is predicated on the congressionally mandated timeline for NGNP[2] (i.e., design and construction of a prototype NGNP with fuel loading and operation to begin in 2021). This translates to a construction start in 2017 and the license application submittal in 2013, the latter also being the timeline for NRC to be ready with its confirmatory safety analysis tools. In turn, the 2013 date for NRC preparedness is predicated on certain HTGR design and material performance data to be ready ahead of this date. If a slippage occurs in the timeline for producing the needed data and information, it will adversely impact the assessment and validation of confirmatory analysis tools. In this case, additional conservatism may be needed in the safety analysis in consideration of the consequent larger uncertainties. As more data become available in the future and tools are better validated and assessed, conservatism in analysis can be correspondingly reduced. Also, compensatory measures may be applied to both design (e.g., lower outlet temperature, incorporation of a filtered vented confinement, etc.) and operation (e.g., initial operation at a fractional power). This is consistent with the recommended NGNP licensing strategy.[3]

The plan in each technical area has the following basic elements:

- *Safety-Significant Issues and Phenomena*. This element delineates safety-significant issues and phenomena in the respective technical disciplines that must be addressed in the course of licensing an HTGR.

- *Evaluation Models/Tools*. This element identifies evaluation models/tools that need to be developed for confirmatory safety analysis and provides a conceptual description of such models and tools. The models and tools here are defined in the broadest context and include phenomenological models and safety analysis computer codes as well as other tools to support material and structural analysis, risk assessment, etc.

- *Data Needs*. This element identifies data (experimental or analytical) that will be needed for models/tools development and assessment. The data needs will consist of those data that are essential in developing and assessing evaluation models and tools, as well as those that are useful for achieving better precision in models and tools so that unnecessary conservatism can be eliminated. Schedule for obtaining needed data is predicated upon the milestones in the licensing strategy report which stipulates the issuance of a license in 2017. As mentioned previously, if required data are not available in a timely manner, additional conservatism will need to be incorporated in models and codes used for safety analysis.

- *Data Sources*. This element addresses the sources of the research data with possible sources including DOE, vendors, international efforts, NRC-sponsored activities, or a combination of those sources.

- *Milestones*. This element provides a timeline for completion of evaluation models/tools development and assessment including verification and validation. A top-level milestone is provided in the text (by calendar year or CY quarters). More detailed milestones (Gantt charts) in each technical discipline are currently under development. The Gantt charts will be used as a project management tool during the implementation phase of the plan.

---
[2] See Energy Policy Act of 2005 (EPAct).
[3] The Next Generation Nuclear Plant Licensing Strategy – Report to Congress

Plan for Safety Analysis (Thermal-Fluids and Accident Analysis) Tools Development

Safety-Significant Issues and Phenomena

A primary infrastructure need in the safety analysis area (defined here to include thermal fluids, accident analysis, fission product release and transport, and consequence analysis) is the development of appropriate databases and validated analysis tools to help make sound regulatory decisions on key technical issues concerning HTGR safety and licensing. The safety analyses pertaining to other technical areas (notably, graphite and high-temperature material performance, process heat applications, etc.) are addressed in the respective sections of this document.

Although the full spectrum of accident scenarios for HTGR design basis has not yet been firmly established, events involving the loss of helium pressure boundary are likely to generally be the most significant for dose consequences. Safety-significant issues in the reactor plant systems analysis area include those pertaining to thermal-fluid analysis, nuclear analysis, accident source term, and fission product transport. Of these, thermal-fluid analysis is covered in this section, and the nuclear analysis area is discussed in the next section. Accident source term and fission product transport are addressed, in part, here in relation to reactor plant system analysis and, in part, in the plan for fuel performance.

The most safety-significant issues and phenomena in the thermal-fluids area include those pertaining to primary system heat transport phenomena that impact fuel and component temperatures and also the phenomena involving air/steam ingress that, however unlikely, could lead to major core and core support damage. The two most significant issues in the accident source term and fission products transport area include:

- Source term during normal operation, which provides initial and boundary conditions for accident source term calculations.

- Fission product (FP) transport phenomena during an accident, which includes release and transport from fuel into the helium pressure boundary and then into the confinement building and the environment.

Evaluation Models/Tools

The planned approach for thermal-fluid analysis is to provide data and modeling tools needed for predicting HTGR-specific heat transfer and fluid-flow phenomena, including “multi-fluid (helium with air and/or steam)” fluid flow with convection, conduction, and radiation heat transfer mechanisms in irregular and complex geometries. Figure 1 shows a schematic of the NRC accident analysis evaluation model concept for HTGRs. The evaluation models and associated analysis tools cover four areas: thermal-fluid analysis, nuclear analysis, fuel performance, and fission product release and transport. Together, these component evaluation models form the framework for an integrated accident analysis evaluation model shown in Figure 1. This section describes the thermal-fluids evaluation model and the integrated accident analysis evaluation model. Other component evaluation models are discussed later in respective sections of the research plan.

As depicted in Figure 1, the accident analysis evaluation model has four distinct parts: nuclear data preprocessing, normal operation, fission product distribution in normal operation, and transient analysis of licensing basis events. These four parts are described in turn below.

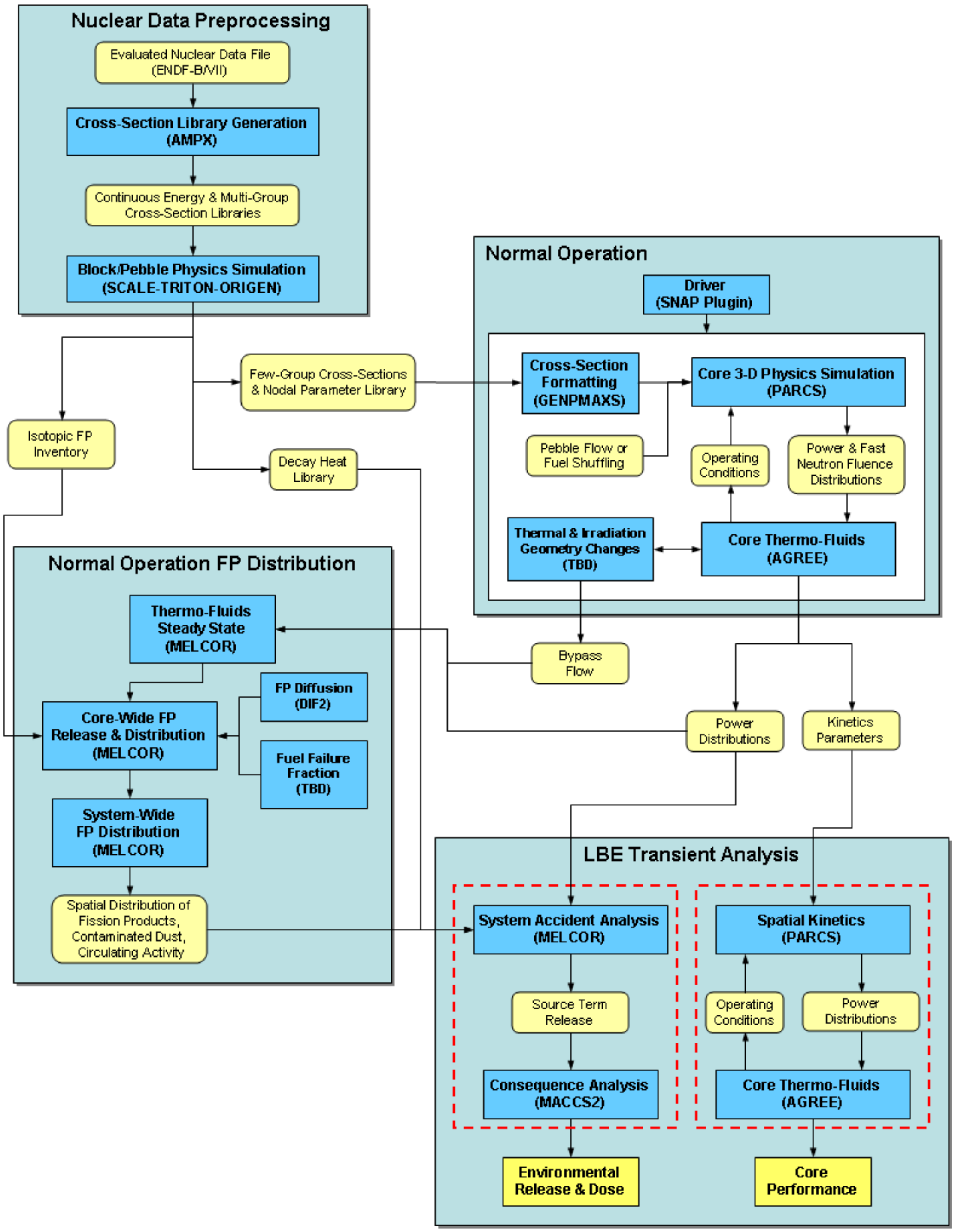

Figure 1. Schematic of the NRC Accident Analysis Evaluation Model Concept for HTGRs.

The "Nuclear Data Preprocessing" block is primarily concerned with the generation of homogenized multi-group cross-sections for usage in calculating the core power distribution during normal operation. The calculational tools used in this block are all contained in the SCALE code suite. The SCALE code suite and its modifications for HTGRs are described in more detail in the "Plan for Nuclear Analysis" section below. In addition to the macroscopic cross-sections, other outputs from this block include the isotopic fission product inventory as a function of burnup to be used in the determination of the fission product distribution within the helium pressure boundary during normal operation, and the resulting decay heat for transient heatup analyses.

The role of the "Normal Operation" block is to calculate the 3D power, temperature, and fluence distributions during normal operation. The primary analysis tools used in this block are PARCS (Program for Advanced Reactor Core Simulation) and AGREE (Advanced Gas REactor Evaluator). PARCS is a 3D core neutronics simulation code with both steady state and transient analysis capabilities. For the analysis of pebble bed reactors a cylindrical coordinate system is available, while for prismatic reactors the unit cell is a triangle that corresponds to 1/6th of a hexagonal fuel element or reflector block. To provide the fine resolution of the fuel and moderator temperature profiles necessary for an accurate estimation of the core power shape the AGREE code is coupled to PARCS. For pebble-bed reactors, AGREE uses a 3D two-temperature porous body model, while for the prismatic design the unit cell is triangle-based and uses a subchannel analysis type of flow model for flows in the coolant channels and the gaps between the graphite blocks.

The determination of the 3D power distribution in a pebble-bed reactor involves the calculation of an equilibrium core, whereas for a prismatic core reactor the power shape must be calculated over the entire fuel cycle considering both the fuel reloading pattern and the operation of the control rods. Special purpose modules ("Pebble Flow" and "Fuel Shuffling") will be developed to drive the PARCS-AGREE calculations for these reactor types.

During normal operation, the calculation of fuel "hot spots" is of primary importance due to its effect upon fission product release from both intact and damaged fuel particles. Bypass flow is one of the principal contributors to these hot spots and so must be accounted for. To do so requires an estimation of the core and reflector geometry changes that arise from both thermal (differential thermal expansion between metallic and ceramic components) and irradiation effects. A "geometry change" module will be developed to provide this input to the AGREE code and will be benchmarked against the results of detailed 3D graphite component stress analysis calculations (see discussion in the graphite section). The resulting bypass flow distribution will be output for usage by the MELCOR code.

The coupled PARCS-AGREE code does not address fission product release and transport; this is handled by the MELCOR code as detailed in the "Normal Operation FP Distribution" and "LBE Transient Analysis" blocks of Figure 1.

MELCOR is a system-level thermal-fluids and accident analysis code which, combined with a fuel performance model, provides an integrated evaluation model that calculates the steady-state as well as the transient behavior of the entire system (core, vessel, confinement) in an integral manner. In this respect, the MELCOR code will play the most significant role in the integrated accident analysis evaluation model development effort. In addition to providing a system-level analysis of the plant's thermal-fluid response to postulated transients, MELCOR will have the capability to calculate fission product transport in presence of dust including dust plateout, lift-off, and re-suspension as well as graphite oxidation resulting from air and steam ingress during a depressurized loss-of-flow accident. The thermal-fluid evaluation model provides input to the fission product release and transport model and both are integrated within MELCOR.

The approach being pursued for HTGR and VHTR licensing with regard to source terms involves the use of a mechanistic, scenario-specific accident source term rather than a conservative bounding source term. The source term is to be based on the “mechanistically” calculated release of radionuclides from all FP sources within the fuel element. Therefore, the mechanistic source term must include the transport, retention, and release of FPs within the fuel element; transport, retention, and release within structures and surfaces within the primary pressure boundary; and transport, retention, and release from the reactor confinement structure. The source term calculation will require a sound technical basis that depends on a sufficient database and modeling of fuel FP transport and release. Because of the limited operating experience and database for FP transport, testing of HTGR and VHTR production fuel and fuel materials is needed to develop and benchmark the FP release and transport models to be used in the mechanistic accident source term calculations over the range of applicable HTGR and VHTR plant operating conditions, transient conditions, and postulated accident conditions.

The “evaluation model” for fission products consists of a number of phenomenological models for fission product release and transport under normal operation as well as accident conditions. These models are described below in some detail.

- *Core-Wide Fission Product Release under Normal Operation*.[4] This element involves the development of core-wide fission product (both gaseous and metallic) release models under normal operation for incorporation into the system-level code MELCOR. The metallic fission product release model will be used to predict the time-dependent integrated release of metallic fission products from the coated fuel particles to pebbles (for the pebble-bed core) or graphite fuel compact (for the prismatic core) and to the helium coolant during normal operation over the plant lifetime. The gaseous fission product release model will be used to predict the time-dependent integrated release of noble gas and halogen fission products during normal operation. Both the metallic and the gaseous fission products would be available for immediate mobilization and release from the HTGR in the event of a break in the helium pressure boundary.

- *Core-Wide Fission Product Release under Accident Conditions.* This element involves the development of core-wide fission product (both gaseous and metallic) release models under accident conditions for incorporation into the system-level code MELCOR.

- *Fission Product Transport in Reactor Coolant System (RCS) and Containment*. This element involves modeling fission product transport phenomena in the helium pressure boundary and in confinement/containment. The fission product transport model for light-water reactors exists in MELCOR. The model will be modified, as appropriate, for HTGR fission product transport. The transport phenomena include fission product speciation in graphite and carbonaceous material, speciation during mass transfer, diffusivity and sorptivity in non-graphite surfaces, graphite dust generation, fission product transport in presence of dust, dust and aerosol plateout and liftoff, aerosol nucleation and growth, resuspension and revaporization of fission products, and potential for dust combustion.

[4] Fission product release and transport models are also referred to in the fuels performance section of the R&D plan in the appropriate context (i.e., in terms of data needs and data sources).

In addition to the role of the MELCOR code described above, consequence analysis in the "LBE Transient Analysis" block, which involves calculation of offsite impacts including dose, will use the MACCS (MELCOR Accident Consequence Code System) code with initial and boundary conditions provided by MELCOR. Also, for reactivity insertion events that are beyond the capability of MELCOR's point kinetics model, a three-dimensional spatial kinetics capability is provided by the PARCS-AGREE code. Should the calculated fuel temperatures or energy deposition in such a reactivity insertion event exceed fuel failure limits so that FP release is enhanced, the transient analysis would be repeated using MELCOR (for the FP release and transport) with the power excursion provided from PARCS-AGREE.

HTGR design and development organizations have used FLUENT and other computational fluid dynamics (CFD) codes to examine the details of flows in HTGRs. The ability of these codes to simulate turbulent mixing in complex geometries makes them well suited for analysis of flows in the upper and lower plena of HTGRs where buoyant plumes and hot jets may exist. Natural convection flow and heat transfer dominate the reactor cavity cooling system (RCCS) heat removal performance, and CFD may be an appropriate tool to effectively examine the details involved in operation of the RCCS. The staff may also explore the CFD capability to calculate the steady-state distribution of radionuclides (e.g., fission products absorbed in dust) within the primary system to provide the initial source term distribution for the calculation of the accident source term.

To summarize, the integrated accident analysis evaluation models/tools consist of an adequate suite of reactor systems analysis tools (i.e., computer codes and methods) that provide the staff with an independent capability to reliably predict HTGR reactor plant system behavior accident source term and fission product transport and release in response to licensing basis events. The staff will use the suite of analysis tools to (1) conduct confirmatory analyses of licensing-basis events, (3) conduct sensitivity studies to better understand uncertainties and safety margins, and (4) support development of the HTGR regulatory requirements and to inform policy decisions. Figure 1 provides a schematic of interrelationship between various codes in the suite of reactor system analysis tools.

Data Needs

The data needs[5] are discussed here in accordance with the reactor operation mode (i.e., normal operation) and accident categories. Both model development and code validation give rise to data needs. Examples of data needs for model development would include separate effects tests for the effective thermal conductivity of a pebble-bed core and the estimation of pressure loss coefficients for the bypass flows in a prismatic core. In many cases, recently developed and/or legacy data may be adequate to meet these data needs. However, in other cases, new data will have to be generated through existing or planned experimental programs—both domestic and international. Consistent with the target completion schedule of 2013 for tools development in the safety analysis area, new data in most cases will need to be generated by 2013.

- *Normal Operation (NO).* Flow distribution in core, plena, and cavity; heat-transfer data; coolant properties; moderator properties (irradiation effect); other physical properties of the

[5] Data needs described in this section are for thermal-fluids only. Needs for fission product release and transport data are described in the fuel performance and fission product section as they relate to DOE's fuel testing programs.

structures (e.g., emissivity); fuel performance (e.g., failure fraction); and power and flux profiles (from reactor physics calculations) are needed.

- *Anticipated Transients Without Scram (ATWS).* Scoping calculations can be performed for reactivity insertion due to steam-water ingress accidents. Control rod worth and reserve shutdown worth calculations are normally performed as part of the reactor startup, and are discussed further in the nuclear analysis section. Existing data from HTTR and HTR-10 (and legacy data from other test and production reactors) could be used for the purpose. Additional data from HTTR and HTR-10, if available, can be used for further refinement of calculations.

- *Pressurized Loss of Forced Circulation (P-LOFC).* Same or similar data as in normal operation are needed. In addition, data on RCCS spatial heat loading, radiation heat transfer, coolant bypass, core effective thermal conductivity, decay heat, etc. are needed.

- *Depressurized Loss of Forced Circulation (D-LOFC).* Additional to P-LOFC requirements, data on fuel performance during heatup, dust mobilization, pressure pulse in confinement, and cavity filtering performance will be needed. Moreover, in a D-LOFC accident with air ingress (AI) or steam ingress, data will be needed on fuel and graphite oxidation, heat transfer correlation, and flow characterization for multiphase system.

Data Sources

Data for the safety analysis of HTGR will be obtained from a variety of sources including legacy data in the open literature, DOE-funded NGNP research, international cooperative agreements, vendor data, and NRC-funded programs. Table 1 presents a preliminary list of these data sources organized by the relevant phenomena identified during the NGNP Phenomena Identification and Ranking Tables (PIRT)[6] exercise. The Table also identifies the scenario or mode to which the phenomena apply as discussed above under "Data Needs." The phenomena list covers a wide range of scenarios including those listed above[7]. The set of phenomena listed in the table is a subset of phenomena identified by the PIRT panels, the subset being one that includes mostly those phenomena ranked high in importance but low in knowledge level. It is assumed that the majority of the data will be available from DOE or an applicant in the timeframe for licensing a prototype NGNP as mandated by the EPAct. Some data exist in the open literature and will be used as appropriate. In a few cases, the schedule for data availability is currently unknown (to be determined or TBD), but it is expected that the data will be available by 2013 or shortly afterwards. Table 1 identifies the organization(s) expected to provide the data. Table 2 identifies the major milestones for safety analysis tools development. In both these tables, and those that follow, the notation (c) refers to an activity already complete.

[6] See NUREG/CR-6944 (Vols. 1 through 6) for details.
[7] The steam or water ingress scenario, although summarily considered during the PIRT deliberation, was not ranked high in importance because of the then prevailing HTGR designs. Since then, the designs have evolved to a state where further consideration of steam/water ingress may be necessary. As such, the data needs and sources require further evaluation. The evaluation will be performed using a PIRT-like process as soon as sufficient information on the concept design becomes available.

Table 1. Data Needs and Sources for Thermal Fluids and Accident Analysis

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider[8]	Schedule
Coolant properties	All	NIST data for He and air constituents	NRC	--[9]
Gas mixture rule	All	NIST data for air	NRC	--
Core bypass flow	NO	NGNP INERI experiments (KAERI/SNU) INL MIR experiments (isothermal) NGNP heated bypass experiments HENDL data (JAERI-1333) GA legacy test data (FSV support) NGNP design specific (keys & seals) data NGNP graphite block geometry changes TAMU co-op R&D on bypass flow	DOE DOE DOE NRC Vendor DOE DOE NRC	2011 2011 2012 2010 (c)[10] 2011[11] TBD[12] TBD 2013
Core Flow Distribution	NO	HTTU flow tests (PBMR Ltd) OSU IET (HTTF-PMR)	DOE DOE/NRC	2012 2012
PBR: core-wall interface effects on bypass flow.	NO	OSU IET (HTTF-PBR) HPTU/HTTU data (PBMR Ltd.) TAMU PBR SET	DOE DOE NRC	TBD 2012 2010 (c)
Coolant Heat Transfer Correlations: • Prismatic Core • Pebble Bed	NO	OSU IET (HTTF- PMR) TAMU PMR SET (var. properties effect) HTTU data with flow (PBMR Ltd.) TAMU PBR SET (one sphere in bed)	NRC NRC DOE NRC	2013 2012 2012 2012
PBR: dispersion (“braiding”) effect upon core temp. distribution.	NO	HTTU data with flow (PBMR Ltd.) Coop: TAMU PBR SET	DOE NRC	2012 2012
Outlet Plenum Flow Distribution[13]	NO	OSU IET (simulating high pressure He with N_2)	NRC	2013
Pebble Flow[14]	NO	Specific experiment(s) and/or data source(s) will be identified in the future.	DOE	2013
PMR: effective fuel element block thermal conductivity.	NO	INL Advanced Graphite Creep (AGC) experiments[15] OSU IET (HTTF-PMR fuel design)	DOE NRC	2013 2012
RCCS heat removal for normal operating conditions.	NO	IAEA HTTR benchmark ANL RCCS experiments (1/2 height, air and water cooled) UW RCCS experiments under NEUP (1/4	NRC DOE	2010 (c) TBD

[8] Organization expected to provide the data. Determined through interactions with DOE. In some cases, “DOE” might be replaced by the applicant.
[9] Data currently exists.
[10] Schedule with notation (c) indicates the activity is complete.
[11] Schedule contingent on vendor making the data available to NRC.
[12] Schedule contingent upon selection of specific design(s) and development of detail design data.
[13] A design/operation issue for which data is expected to be generated by the designer/applicant. Additional data at low flow identified here for validation purpose.
[14] Data will be needed for conditions that scale to graphite in high-temperature helium. Both radial and azimuthal (for annular core design) data will be needed.
[15] Ongoing and planned experiments under the INL/AGC program. Initial irradiation data @ 600°C will be available in 2013. See Graphite section in this Plan for details.

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider[8]	Schedule
		height, water cooled) TAMU RCCS experiments under NEUP (small-scale, water cooled) OSU IET (HTTF-PMR)	DOE DOE NRC	2012 2012 2013
PBR: effective core thermal conductivity	G-LOFC[16] D-LOFC	SANA (Note: Tmax < 1200 °C) HTTU data (PBMR Ltd.) OSU IET (HTTF-PBR)	NRC DOE DOE	2010 (c) 2012 TBD
Vessel emissivity: includes includes aging and dust effects	G-LOFC	UW (DOE NERI/NEUP) UW (NRC Coop Program)	DOE NRC	2011 2010 (c)
RCCS Panel Emissivity	G-LOFC	UW (DOE NERI/NEUP) UW (NRC Coop Program)	DOE NRC	2011 2010 (c)
Core barrel emissivity: dust issues	G-LOFC	UW (DOE NERI/NEUP)	DOE	2012
Reactor vessel cavity air circulation and heat transfer[17]	G-LOFC	ANL RCCS experiments UW RCCS experiments (NEUP) TAMU RCCS experiments (NEUP) OSU IET (HTTF-PMR)	DOE DOE DOE NRC	TBD 2012 2012 2013
Reflectors: thermal conductivity effects of radiation damage and annealing[18]	G-LOFC	Specific experiment(s) and/or data source(s) will be identified in the future	DOE	2013
RCCS fouling on coolant side[19]	G-LOFC	Specific experiment(s) and/or data source(s) will be identified in the future	--	--
RCCS spatial heat loadings: includes parallel channel and stratification effects	G-LOFC P-LOFC	ANL RCCS experiments OSU IET (HTTF-PMR)	DOE NRC	TBD 2013
RCCS performance including failure of 1 of 2 channels: • Forced to natural circulation • Single-phase to boiling transition • Parallel channel interactions • NC in horizontal	G-LOFC	ANL RCCS experiments OSU IET (HTTF-PMR) Purdue THI boling/condensing loop for flashing induced oscillations	DOE NRC NRC	TBD 2013 2011

[16] G-LOFC denotes generic LOFC phenomena, that is, those that occur in both pressurized (P-LOFC) and depressurized (D-LOFC) events.
[17] Includes consideration of vessel to RCCS effective view factors. No specific assessment of the latter is planned since that is subsumed in the ANL RCCS experiments.
[18] Needed for auditing safety calculations; in lieu of data, conservatisms will be introduced.
[19] Use tech specs to avoid condition, otherwise a conservative bounding approach.

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider[8]	Schedule
panels				
RCCS failure of both channels; heat transfer from RCCS to cavity wall	G-LOFC	Need for specific experiment(s) and/or additional data source(s) not identified; bounding calculations may be adequate	--	--
Inlet plenum stratification & plumes	P-LOFC D-LOFC	NGNP plenum-to-plenum experiment OSU IET (HTTF-PMR)	DOE NRC	2013 2013
Radiant heat transfer from top of core to upper vessel head	P-LOFC	OSU IET (HTTF-PMR)[20]	NRC	2013
Core coolant flow distribution: low-flow correlations and natural circulation.	P-LOFC	SANA (He vs. N2 data) HTTU NC and low-flow data (PBMR Ltd.) OSU IET (HTTF-PMR)	NRC DOE NRC	2010 (c) 2012 2013
Core coolant (channel) bypass flow, involves low-flow correlations and flow reversal	P-LOFC	NGNP INERI experiments (KAERI/SNU) INL MIR experiments (isothermal) NGNP plenum-to-plenum experiment TAMU co-op R&D on bypass flow	DOE DOE DOE NRC	2011 2011 2013 2013
Coolant flow friction/viscosity effects	P-LOFC	TAMU PMR SET (var. properties effect)	NRC	2012
Hydrodynamic conditions for dust suspension	D-LOFC	Data source(s) not identified.[21]	DOE	2013
Pressure pulse in confinement	D-LOFC	NGNP cavity blow down experiments[22]	DOE	TBD
Heat transfer correlations for mixed gases in core	AI[23]	NGNP core heat transfer experiments Extension of TAMU PMR SET	DOE NRC	TBD 2013
Core support structures oxidation	AI	INL/KAIST INERI experiments NACOK-PBR experiments	DOE DOE	2010 (c) 2010 (c)
Core oxidation	AI	NACOK-PBR experiments NACOK-PMR experiments	DOE DOE	2010 (c) TBD
Reactor cavity-to-vessel air ingress • duct exchange flow • molecular diffusion	AI	THI stratified flow tests (PSU) Diffusion data (Duncan & Toor tests) JAERI inverted U-tube tests OSU IET (HTTF-PMR) CFD benchmark analyses INL – Test 1 (isothermal CC flow)	NRC NRC NRC NRC NRC/DOE DOE DOE	2011 2011 2011 2013 2012 2010 (c) TBD

20 Apply uncertainties to model inputs.
21 A future PIRT-type workshop will address this issue in greater details.
22 The nature and the scope of these experiments are not defined yet. The phenomenon is relevant to structural load calculation.
23 Specific subset of D-LOFC involving air ingress. Another subset of D-LOFC involving steam/water ingress need be considered for HTGR designs that employ steam cycle for heat utilization. A future PIRT-type workshop will address the water ingress issue in greater details.

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider[8]	Schedule
		INL – Test 2 (non-isothermal) INL – Test 3 (#2 with core flow resistance)	DOE	TBD
Phenomena that affect cavity gas composition	AI	NGNP cavity blowdown experiments Bounding calculations in lieu of data	DOE	TBD
Confinement-to-reactor cavity air ingress	AI	Performance criteria to be supplied by vendor	Vendor	TBD
Cavity filtering performance (affects dust-born releases)	AI	Dust filtering options should be investigated and tested.	DOE	2013

Table 2. Major Milestones for Safety Analysis Tools Development

Activity Title	Activity Begins	Activity Completed
MELCOR thermal-fluid models development and improvement	4Q08	1Q12
MELCOR fission products models develop and improvement	2Q09	3Q11
MELCOR plant input model development (conceptual and NGNP)	4Q08	1Q12
MELCOR code assessment using legacy data	1Q11	1Q12
Development of separate effect (SE) and integral effect (IE) data	4Q08	2Q13
MELCOR code assessment using new data[24]	1Q12	3Q13
Extended MELCOR assessment using AGR data[25]	1Q13	4Q18
PARCS and AGREE improvements	3Q09	2Q12
SNAP driver for core analysis	1Q12	4Q12
Initial PARCS-AGREE benchmarking and assessment	2Q12	3Q13
Final PARCS-AGREE benchmarking and assessment	3Q13	3Q14
Code integration	3Q11	2Q13
Uncertainty analysis methodology development	2Q11	4Q12
Code applicability report	3Q13	4Q13
Computational fluid dynamic (CFD) analysis – complementary	TBD	TBD

[24] Contingent on availability of the AGR data in a timely manner.
[25] MELCOR assessment activity will be conducted in parallel with the AGR fuel irradiation and heatup experimental program; AGR data will be used as it becomes available.

Plan for Nuclear Analysis

Safety-Significant Issues and Phenomena

Nuclear analysis of a reactor system is performed to predict criticality; heat generation during operation, shutdown, and transient conditions including direct fission and decay heat; radionuclide inventories and source terms; material irradiation fluence and activation; direct radiation dose to workers or public; and out-of-reactor criticality safety and dose assessments during all phases of the front and back ends of the fuel cycle. Generally, the scope of nuclear analysis described here is consistent with that for light water reactors with few exceptions which refers to unique features of NGNP. The nuclear analysis plan addresses the following safety-significant issues and phenomena:

- Reactor steady-state operation to confirm that the fuel and other materials do not exceed acceptable operating limits such as fuel and structural temperatures, and mechanical properties that are influenced by neutron irradiation.

- Adequacy of reactor shutdown and reactivity control systems to ensure that the system has suitable shutdown margin and that it can safely respond to reactivity transients.

- Reactor and fuel isotopic inventories that are used to generate radionuclide source terms for severe accidents, spent fuel storage and handling, and spent fuel transportation.

- Behavior under accident conditions including decay heat sources and inherent reactivity feedback effects (e.g. Doppler fuel temperature feedback).

- Dose to personnel during at-power (e.g., radiation streaming paths are more prevalent in gas-cooled systems) and shutdown (e.g., fuel movement) operations.

- Criticality safety issues in the fabrication, transportation, handling, and storage of fresh fuel caused by lack of experimental data for fuel with enrichment > 5 wt% U-235.

- Criticality safety (e.g., burnup credit), decay heat, and radiation shielding issues related to handling, storage, and transport of the very high burnup spent fuel expected from NGNP.

Evaluation Models/Tools

The nuclear analysis evaluation model relies primarily on the adaptation of existing NRC-developed tools and data libraries and fall into the following categories:

- *Nuclear Data.* Development activities require producing updated ENDF/B-VII nuclear data libraries and associated AMPX nuclear data processing system to address NGNP-specific needs for graphite-moderated systems.

- *HTGR Lattice Physics/Cross Section Processing.* The activities include modification of the SCALE/TRITON system with updated physics methods to analyze the TRISO (TRI-structural iSOtropic) coated particle fuels, graphite moderators, and significant changes in neutron spectrum in the reactor for both pebble bed and prismatic designs and, finally, to analyze integration with the core physics code.

- *Radionuclide Source Terms/Decay Hea*t. The activities include development of SCALE/ORIGEN system with appropriate cross section and decay data, and verification and validation (V&V) to support very high-burnup NGNP applications that must consider spatial and burnup dependence of decay heat sources.

- *HTGR Core Physics*. The activities include development of the PARCS code for steady-state and transient conditions coupled with a suitable thermal-fluids code system to provide thermal feedback, integration with SCALE/TRITON, and improvements in the ability to model reactor spectral variations, such as microscopic depletion. SCALE/TRITON three-dimensional Monte Carlo depletion capability for high-fidelity reference calculations is required for verification purposes.

- *Criticality Safety*. The activities include assessing the suitability and sufficiency of the available experiment database for determining safety margins with > 5 wt% U-235 graphite-moderated systems. Sensitivity/uncertainty methodology (i.e., SCALE/TSUNAMI), developed within the SCALE system and modified further for NGNP, will be used for this purpose.

- *Shielding/Radiation Transport.* The activities include SCALE system MAVRIC hybrid Monte Carlo/deterministic radiation transport methods with enhancements for NGNP fuels, storage of core materials and graphite, and shipping containers.

- *Bias and Uncertainty Quantification.* The activities include SCALE/TSUNAMI capability to provide bias and uncertainty quantification for both criticality safety and reactor safety parameters (e.g. temperature coefficients) updated to include physics methods improvements discussed under lattice physics area and enhancements in nuclear data.

Many of the methods and tools developed in the HTGR reactor analysis area will be applicable to the spent fuel issues. The main computer code system used in the analysis of spent fuel issues will be the SCALE code, which NRC currently uses extensively for LWR-related evaluations. The SCALE upgrades common to the conditions inside and outside the core will include nuclear data processing improvements, cross section evaluations for double heterogeneity (i.e., fuel, structure and coolant heterogeneity in reactor core plus heterogeneity within the fuel form itself), and improvements in transport methods. SCALE activities specific to HTGR spent fuel will include nuclear data libraries for criticality and shielding and radiation shielding methods.

Data Needs

The quantification of the uncertainty and biases (using SCALE/TSUNAMI) in the evaluation model requires experimental data. Measured data from criticality experiments and isotopic assays would provide valuable input for the uncertainty evaluations of the proposed analytical and computational methods, and would allow the determination of adequate acceptance criteria. These data can be classified as follows:

- *Nuclear Data*. Scattering data under irradiated and unirradiated conditions may be needed for graphite types proposed for NGNP to provide updated neutron-scattering kernels in order to investigate changes in the reactor safety parameters over time. This need will be established only after sensitivity/uncertainty analysis is performed using available graphite scattering data.

- *Critical Experiments.* Critical experiment data are used to assess codes and data for the reactor system and away-from-reactor applications. It is expected that criticality validation needs will arise for HTGR safety because of the shortage of evaluated critical benchmark experiments involving neutron moderation by graphite and graphite with water, fuel material with 5 to 19.9 wt% ^{235}U enrichment, and particle fuel geometries. Data will be needed for the validation of criticality safety of fuel with enrichment greater than 5 wt%, radiation shielding, decay heat, and source term. Measured parameters (including uncertainties) are needed for critical configuration, power and flux distribution, reactivity impacts of control materials, reactivity impacts of water ingress, and temperature effects (fuel, moderator, coolant). Data are available from previous and current U.S. and international programs. The most relevant are from data associated with the startup of the HTR-10 and HTTR reactors in China and Japan, respectively. Additional data are available from relatively recent experiments at the Paul Scherrer Institute (PSI) PROTEUS facility and the Russian Research Center – Kurchatov Institute ASTRA facility. Not all of the available data from these experiments have been publically released, and limited data are available at high temperatures. Critical experiments to support the fuel fabrication, storage, and transportation are also required for fuels with > 5 wt% U-235 enrichment.

- *Core Operational Data*. Operational data are required for full reactor operation to provide the ability to validate predictions of power distribution, criticality, control system reactivity worth, thermal feedback, and recriticality during accident conditions. While new data from HTR-10 and HTTR reactors will be useful in this respect, lacking such data it will still be possible to validate the neutronic calculations, albeit with less accuracy. Sensitivity analysis may be needed to account for any uncertainties in data and the effect of such uncertainties on predicted results.

- *Fuel Isotopic and Decay Heat Data.* Assessment of the source terms and decay heat requires measurements of spent fuel isotopic compositions for a range of fuel burnup up to the reactor discharge burnup and with well-characterized irradiation histories. In addition, direct measurements of decay heat as a function of burnup are necessary to quantify uncertainties and biases in decay heat predictions. Also, fuel isotopic data are needed for the verification of the lattice and core physics prediction of fuel depletion.

Data Sources

As in the case of safety analysis tools development, it is assumed that the majority of data for nuclear analysis will be available in the timeframe for licensing a prototype NGNP as mandated by the EPAct. Table 3 identifies the organization(s) expected to provide the data. Table 4 identifies the major milestones for nuclear analysis R&D.

Table 3. Data Needs and Sources for Nuclear Analysis

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider	Schedule
Criticality safety evaluation	Out-of-reactor	Experimental data (OECD ICSBEP). Fundamental experimental data for uranium/graphite systems for	NRC	2010[26] (c)

[26] Existing experimental data including those from nuclear data libraries and handbook will be used.

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider	Schedule
		nuclear data validation.		
HTR-10 and HTTR startup	Startup (S/U)	Startup measured data: - Critical configurations - Control rod worth - Temperature and power coefficients - Power distributions Existing experimental data through OECD, IRPhEP, and IAEA/CRP. Startup configurations and measurements documented in international handbook and IAEA coordinated research program	NRC	2010[27] (c)
	S/U	HTTR experimental Data (DOE-JAEA Agreement)[28] - Reactor data (materials, impurities, geometries). - Additional measurements as can be made available for HTTR	DOE	2013
	S/U	Additional HTR-10 experimental Data - Reactor data (materials, impurities, geometries). - Additional measurements as can be made available for HTR-10	DOE[29]	2013
PROTEUS-HTR, ASTRA plus additional experiments	S/U, NO	Existing experimental data through IAEA/OECD. Existing data is available from previous IAEA CRP and OECD/IRPhEP reactor physics handbook	NRC	2011
	S/U, NO	Experimental data (new). Additional measurements are being performed in Astra to support PBMR and would be useful for validation	DOE	2013

[27] Existing HTTR and HTR-10 startup data will be used. Limited data available, in particular, the availability of HTR-10 data is not assured at this time.
[28] Agreement currently under discussion. Data availability contingent on timely implementation of the agreement.
[29] Subject to availability of the data under the Generation IV International Forum (GIF).

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider	Schedule
Code-to-code benchmarking/international exercises on HTGR	S/U, NO	International activities have been performed by - Computational benchmark problems on prismatic and pebble-bed reactors - Reactor unit cells, fuel blocks, cores - Steady-state neutronics configurations – multiplication factors, temperature and power coefficients, power and flux distributions - Depletion/burnup – isotopic compositions, decay heat - Steady-state coupled neutronics and thermal hydraulics Transients IAEA Coordinated Research Programs and OECD/NEA Working Party of Reactor Systems. Additional joint benchmarks of NRC and DOE/Vendor codes and data	NRC NRC/DOE	2010 (c) 2013
HTTR normal operation	NO	- Data from operation of HTTR at power - Multiplication factors, control rod positions, etc. as a function of measured conditions (DOE-JAEA Agreement)	DOE	2011
HTR-10 normal operation	NO	- Data from operation of HTTR at power - Multiplication factors, control rod positions, etc. as a function of measured conditions (DOE- INET Agreement needed)	NRC	2011[30]
Radiation shielding validation and verification	In-and-out of reactor	Measured data from past and present facilities to test enhanced radiation transport codes and data.	NRC DOE[31]	2013 TBD[32]
Criticality safety, >5 wt% ^{235}U enrichment, graphite systems	Out-of-Reactor	Criticality experimental data for >5 wt% ^{235}U enrichment	NRC DOE	2013 TBD

[30] Subject to the availability of data.
[31] Data from new experiments to be procured by DOE at a later time frame.
[32] TBD in the context of nuclear data generation is currently estimated to be prior to 2016.

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider	Schedule
Decay heat, source terms, isotopics V&V	In-and-out of reactor	- Measured isotopic composition of HTR fuels at typical burnups - Measured decay heat of HTR fuels at a range of burnups. (Little existing relevant data have been located to date. Collaborative efforts with China and Japan may yield useful data) Fort St. Vrain operational data for decay heat, isotopics, source term, etc.	NRC DOE DOE/NRC	2010[33] (c) TBD 2010 (c)

Table 4. Major Milestones for Nuclear Analysis R&D

Activity Title	Activity Begins	Activity Completed
Nuclear data processing methods and data libraries	3Q09	3Q12
Sensitivity and uncertainty analysis	1Q10	1Q13
Radiation shielding methods and data	3Q09	3Q12
Decay heat and source term validation	1Q11	1Q13
Criticality safety assessment and validation	3Q09	3Q12
SCALE-PARCS integration	3Q09	4Q12
Code-to-code benchmarking/international exercises	3Q08	4Q12
HTTR/HTR-10 existing operational data evaluations	2Q11	2Q12
Evaluation of code biases and uncertainties	4Q11	4Q12

Plan for Fuel Performance and Fission Product Release and Transport

Safety-Significant Issues and Phenomena

HTGR and VHTR cores will contain several billion coated fuel particles (CFPs) dispersed within several hundred thousand graphite fuel elements (i.e., fuel pebbles, fuel compacts). The CFPs are the primary safety barrier to the release of radionuclides to the environment during normal operation and under accident conditions. Thus, the basis of the safety case and safety analysis of HTGR and VHTR designs is the expectation and the requirement that CFPs will have a very low failure rate within the licensing-basis envelope. Accordingly, the HTGR and VHTR designs must ensure with high confidence that only a very small fraction (i.e., 10^{-5} to 10^{-4}) of the CFPs within the core will fail as a result of the combined effects of manufacturing defects, operational service conditions, and any accident conditions. HTGR and VHTR fuel qualification test programs are expected to demonstrate that fuel performance requirements are met.

[33] Additional data may be available by 2013.

The CFP performance analysis models and methods are not yet fully developed, benchmarked, and validated for the specific CFPs that must be qualified for loading into the future HTGR cores. HTGR fuel qualification is primarily based on irradiation testing and accident condition testing of prototypical production fuel to demonstrate the in-reactor performance (i.e., CFP failure fraction) and fission product (FP) release behavior of the production fuel. Test conditions are intended to conservatively envelope the core conditions, especially with respect to those parameters that have a strong effect on the degradation and failure of CFPs.

Evaluation Models/Tools

The evaluation models for fuel performance pertain to the development of a stand-alone mechanistic fuel performance analysis code which is described below.

Fuel Particle Failure Rate Response Surface Model. This element involves the development of a stand-alone mechanistic fuel performance analysis model and the associated code. The activity will support the development of the NGNP fuel particle failure rate response surface that is needed by the NGNP accident analysis evaluation model. NRC has recently received the stand-alone PARFUME (PARticle FUel ModEl) code from DOE/INL. The code and associated models will be used for predicting the behavior of TRISO particle fuel during reactor normal operation and heatup accidents, including the fuel particle failure probability. However, the PARFUME code will not be incorporated directly in NRC's system-level accident analysis code, MELCOR. Instead, it will be used to benchmark a simplified version of the fuel performance analysis module that is appropriate for incorporation into MELCOR.

In addition to the above, the R&D activities in the fuel performance area include development of fuel fabrication and quality control inspection guidance. This element involves the development of guidance for important fuel manufacturing process parameters and fuel product parameters and their associated specifications and the relationship of these parameters to fuel performance during operational fuel irradiation and accidents. The element also involves manufacturing process controls and product controls that keep variation of the fuel characteristics within allowable tolerances. Finally, the element involves statistical methods, product sampling analysis methods, and statistical acceptance calculations.

Data Needs

Fuel Performance Data. Material properties data (e.g., elastic modulus of different coating and buffer layers, SiC strength, PyC anisotropy, etc.) and physical-chemical properties data for unirradiated fuel are needed to support development of the fuel performance model. Similar data for irradiated fuel are also needed to develop a fuel failure rate response surface model. These data are expected to come primarily from the Advanced Gas Reactor (AGR) fuel campaign in the advanced test reactor at INL as part of the DOE-sponsored VHTR R&D. The AGR-1 test series with lab-developed fuel has been completed. The post-irradiation examination of the AGR-1 fuel has begun and post-irradiation heat-up testing (simulating accident conditions) is planned. The AGR-2 and the follow-on series will examine production-quality fuel with both UO_2 and UCO kernels. Data on fuel particle defect rate during normal operations and accident conditions will be generated in AGR-5 and AGR-6. This will support development of the statistical failure rate model. The post-irradiation examination (PIE) of the fuel from the AGR campaign is expected to produce needed properties and performance data. Additional data may be obtained from other sources including historic and current test reactor data.

Fission Product Release Data. The AGR campaign also will produce fission product release data including metallic and gaseous fission products. In AGR-3, metallic and gaseous fission product concentrations will be measured to provide data on fission product diffusivities in kernel. In AGR-4, metallic and gaseous fission product concentrations will be measured to provide data on fission product diffusivities through matrix materials. The concentration data will be measured under conditions that will simulate normal operations and core heatup. Also, as in AGR-1 and AGR-2, the PIE of AGR-3 and AGR-4, including scanning electron microscopy (SEM) to determine failure mechanisms, will produce material properties and performance data for irradiated fuel.

Fission Product Transport Data. The AGR campaign will produce fission product transport data for the fuel kernel, particle coatings and fuel matrix material. However, data will also be needed for modeling fission product transport within in the helium pressure boundary and the confinement structure. Data needs exist in several areas, including dust generation and transport, fission product speciation, plate-out, lift-off, resuspension, and sorptivity in graphite and nongraphite surfaces. The AGR fuel development and qualification program plan document, which was updated and issued by INL in September 2010, describes the plan for developing the data needed to model fission product transport within the NGNP helium pressure boundary and confinement structure. Although test plans have been developed for obtaining the needed data, the schedules and facilities involved have not yet been established.[34]

Fuel Fabrication and Quality Control Data. Data on important fuel manufacturing process parameters and fuel product parameters and their associated specifications are needed to develop a guidance document for NRC inspectors. Such data will include fabrication parameters for kernels and coatings as well as matrix and fuel elements, manufacturing process controls and product controls that keep variation of the fuel characteristics within allowable tolerances, and product-sampling analysis methods and data for acceptance verification.

Data Sources

As in thermal-fluid and accident analysis, data for fuel performance and fission product behavior will be obtained from a variety of sources, including legacy data in the open literature, DOE-funded NGNP research, international cooperative agreements, vendor data, and NRC-funded programs. A preliminary list of these data sources is shown below organized by the relevant phenomena identified during the Phenomena Identification and Ranking Tables (PIRT)[35] exercise. The phenomena list covers the scenarios listed previously under safety analysis. Much of the fuel fission product data from the DOE-funded NGNP research will be generated in a timeframe beyond current expectations for licensing a prototype NGNP. Thus, some legacy data, combined with initial set of data obtained from the DOE-funded research, will be used to develop fuel performance and fission product models and also to perform preliminary assessment of these models.[36] As more data become available from the ongoing research, and models and codes are assessed more thoroughly, conservatism in the analysis tools could be reduced and correspondingly, compensatory measures could be gradually eliminated. Table 5 identifies the organization(s) expected to provide the data. Table 6 identifies the major milestones.

---

[34] Until data becomes available, selected conservative measures will be incorporated into the evaluation model to account for uncertainties which may result in compensatory measures in the design and operation of the NGNP.

[35] See NUREG/CR-6944 (Vols. 1 and 3) and NUREG/CR-6844 for details.

[36] See footnote 34.

Table 5. Data Needs and Sources for Fuel Performance and Fission Product Behavior

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider	Schedule
Fuel particle failure fraction (failure) rate response surface for PARFUME development and validation	All	Unirradiated material properties (thermo-mechanical, thermo-physical, and chemical properties) of coated particles (e.g., SiC strength, elastic modulus)	DOE/INL	2011[37]
		Content of gaseous fission products (FP), CO and CO_2 in intact particles from AGR-2 experiment and PIE	DOE/INL	2013
		Irradiated material properties (e.g., SiC strength, PyC anisotropy, elastic modulus) from AGR-3, PIE and post-irradiation testing	DOE/INL	2015
		SiC palladium attack during accident conditions	DOE/INERI	2012
		Particle failure prediction model development (from some measurable level of coated fuel particle [CFP] failures during the irradiation) using AGR-1 through AGR-6 experiments and PARFUME code validation using AGR-7 experiment	DOE/INL	2017
		Fuel particle defect rate from manufacture and AGR-5 and AGR-6	Vendor DOE/INL	2012 - 2013
		Fuel particle failure fraction vs burnup and irradiation temperature from AGR-5, AGR-6 experiments and PIE; also particle failure rate statistics under normal operation and accident conditions	DOE/INL	2012 - 2013
Fuel particle failure fraction (failure rate) response surface for MELCOR development		Fuel particle failure fraction vs. core heatup temperature from AGR-5 and AGR-6 experiments; also particle failure rate statistics under normal operation and accident conditions	DOE/INL	2015 - 2016
Core-wide gaseous fission product release during normal operation (NO)	NO	Gaseous FP transport from failed particles (e.g., diffusion in kernel) from AGR-3 experiment to measure gas concentrations and provide data on kernel diffusivities	DOE/INL	2011 - 2012

[37] New data will augment existing database.

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider	Schedule
		Gaseous fission product transport from intact particles (e.g. diffusion through coatings) from AGR-3 experiment to provide data on coating diffusivities	DOE/INL	2011 - 2012
		Gaseous FP diffusivities and sorptivities in compact matrix and graphite using design-to-fail (DTF) particles from AGR-4 irradiation experiment and PIE	DOE/INL	2011 - 2012
Core-wide metallic fission product release during normal operation (NO)	NO	Metallic FP transport from failed particles(e.g., diffusion in kernel) from AGR-3 and PIE	DOE/INL	2013 - 2014
		Metallic FP diffusivities and sorptivities (through measurements of metallic FP concentrations across compact matrix and graphite specimens) from AGR-4 and PIE	DOE/INL	2013 - 2014
Core-Wide FP release for core heat-up accidents	P-LOFC D-LOFC	Gaseous FP transport (e.g., diffusion in kernel) and release from irradiated failed particles during AGR-3 test and PIE and also core conduction cooldown (CCCD) test	DOE/INL	2013 - 2014
		Metallic FP transport (e.g., diffusion in kernel) and release from irradiated failed particles during AGR-3 test and PIE and also CCCD test	DOE/INL	2013 - 2014
		Gaseous FP diffusivities and sorptivities in compact matrix and graphite using DTF particle during AGR-4 test and PIE and also CCCD test	DOE/INL	2013 - 2014
		Metallic FP diffusivities and sorptivities in compact matrix and graphite using DTF particles during AGR-4 test and PIE and also CCCD test	DOE/INL	2013 - 2014
		Gaseous FP releases from intact particles and compact with irradiated intact particles during AGR-5, AGR-6 and CCCD tests	DOE/INL	2015 – 2016
		Metallic FP releases from intact particles and compact with irradiated intact particles during AGR-5, AGR-6 and CCCD tests	DOE/INL	2015 - 2016

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider	Schedule
Core-Wide FP release for core heat-up accidents with air ingress (AI)[38]	AI	Particle failures due to air ingress[39] by measuring enhanced FP releases from compact with irradiated intact particles during AGR-5, AGR-6, and CCCD with oxygen partial pressure	DOE/INL	2015 - 2016
Core-Wide FP release for core heat-up accidents with moisture ingress[40]	SI[41]	Enhanced fission product release from exposed kernels failure due to steam/water ingress[42]	DOE/INL	2015 - 2016
Core-Wide FP release during reactivity insertion accidents[43]	Reactivity Initiated Accidents (RIA)	Particle failures due to energy deposition in kernel with temperature rise from AGR-5 and AGR-6 tests with reactivity Experiments in Nuclear Safety Research Reactor (NSRR) Japan	DOE/INL NSRR, Japan	TBD[44]
FP speciation in carbonaceous material	All	Chemical form in graphite affects FP transport in RCS and in containment. Vented Low Pressure Containment (VLPC) tests proposed; however, details need to be worked out	DOE/INL	2013
FP speciation during mass transfer	All	Chemical change can alter FP volatility.[45] VLPC tests proposed; however, details need to be worked out	DOE/INL	2013
FP diffusivity and sorptivity on non-graphite surfaces	All	In-pile and out-of-pile loop tests to determine FP trajectory during normal operation and any trapping during transients	DOE/INL	2014 - 2020[46]
Coolant chemical interaction with surfaces	All	Out-of-pile loop tests to determine FP sorptivity due to changes in oxygen and carbon potential	DOE/INL	2014 – 2017
FP resuspension	All	In-pile and out-of-pile loop tests to determine FP resuspension due to flow-induced vibration	DOE/INL	2014 – 2020

[38] AGR experimental plan mentions air ingress testing but details have not being worked out yet, pending more information on specific NGNP configuration to determine the scope of air ingress experiments.
[39] Oxidation can fail particles by Outer Pyrolytic Carbon (OPyC) thinning due to oxidation and/or SiC oxidation ($SiC + O_2$ -> SiO or SiO_2); Particle failure fraction depends on the extent of the air supply, particle temperature and can be much greater than heatup without air ingress.
[40] Details of AGR experimental plan for moisture ingress testing need to be worked out, pending more information on specific NGNP configuration.
[41] SI stands for steam/water or moisture ingress.
[42] Oxidants reaching exposed kernels can significantly increase fuel particle fission product release depending on H_2O partial pressure and fuel temperature.
[43] Large pressure pulse can potentially over-pressurize and fail CFPs depending on energy deposition.
[44] DOE/INL currently plan to evaluate the results of Japanese testing of TRISO in NSRR to determine if more testing is needed in which case AGR particles can be tested in NSRR.
[45] Historical data for metals and oxides available. Their applicability to carbides and carbonyls is uncertain.
[46] Schedule contingent on specification of non-graphite surfaces. Little information is currently available on materials of interest.

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider	Schedule
Dust generation and characterization[47]	All	DOE NEUP, VLPC tests Potential experimental work at national lab(s) and international organizations	DOE/INL DOE/INL NRC	2015 TBD
FP absorption and desorption on dust	All	VLPC and in-pile loop tests to determine FP absorption and desorption	DOE/INL	2015 – 2020[48]
Aerosol nucleation and growth[49]	All	VLPC tests	DOE/INL	2015
Aerosol/FP/dust deposition including inertial, turbulent, thermophoresis, diffusophoresis, etc.	All	In-pile and out-of-pile loop tests Potential experimental work at national lab(s) and international organizations	DOE/INL DOE/INL NRC	2017 – 2020 TBD[50]
Dust deposition on vessel and RCCS hardware[51]	All	VLPC and out-of-pile loop tests to determine FP deposition	DOE/INL	2013 – 2016
Dust combustion in confinement	G-LOFC	TBD[52]	TBD	TBD
Cable pyrolysis and fire	G-LOFC	Affects iodine chemistry. LWR experience may be adequate	--	--
Ag-110m generation and transport	All	AGR-3, 4, and 8 tests, loop tests, FP sorption	DOE/INL	2014 - 2020

[47] A future PIRT-type dust workshop in March 2011 will address this issue in greater details. The findings from the workshop will be used to define the nature and scope of additional work in this area.
[48] Limited data will be available by 2015. These data will be used to assess models based on FP absorption and desorption data from LWR experience, and conservatism will be applied to account for HTGR-related uncertainties.
[49] Some historic data may be available but may not cover the regime relevant to HTGR. Again, conservatism will be applied to account for HTGR-related uncertainties.
[50] See footnote 41 above.
[51] Important for assessing radiation heat transfer in reactor cavity. Not important for FP transport.
[52] Data sources currently not identified. The dust workshop will also address the issue and define a path for resolving the issue.

Table 6. Major Milestones for Fuel Performance and FP Code Development

Activity Title	Activity Begins	Activity Completed[53]
PARFUME 2.2 version acquisition and benchmarks with historic data	1Q11	3Q11
PARFUME upgrade version acquisition and benchmarks with historic data	3Q11	1Q12
PARFUME model revisions with AGR fuel materials/physical properties data	4Q11	4Q15
PARFUME benchmarks against AGR-1 post-irradiation/accident FP release data	3Q11	2Q12
PARFUME AGR-2 irradiation/accident condition predicted FP releases (vs. data)	4Q11	3Q12
PARFUME AGR-3/4 irradiation/accident condition predicted FP releases (vs. data)	1Q12	4Q12
PARFUME model revisions with AGR 3/4 fuel FP diffusion data	2Q14	4Q14
PARFUME benchmarks against AGR-3/4 FP release test data	4Q14	1Q15
PARFUME AGR-5/6 irradiation/accident condition predicted fuel failure rates	1Q15	3Q15
PARFUME particle failure model validation using AGR-7 particle failure data	1Q17	1Q20
PARFUME FP transport model validation using AGR-8 FP release data	1Q17	1Q20
PARFUME NGNP fuel performance sensitivity studies	4Q12	1Q20
PARFUME NGNP fuel failure fraction response surface studies/predictions	4Q13	4Q20
PARFUME-MELCOR code-to-code benchmark studies	3Q11	2Q20
PARFUME code-to code/data international benchmark studies (if available)	1Q12	4Q16
Fuel fabrication and quality control inspection guidance	1Q08	2Q09 (c)

Plan for High-Temperature Materials Performance

Safety-Significant Issues and Phenomena

Creep and creep-fatigue crack growth of pre-existing flaws or flaws that initiated early in the service life of components such as reactor pressure vessels (RPVs), cross ducts, intermediate heat exchangers (IHXs), steam generators, etc., are a potential concern if they are not detected during in-service inspections (ISI). A macroscopic crack might grow to a critical size that triggers other structural failure modes such as creep rupture due to reduced section thickness or brittle fracture of ferritic components during heatup/cooldown. A crack may also grow through the wall of the component, leading to a breach of the pressure boundary or the primary/secondary boundary and causing fission product release and/or air/steam/water ingress.

[53] For activity completion dates beyond 2017, selected conservative measures will be incorporated into the evaluation model until data becomes available to account for uncertainties. This may result in compensatory measures in the design and operation of the NGNP.

Subcritical crack growth due to creep and creep-fatigue loading of NGNP components has been identified as a phenomenon that has a high importance ranking and a low knowledge level. Creep and creep-fatigue crack growth evaluation methodologies and analysis tools are necessary to support the independent assessment of the structural integrity of NGNP pressure boundary and metallic components under normal operating conditions, design-basis accident and beyond-design-basis conditions, and other conditions that may result in significant component degradation and failure.

Evaluation Models/Tools

The focus of the metallic materials evaluation model is the development of a time-dependent creep and creep-fatigue crack growth predictive methodology that could be integrated into the modular probabilistic fracture mechanics (PFM) computer code evaluating multiple degradation mechanisms in different components. The latter is currently under development at NRC. Figure 2 shows the metallic materials evaluation model that consists of three modules. These three modules comprise Methods Development, Implementation, and NGNP-Specific Crack Growth (CG) Data.

- *Methods Development.* The development of crack-tip parameters (CTPs) is the main focus of this group of R&D efforts. The NGNP candidate metallic materials exhibit three stages of creep behavior (primary, secondary, and tertiary). The approach to the development of crack-tip parameters (CTPs) is to perform crack-tip singularity analysis for each of the three creep deformation regimes. Once the time-dependent fracture mechanics methodology is developed, data analysis procedure would be available to determine the correlation between the CTP and the creep crack growth rate data.

- *Model Implementation into Probabilistic Fracture Mechanics* (*PFM) Code.* This module is involved with the implementation of the deterministic flaw evaluation procedure in the computer program module. It is anticipated that flaw evaluations using either best estimate or statistical upper limits for the crack growth rates could be performed by the computer program. After verification and validation, the deterministic flaw evaluation computer program will be incorporated into NRC's modular PFM computer code.

- *NGNP-Specific CG Correlations.* This module is involved with the development of CG correlations specifically for the materials of construction for NGNP RPV, IHX, cross duct, steam generator, etc. Both base metal and weldments will be included in the metallic materials evaluation model development. The required NGNP-specific CG correlations and material constants will be developed from confirmatory or new CG data.

Data Needs

A set of scoping tests to generate creep CG data will be needed to develop creep CG correlations. The specific material to be used for such scoping tests is not critical as this effort is a proof-of-concept demonstration of whether the CTP is effective in correlating CG data. Judging from the available information in the literature, representative Ni-based Alloy A800H and its associated weldment would be good candidates for the scoping tests because A800H has good creep strength in the temperature range of 750°C to 800°C, and is being considered as one of the candidate material for NGNP components operating at 750°C temperature. If the selected NGNP design accommodates an outlet temperature exceeding the above range and up to 950°C, the creep CG correlations need to be validated by conducting additional tests at higher temperatures for the selected materials. Knowledge gained from the ASME and ASTM codes and standards activities for these high temperature materials will be used in the methodology development.

Moreover, confirmatory NGNP-specific CG data may be needed to validate creep CG correlations and to support NGNP license review. Although the environment that these materials will be exposed to during service is impure helium and possibly steam, the test data will be generated primarily in the air environment. Thus, any potential material degradation mechanisms in impure helium and possibly in steam that could accelerate the CG rates as compared with those in the air environment will need to be addressed through additional tests simulating environmental effects.

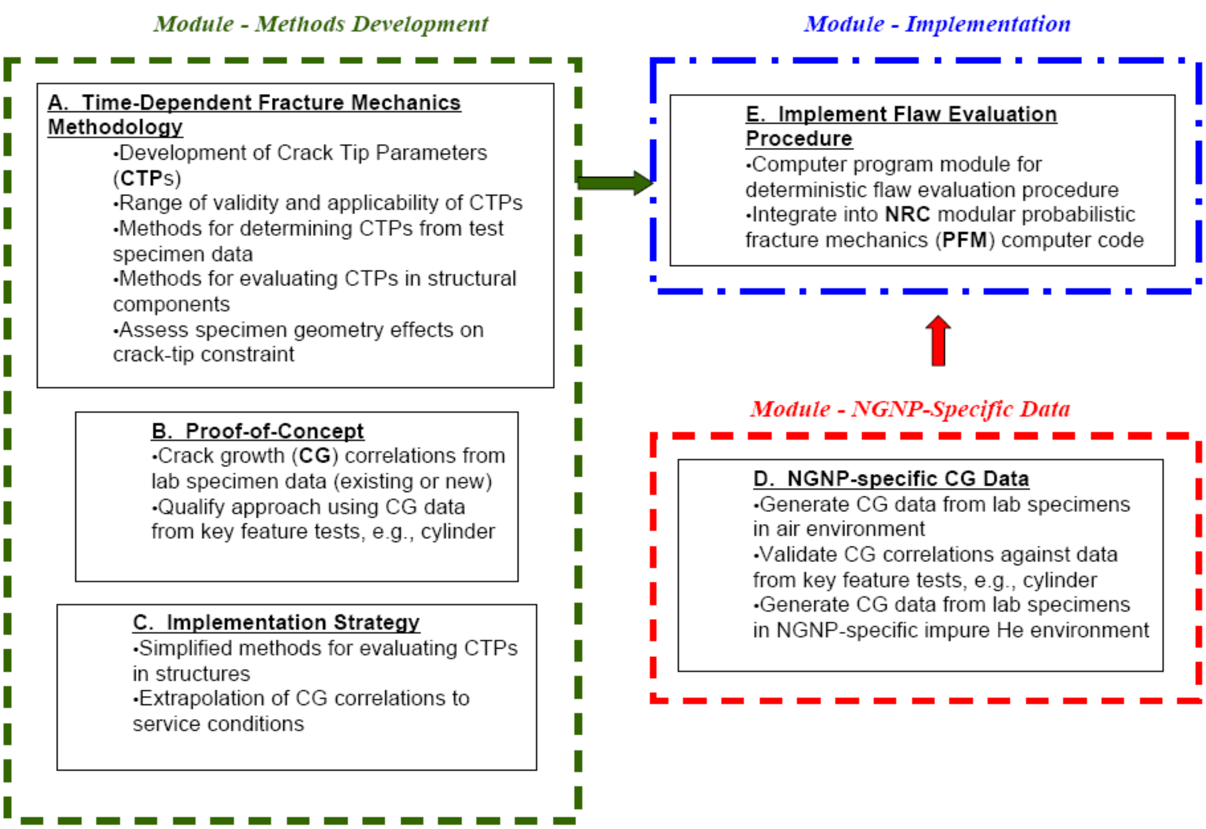

Figure 2. Metallic Material Evaluation Model and Modules.

Data Sources

Creep and creep-fatigue CG data for the NGNP materials in air, impure helium, and possibly in steam environment may be available in the DOE-supported Gen. IV Materials Handbook development. Any additional data generated by the NGNP vendors and the INL/NGNP materials group will be of value in determining what additional material performance data need to be generated. Data generation will be the responsibility of DOE or an applicant. NRC will use the data to develop and assess analysis tools. Table 7 identifies the organization(s) expected to provide the data. Table 8 identifies the major milestones for CG data development.

Table 7. Data Needs and Sources for High Temperature Material Performance[54]

Purpose/Application	Data Needs[55]	Scenario	Expected Provider[56]	Schedule[57]
1. Scoping data – <u>single heat, two temperatures</u>, Alloy 617 and/or Alloy 800H, to support the Methods Development Module of the flaw evaluation development roadmap, can leverage literature data if exist. **<u>Test Schedule</u>:** May 2011 to April 2012.	Air data from laboratory specimens, base metal and weldment (including heat-affected zones (HAZ)), weld process to be determined - to aid the analysis of crack growth test results o Monotonic stress-strain curves o Cyclic stress-strain curves o Creep curve data (with applied stresses from low (prototypical operating condition) to high stress)	NO	DOE	April 2012
	Crack growth data in air from laboratory specimens, for base metal and weldment (including HAZ), weld process to be determined o Creep crack growth (data record sufficient to determine crack growth incubation) o Fatigue crack growth (test conditions sufficient to determine Paris law for multiple R-ratios, one frequency) o Creep-fatigue crack growth (data record sufficient to determine interaction of creep and fatigue)	NO	DOE	April 2012

[54] Assumption: Materials of construction (e.g.: Alloy 800H, Alloy 617) and coolant chemistry selected by May 2011.
[55] An affirmative response (e-mail, dated 12/08/2010) to this NRC data needs and schedule was received from DOE/INL NGNP High Temperature Materials Lead staff (with CC to DOE/INL NGNP Regulatory Affairs staff), with the following remarks--"*Assuming we stay with Alloys 800H and 617 and we have the necessary levels of funding the dates in your table are consistent with our planning with one possible exception. That is the rather optimistic assumption that the decision on a steam generator or IHX will be finalized in spring 2011. That decision of course impacts the scoping studies of crack growth in the proper environment. The current plan is to run one crack growth system in air and the other in prototypical NGNP helium (since one side of either the steam generator or IHX would see the primary gas). We do not have steam explicitly in the current plan. It will be a major commitment to test under appropriate steam conditions and right now we are not prepared to make that investment*".
[56] Sources of data are DOE-sponsored high temperature materials programs at the Idaho National Laboratory (INL) or other organizations under cognizance of INL or the applicant.
[57] Refinement of the data needs and test schedule, and the development of detailed test matrices is planned to be carried out by INL June 2011.

Purpose/Application	Data Needs[55]	Scenario	Expected Provider[56]	Schedule[57]
	Proof-of-concept validation - key feature tests in air (surface crack in pipe or cylinder) with crack in base metal and in weldment for separate tests o Creep crack growth o Fatigue crack growth o Creep-fatigue crack growth	NO	DOE	April 2012
	Failure data in air – for failure mode criteria in remaining ligament o Tensile creep rupture strain, with limited creep rupture strain data from internal pressure and torsional loads o JIC data	NO	DOE	April 2012
2. Scoping data to assess environmental effect, <u>same heat and test conditions as scoping tests in air</u>, Alloy 617 and/or Alloy 800H. **<u>Test Schedule</u>:** May 2011 to Sept 2012.	Data to aid the analysis of test results - NGNP-specific coolant chemistry o Monotonic stress-strain curves o Cyclic stress-strain curves o Creep curve data (with applied stresses from low (prototypical operating condition) to high stress)	NO	DOE	Sept 2012
	Crack growth data from laboratory sp environment (impure He and/or stean weldment (including HAZ) o Creep crack growth (data record incubation) o Fatigue crack growth (test conditions sufficient to determine Paris law for multiple R-ratios, one frequency) o Creep-fatigue crack growth (data record sufficient to determine interaction of creep and fatigue)	NO	DOE	Sept 2012
	Failure data in NGNP coolant environment (impure He and/or steam), base metal and weldment (including HAZ) - for failure mode criteria in remaining ligament o Tensile creep rupture strain, with limited creep rupture strain data from internal pressure and torsional loads o J_{IC} data	NO	DOE	Sept 2012
3. Scoping data in air to assess aging effect, aging conditions to be determined, <u>same heat</u>	Air data to aid the analysis of test results o Monotonic stress-strain curves o Cyclic stress-strain curves	NO	DOE	Sept 2012

<table>
<tr><th>Purpose/Application</th><th>Data Needs[55]</th><th>Scenario</th><th>Expected Provider[56]</th><th>Schedule[57]</th></tr>
<tr><td rowspan="2">and test conditions as scoping tests in air, Alloy 617 and/or Alloy 800H.

Test Schedule:
May 2011 to Sept. 2012.</td><td>Crack growth data from laboratory specimens in air, for aged base metal and weldment (including HAZ)
o Creep crack growth (data record sufficient to determine crack growth incubation)
o Fatigue crack growth (test conditions sufficient to determine Paris law for multiple R-ratios, one frequency)
o Creep-fatigue crack growth (data record sufficient to determine interaction of creep and fatigue)</td><td>NO</td><td>DOE</td><td>Sept 2012</td></tr>
<tr><td>Failure data in air, aged base metal and weldment (including HAZ) - for failure mode criteria in remaining ligament
o Tensile creep rupture strain, with limited creep rupture strain data from internal pressure and torsional loads
o J_{IC} data</td><td>NO</td><td>DOE</td><td>Sept 2012</td></tr>
<tr><td rowspan="2">4. Design data – multiple heats (minimum 3), multiple temperatures, Alloy 617 and/or Alloy 800H, to support deterministic and probabilistic procedure development.

Test Schedule:
May 2011 to May 2013.</td><td>Air data to aid the analysis of test results
o Monotonic stress-strain curves
o Cyclic stress-strain curves
o Creep curve data (with applied stresses from low (prototypical operating condition) to high stress)</td><td>NO</td><td>DOE</td><td>May 2013</td></tr>
<tr><td>Crack growth data in air from laboratory specimens, for base metal and weldment (including HAZ), weld process to be determined
o Creep crack growth (data record sufficient to determine crack growth incubation)
o Fatigue crack growth (test conditions sufficient to determine Paris law, including R-ratio effect and frequency effect)
o Creep-fatigue crack growth (data determine interaction of creep a</td><td>NO</td><td>DOE</td><td>May 2013</td></tr>
</table>

Purpose/Application	Data Needs[55]	Scenario	Expected Provider[56]	Schedule[57]
	Failure data in air, for base metal and weldment (including HAZ) - for failure mode criteria in remaining ligament that could be triggered as a result of crack growth ○ Tensile creep rupture strain, with limited creep rupture strain data from internal pressure and torsional loads ○ J_{IC} data	NO	DOE	May 2013
5. Required **only** when scoping data demonstrate that there is a need to address environmental effects and/or aging effects. The test conditions will be a significantly smaller subset of those tests in air for the virgin materials. The same heats of Alloy 617 and/or Alloy 800H, and temperature conditions, as the tests for design data in air and virgin conditions will be used. The objective is to develop environmental degradation factors and aging factors, if needed. **Test Schedule**: Oct 2011 to May 2013.	Data to aid the analysis of test results ○ Monotonic stress-strain curves ○ Cyclic stress-strain curves ○ Creep curve data (with applied stresses from low (prototypical operating condition) to high stress)	NO	DOE	May 2013
	Crack growth data, laboratory specimens, base metal and weldment (including HAZ), weld process to be determined ○ Creep crack growth (data record sufficient to determine crack growth incubation) ○ Fatigue crack growth ○ Creep-fatigue crack growth (data record sufficient to determine interaction of creep and fatigue)	NO	DOE	May 2013
	Failure data, for base metal and weldment (including HAZ) - for failure mode criteria in remaining ligament that could be triggered as a result of crack growth ○ Tensile creep rupture strain, with limited creep rupture strain data from internal pressure and torsional load ○ J_{IC} data	NO	DOE	May 2013

Table 8. Major Milestones for Crack Growth Model and Data Development

Activity Title	Activity Begins	Activity Completed
Develop a time-dependent fracture mechanics methodology	2Q09	3Q11
Develop crack growth correlations from laboratory data	4Q11	4Q12
Generate NGNP-specific crack growth data	1Q12	1Q14

Activity Title	Activity Begins	Activity Completed
Validate crack growth correlations against NGNP data	3Q12	1Q14
Implement crack growth model into PFM code	2Q12	4Q14

Plan for Graphite Performance

Safety-Significant Issues and Phenomena

The graphite PIRT[58] identified significant issues which could potentially affect reactor safety due to the degradation of moderator and structural graphite core components (GCC) subjected to irradiation in HTGR. The evaluation considered both routine operation and postulated accidents. The PIRT identified seven major safety significant functional requirements for GCC. There are: (1) ability to maintain passive heat transfer; (2) maintain ability to control reactivity; (3) thermal protection of adjacent components; (4) shielding of adjacent components; (5) maintain coolant flow path; (6) prevent excessive mechanical load on the fuel; and (7) minimize activity in the coolant. A number of phenomena were identified which could potentially affect these requirements. The most significant phenomena included: (a) irradiation effect on graphite properties; (b) consistency of graphite quality and performance over the service life; and (c) potential generation of graphite dust, which could impact source term.

Of the several phenomena identified, five were ranked to be of high importance–low knowledge (I-H, K-L). A further nine were ranked to be of high importance and medium knowledge (I-H, K-M). Subsequently, NRC conducted a technical information gap analysis[59], by conducting a workshop inviting international graphite specialists with expertise in reactor design, gas cooled reactor regulations, and graphite material experts from recognized national laboratories, to identify specific technical areas, where the HTGR applicants'[60] and other current world-wide research do not address issues identified by the graphite PIRT. The workshop panel recommended that NRC staff develop a broad knowledge base in nuclear graphite technology and actively participate in the development of irradiation data, behavior modeling and interpretation, and codes and standards development, in order to conduct effective technical review.

Evaluation Models/Tools

Core Stress Analysis. The maintenance of the structural integrity of the GCC throughout the reactor life is ensured by limiting the stresses in GCC to values dictated by the applicant's reactor design and the code required factor of safety. The deformation and shape change of GCC due to irradiation must be limited to maintain the integrity of the coolant channel, fuel channel, and the control rod channel, assuring the free and unhindered movement of control rods and fuel rods, and the designed flow of the coolant. Independent confirmatory analysis of

[58] "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), Vol. 5, Graphite PIRTS, NUREG/CR-6944 (March 2008).

[59] "Milestone Report on the Workshop on Nuclear Graphite Research', ORNL/NRC/LTR-09/03 (September 2009).

[60] In the absence of a specific commercial applicant for design certification of HTGR, we assume that the DOE/NL NGNP research will provide the data and information on technical basis for the NGNP HTGR design.

GCC stresses and deformation provides confidence that these phenomena are appropriately modeled in NRC's evaluation model. Also, the deformation data will be used as input for confirmatory coolant bypass flow calculations.

To conduct independent confirmatory analysis of applicant's stress safety margins for GCC, the staff will develop a finite element stress analysis (FEA) code, in conjunction with the procedures developed by the ASME for GCC design, and determine spatial stresses for miscellaneous shape GCC, such as reflectors and moderators containing keyways and holes or channels for fuel rods, control rods, and coolant flow paths, and graphite core supports. This code will incorporate the inherently nonlinear elastic stress-strain behavior of graphite, variability in properties due to non-homogeneity; spatial variation in temperature and flux due to core design; and the contribution and role played by irradiation creep of graphite in governing properties. The time (dose)-integrated stress calculation will use as inputs the time (dose)-dependent coefficient of thermal expansion, thermal conductivity, nonlinear elastic modulus, Poisson's ratio, material loss factor due to oxidation or other corrosion mechanisms from coolant chemistry changes, and dimensional changes. The data sources will be from reactor applicant and the DOE research. The model and the procedures will be validated and verified using ASME code[61], and DOE and other vendor data and benchmark calculations on idealized core component shapes. The staff can use this FEA tool, projected to be available by 2013, to confirm applicant's designed deformation limits for GCC.

Graphite Performance and Qualifications. The staff will continue to monitor DOE research in this area. The information provided by the reactor design applicant and the DOE research will be used by NRC staff for review and acceptance.

Predictive Models for Graphite Deformation and Shape Change. The staff will continue to monitor DOE research in this area. It will be the responsibility of the reactor applicant and the DOE research to provide this information to NRC. The FEA code, to be developed as stated above, will be used to predict GCC deformation and eventually to predict deformation of the GCC assembly. It will also be used as input for coolant bypass flow calculations.

Graphite Oxidation Model. The staff will continue to monitor DOE research in this area. The reactor applicant and DOE research will be expected to provide, for NRC staff evaluation, data and the model of graphite oxidation, including information on the diffusion of critical species, which have effects on oxidation, and on the physical, thermal, and mechanical characteristics of nuclear graphite.

Graphite Dust Generation. The staff will continue to monitor DOE research in this area. The staff will evaluate the data provided by the future applicant and use other published information from DOE research. The staff will participate in the dust issues assessment workshop during March 2011, and will continue to monitor data development in this topic so that the dust effects could be appropriately included in the review of applicant's HTGR GCC design.

Consensus Codes and Standards. As per Commission direction[62], the staff will continue to participate in the ongoing codes and standards activities of the standards organizations and provide staff input and guidance to help assure promulgated codes and standards can be

---

[61] The ASME (Div 5, Sec III B&PV) design code for HTGR graphite core components is planned to be published during 4th Qr. 2011.

[62] SRM – SECY-030047 – "Policy issues Related to Licensing Non-Light Water Designs", ADAMS ML ML031770124, July 26, 2003.

endorsed in the regulatory process in assessing the structural integrity of GCC. The staff participation will ensure that the code activities consider methods development for (1) flaw acceptance criteria (numbers, sizes, orientation and their distribution) for as-fabricated graphite components; (2) flaw evaluation procedures, including method for the safety significance assessment of detected flaws, and disposition requirements for flaws detected during HTGR inspection; and (3) baseline and in-service inspection (ISI) requirements using qualified procedures and inspection personnel. These activities will be the responsibility of the reactor applicant and the consensus code development organizations.

Data Needs

As per DOE schedule[63], some of the data for NGNP graphites may not become available by 2013, when a design certification application is expected to be submitted to NRC. Therefore, staff might have to use other data for a similar graphite class[64], along with the applicant's technical basis for designing the HTGR GCC, to support staff technical safety evaluations. In the absence of properties data as a function of complete cumulative damage dose of the reactor life, the staff could decide to give a conditional operational license for limited initial period of operation ensuring adequate public safety, pending complete (namely, high irradiation dose) data from irradiation and confirmation of existing data by actual HTGR initial operation. The staff will also continue to participate in the IAEA CRP[65] on graphite irradiation creep research and continue to acquire and maintain scientific knowledge and expertise in this topic.

As identified by the graphite PIRT and the data gap analysis, the following phenomena have the highest importance with low knowledge (I-H, K-L). However, all these phenomena are being addressed. The data generated for NGNP graphites by either DOE or other worldwide research (or both), along with the applicant's data, will be used for staff evaluation of the HTGR design. These data needs involve:

1. Irradiation-induced creep (irradiation-induced dimensional change under stress)
2. Irradiation-induced change in CTE, including the effects of creep strain
3. Irradiation-induced change in thermal conductivity
4. Irradiation-Induced changes in mechanical properties (strength, toughness), including the effect of creep strain (stress).

No research is currently being conducted for two other phenomena that had an I-H, K-L ranking. These phenomena are: (a) blockage of fuel element coolant channel due to graphite failure and/or graphite spalling; and (b) blockage of coolant channel in reactivity control block due to graphite failure and/or graphite spalling. Thus, the applicant would presumably be expected to ensure a very low probability of occurrence of these phenomena by appropriate design and provide analytical data to support such assurance.

The nine phenomena identified in the graphite PIRT as having high importance to safety with medium knowledge (I-H, K-M) are listed below. DOE and other worldwide research, along with

---

[63] INL/EXT-07-13165, Rev. 1, "Graphite Technology Development Plan", Idaho national Laboratory, October 2010

[64] ASTM D 7219-05, "Standard Specification for Isotropic and Near-isotropic Nuclear Graphites", American Society for testing Materials", (2005).

[65] Coordinated Research Program. Irradiation creep effects have become more dictating for end of life dose range for British AGRs.

the future HTGR applicant's data, should be sufficient for NRC staff review of the applicant's design in these areas. These phenomena are:

1. Statistical variation of non-irradiated properties
2. Consistency in graphite quality over the lifetime of the reactor fleet (for replacement, for example)
3. Irradiation-Induced dimensional change
4. Irradiation-induced thermal conductivity change
5. Irradiation-Induced changes In elastic constants, including the effects of creep strain
6. Tribology of graphite in (impure) helium environment
7. Degradation of thermal conductivity
8. Blockage of reactivity control channel due to graphite failure (spalling)
9. Graphite temperature.

Except for the phenomenon 8 above, the DOE and/or external research is expected to provide sufficient data for staff evaluation of applicant's design. Item 8, which is related to graphite spalling, is expected to be addressed by the reactor vendors in their design and their technical specifications for reactivity control rod movement. As stated previously, as per the current DOE schedule, some of the data for the I-H, K-M phenomena will not be available by 2013 for NGNP graphites, when a design certification application is expected to be submitted to NRC. Therefore, NRC will consider using other data for similar graphite classes to support staff technical assessment, along with the applicant's technical basis for designing the HTGR GCC, to ensure safe initial operation.

Data Sources

The staff expects the HTGR applicant to provide complete data, technical information, and technical basis to support the design functions of GCC throughout the reactor life. The staff evaluation of the design will rely on the applicant-provided information.

The staff will continue to engage in the activities of the codes and standards organizations, such as ASME and ASTM, to ensure that potential regulatory technical issues are addressed by these organizations so that the staff can endorse these codes and standards when they become available for staff review. The staff will also actively participate and monitor DOE and worldwide research, within the provisions of the NRC-DOE MOU[66]. When additional GCC potential technical safety issues for regulatory review are identified, NRC will alert DOE to any additional data needs.

In the near term, the staff will conduct independent research in two major areas. The first is exploratory research on the release of stored energy of graphite due to irradiation when the irradiated graphite is heated subsequently to temperatures greater than the irradiation temperature. Such a scenario is possible, for example, in a loss-of-coolant accident. This could potentially lead to "runaway" and uncontrollable temperature increase, which could then contribute to very high levels of loss of graphite due to rapid oxidation. This type of so-called Wigner energy release has generally been discounted for HTGRs[58]. However, an ACRS

[66] "MOU between DOE and the NRC for NRC participation in the NGNP", ADAMS ML080600047 (2008).

member has raised this as a potential safety issue[67], and the workshop panel for gap analysis also indicated that low knowledge exists for this high safety impact phenomenon. Literature data[68]show that, in addition to the "200 ^{o}C" rate-of-release peak, there is another peak in the rate-of-release curve around 1200 ^{o}C for graphites irradiated at lower temperatures. Therefore, in May 2010, RES started an exploratory research at Oak Ridge National Laboratory on this subject. Results will be available on an ongoing basis. The contract is projected to end in September 2012.

NRC will pursue confirmatory research in a second area. The staff will develop an independent capability to conduct time (dose)-integrated, non-linear, 3-dimensional FEA for GCC. The input data for model and procedure development will be from DOE/INL/ORNL and other worldwide research. The model and the procedures will be validated and verified using ASME code[69], and DOE and other vendor data and benchmark calculations on idealized core component shapes. The staff can use this FEA tool, projected to be available by 2013, to confirm the applicant's assumptions, stresses, safety margins (factor of safety), and the retention of safety margins over the reactor life. The staff can also use this tool to perform confirmatory analysis of applicant's designed deformation limits for GCC.

Table 9 shows the identified data needs and the sources from which such data are expected to be available in the future. As in the case of fuel performance R&D, it is also recognized for graphite R&D that having an adequate amount of data in the timeframe for licensing a prototype NGNP is important; however, much of the data may be generated in a longer timeframe. Table 10 identifies the major milestones for graphite performance R&D.

Table 9. Data Needs and Sources for Graphite Performance

Process/Phenomena[70]	Scenario	Data Sources and Types	Expected Provider	Schedule
Irradiation-induced creep (irradiation-induced dimensional change under stress) leading to fuel element/control rod channel distortion/bowing	NO	DOE (INL and ORNL) research program; EU (Petten) research program; IAEA ICP on graphite creep	DOE	2013 - 2020[71]
Irradiation-induced change in thermal conductivity, CTE, including the effects of creep strain, leading to fuel element/control rod channel distortion/bowing	NO	DOE (INL and ORNL) research program; EU (Petten) research program	DOE	2013

[67] D.A. Powers, "TRIP REPORT" - Travel by D.A. Powers to Attend the High-Temperature Gas-Cooled Reactor Safety and Research Issues Workshop Rockville, MD., October 10-12, 2001. ADAMS Accession Number: ML020450645.
[68] J Rappeneau, J L Taupin, J Grehier, "Energy released at high temperature by irradiated graphite", *Carbon*, **4** (1) 115-124 (1966)
[69] The ASME (Div 5, Sec III B&PV) design code for HTGR graphite core components is planned to be published during 4th Qr. 2011.
[70] Phenomena/processes included in the table were ranked high in importance and low/medium in knowledge base by the PIRT panel and/or recommended by experts at the graphite workshop (see NUREG/CR-6944, Vol. 5 for the PIRT panel recommendations and ORNL/NRC/LTR-09/03 for graphite workshop recommendations
[71] Initial irradiation data at 600^{o}C will be available from AGC experiments in 2013; data at lower fluence and at all temperatures will be available in 2016, and all data including those at higher fluence will be available in 2020. All non-irradiated material properties data will be available by 2015.

Process/Phenomena[70]	Scenario	Data Sources and Types	Expected Provider	Schedule
Irradiation-induced changes in mechanical properties (elastic constant, strength, toughness), including the effect of creep strain (stress), leading to graphite fracture	NO	DOE (INL and ORNL) research program; EU (Petten) research program; IAEA ICP on graphite creep	DOE	2015
Graphite failure and/or graphite spalling leading to blockage of fuel element coolant channel and control rod channel	All	Vendor	DOE vendor	2016
Statistical variation of non-irradiated properties	NO	DOE (INL) research program	DOE	2015
Consistency in graphite quality over the lifetime of the reactor	NO	DOE (INL) research program (vendor)	DOE vendor	2013[72]
Degradation of thermal conductivity	NO	EU (Petten) research program	DOE	2013
Graphite temperature[73]	NO	DOE research	DOE	2016
Tribology of graphite in (impure) high-temperature helium environment	NO	Vendor	vendor	2014
Modeling oxidation behavior, including kinetics and the effects of diffusion of controlling species	D-LOFC (AI, SI)	DOE research	DOE	2016
Graphite codes and standards	All	Active participation in consensus code development efforts by ASME and ASTM	DOE NRC	2016[74]
Oxidative reactivity of graphite dust/powder	All	DOE research	DOE	2016
Mechanistic modeling and predictive capability for irradiation-induced dimensional change and creep (NRC graphite workshop)	NO	DOE (INL and ORNL) research program; EU (Petten) research program; IAEA ICP on graphite creep	DOE	2015
Failure criteria for irradiated graphite	All	DOE (INL and ORNL) research program; EU (Petten) research program; IAEA ICP on graphite creep; ASME Codes and Standards	DOE	2013[75]
Graphite core component volumetric inspection (production and in-	All	DOE research and vendor	DOE vendor	2016

[72] It is expected the applicant will assure consistency at the time of submitting an application (expected to be 2013). However, it is not clear how this will be achieved.

[73] Past British AGR operational data indicate inconsistencies in core graphite temperature that has been demonstrated as contributing to inconsistencies in predicted dimensional changes observed during inspection. In the absence of reliable core temperature information, the staff may assess applicant's temperature sensitivity analyses to important GCC core behavior models, which affect safe reactor operation, to ensure the use of adequate conservatism in GCC design.

[74] The schedule may be somewhat optimistic given that the entire range of graphite data may not be available prior to 2020.

[75] Only preliminary data will be available by 2013. Failure criteria development based on preliminary data will need to be validated and verified with the entire set of data at a later date (2020 and beyond).

Process/Phenomena[70]	Scenario	Data Sources and Types	Expected Provider	Schedule
service) and online monitoring[76]				
Capability to verify graphite core stress analysis methods	All	NRC research	NRC	2015
High-temperature stored energy release	All	NRC research	NRC	2012
Graphite decommissioning technical issues		DOE research	DOE	2018

Table 10. Major Milestones for Graphite Performance

Activity Title	Activity Begins	Activity Completed
Establish the effect of potential for release of stored energy	2Q10	4Q12
Develop probabilistic stress analysis code for graphite structures	3Q10	3Q13
ASTM and ASME consensus standards review and endorsement	1Q12	4Q16

Plan for Process Heat Applications

Safety-Significant Issues and Phenomena

The original intent of the EPAct was to utilize an NGNP prototype plant for cogeneration of hydrogen. The more recent trend in the industry and the end-user community is to explore wider utilization of the NGNP heat output to an array of diverse chemical process products. The plan described in this section addresses the hydrogen cogeneration aspect in compliance with the EPAct mandate. However, the plan is sufficiently general in most aspects so it is able to accommodate other forms of process heat applications.

Deflagration/detonation of hydrogen from a collocated hydrogen cogeneration plant may produce a blast wave resulting in incident blast loading of the reactor plant. Combustion of other flammable gaseous products from a process plant and any potential impact on structural integrity of process heat components is another safety issue. Failure of process heat components could lead to failure of intermediate heat exchanger loop or could compromise its performance. The deleterious effects of toxic and corrosive “ground hugging gases” are of safety concern to both plant equipment and plant personnel. Finally, deleterious effects of tritium migration in process heat utilization products, as well as radiological health effects of tritium exposure, are potential concerns.

Evaluation Model

Blast Over-Pressure Loading. An evaluation model will be implemented to predict incident blast over-pressure loading on a reactor. This model will be based on already existing analytical tools, correlations, or software and will be capable of predicting the incident blast over-pressure loading on the reactor containment as a function of the separation distance between the

[76] Part of reliability and integrity management activity.

containment and the hydrogen plant. Moreover, the model will be capable of simulating general deflagration and combustion events and calculating radiative and convective heat flux projected upon the reactor building(s) and the blast overpressure and impulse shape from a combustion event. The model will be assessed against existing experimental data based on scaling, developmental assessment database, calculation uncertainty, and simplifying assumptions.

Plume Dispersion. An evaluation model will be implemented to predict development of toxic gas plumes, particularly concentrations of heavy gas release at specified distances from the reactor building(s). Again, this model will be based on already existing analytical tools, correlations, or software.

Thermal-Fluid Behavior of Process Heat Components. An evaluation model is needed to predict thermal-fluid behavior of process heat components. The model will provide secondary flow conditions for modeling/evaluation of intermediate performance and integrity. Moreover, the model will be able to predict interaction of NGNP and hydrogen production plant because of transients in either. This will be done by extending the NGNP core/system evaluation model concept to envelope the process heat components. It may be necessary to couple this extended evaluation model to any existing chemical process software through the development of a software interface.

Tritium Migration. Currently, no tritium activity limits exist for product hydrogen. It will be necessary to establish a measurable regulatory activity of tritium that can be detected in the intermediate coolant loop, possibly through the use of a beta radiation detector submerged within the gas during NGNP operations. It should be noted that the U.S. Environmental Protection Agency is responsible for defining tritium content guidelines for product hydrogen.

Data Needs

Data for blast-wave-related model development and assessment already exist and can be utilized. Likewise, data for plume behavior of toxic, corrosive, and ground hugging gaseous products exist and can be utilized. Plant-specific design data for both the reactor plant and the process heat plant will be needed to develop models for heat and mass exchanges between NGNP and process plants and for modeling NGNP response to hydrogen plant transients.

Data Sources

As already mentioned above, data for blast-wave-related model development and assessment already exist and can be utilized. Likewise, data for plume behavior of toxic, corrosive, and ground hugging gaseous products exist and can be utilized. All other data are expected to be generated by the designer/vendor organizations. Table 11 identifies the major milestones for process heat applications R&D.

Table 11. Major Milestones for Process Heat Applications

Activity Title	Activity Begins	Activity Completed
Evaluation model for incident blast overpressure loading	4Q11	4Q12
Evaluation model for development of toxic/corrosive gas plumes	4Q11	4Q12

Activity Title	Activity Begins	Activity Completed
Modeling of heat and mass exchanges between NGNP & process heat utilization plant	4Q11	4Q12
Evaluation model for NGNP response to process heat plant transients	4Q12	4Q13
Tritium migration modeling	4Q11	4Q12

Plan for Structural Analysis Tools Development

Safety-Significant Issues and Phenomena

The operating temperatures of the primary reactor vessel for high temperature reactor designs being considered are significantly greater than those for currently licensed nuclear power reactors. Therefore, depending on the effectiveness of the reactor vessel insulation and cooling system, the concrete reactor building could experience a high-temperature environment. The performance of reinforced concrete structures is severely impaired by the loss of bond under high temperature when thermal mismatch and load-induced thermal strains effects lead to separation between concrete and reinforcing steel. The rate of heating and cooling at elevated temperatures also affects the distribution of self-equilibrating thermal stresses and the formation of large compressive stress gradients leading to spalling. In addition, part of the plant concrete structure may experience radiation fluencies levels that have to be considered in the initial plant design loads.

In a multimodule HTGR plant, the nuclear island consists of several modules constructed at various stages and placed on a common foundation mat. Both the seismic capacity and the seismic response of the plant depend on the overall foundation size of the plant and the interaction of various modules. Besides, stability of stacked fuel compacts (in a prismatic reactor) or compaction of fuel spheres (in a pebble-bed reactor) under seismic loading may become an issue for structural integrity of the core as well as its neutronic and thermal response.

Current soil-structure interaction (SSI) computer codes and design criteria used in the industry have been based on structures that are founded at or near the earth surface and only have partially embedded foundations. Because of the lack of experience on seismic response of deeply buried nuclear type structures, research insights will be needed to evaluate the responses of new reactor plant structures that may be deeply or completely buried in-ground.

Evaluation Models/Tools

The focus of the evaluation model development effort is on methodologies to consider the reduced stiffness and strength of concrete by a combination of damage and plasticity models. Temperature dependence of the stiffness and strength may be calibrated from available test data in the literature. To address the spalling of concrete, the effect of the change in porosity because of progressive micro-cracking on the buildup of pore pressure needs to be considered. Hence, fully coupled hygro-thermo-mechanical simulation may the most appropriate approach to analyze spalling in a quantitative manner. Depending on whether a concrete wall is subject to uniform (same temperature gradient applied to the entire reinforced concrete wall) or nonuniform heating of the concrete surface, the coupled temperature and pore pressure could be solved as a one-dimensional or multidimensional field problem. The work scope further includes a review of existing codes used to address and evaluate the behavior of reinforced concrete structures when subjected to sustained high temperatures.

The evaluation model also will address the effect of irradiation on structural graphite and other structural components. Moreover, the seismic response of core and other structural components will be evaluated using nonlinear seismic analysis of reactor vessel and core support structures and taking into consideration seismic capacity of nuclear fuel and the effect of modular construction. The evaluation of the core and other structural components is planned to be done in-house with the use of the LS DYNA finite element computer code.

Analytical and experimental research needs to be conducted to develop independent capability for determining SSI effects and passive earth pressures on deeply embedded or buried structures during earthquakes. In February 2006, RES published NUREG/CR-6896 "Assessment of Seismic Analysis Methodologies for Deeply Embedded Nuclear Power Plant (NPP) Structures." In March 2008, NUREG/CR-6957, "Correlation Analysis of JNES Seismic Wall Pressure Data for ABWR Model Structures," was published as a follow-up study. NUREG/CR-6957 compared measured and computed earth pressures for low-level shaking of a 1/10 scale power plant. Results showed a poor correlation between calculated and measured earth pressures. NRC activities to address this issue are described below.

Data Needs

In the past, tests were conducted in the United States, Japan, France, Germany, and Spain to generate data on concrete behavior exposed to high temperatures. The existing concrete high-temperature data will be evaluated to determine their applicability to HTGR structures and components. The evaluation will cover concrete physical properties (stiffness, strength, bond, etc), and a review of design and evaluation criteria. For design conditions that exceed established limits, experimental work may be necessary to characterize mechanical and physical concrete properties to avoid conservatism.

Data on nonlinear static and dynamic analysis of long fuel compacts, pebbles, and core support structures will be needed to develop analytical models, perform analyses, and determine the limitations of existing finite element codes and data on the seismic margin of HTGR fuel as well as seismic capacity and response of multimodule HTGR plants.

The poor correlation between measured and calculated earth pressures reported in NUREG/CR-6957 demonstrates a need for additional research to assess existing analytical tools and develop recommendations on their appropriate use for calculating soil pressures and structural response of partially to fully embedded nuclear power plants under seismic loading conditions. The timeline for research that addresses this need is provided in Tables 12 and 13.

Data Sources

Data for the safety analysis of HTGR will be obtained from a variety of sources including data in the open literature, DOE-funded NGNP research, international cooperative agreements, vendor data, and NRC-funded programs.

Table 12. Data Needs and Sources for Structural Analysis of HTGR

Process/Phenomena	Scenario	Data Sources and Types	Expected Provider	Schedule
Concrete properties at high temperature	NO	DOE (ORNL); open literature	NRC	2010 (c)
Confirmatory analysis of seismic response of reactor vessel and core support structures	NO	Future NRC research program	NRC	2015
Irradiation-induced changes in mechanical properties (elastic constant, strength, toughness), including the effect of creep strain (stress), leading to graphite fracture **<u>Note:</u>** T*his is one of the sources listed under the Graphite section of plan that will be used for structural analysis and evaluation.*	NO	DOE (INL and ORNL) research program; EU (Petten) research program; IAEA ICP on graphite creep	DOE	2015
Irradiation effects on concrete properties	NO	DOE (ORNL), research program Japan- Nuclear and Industrial Safety Agency research (?)	NRC	2012
Lateral earth pressures on foundation walls and floors during seismic events	NO	Commercial contract research program. Japan – Japan Nuclear Energy Safety (JNES) for test data	NRC	2015

Table 13. Major Milestones for Structural Analysis Tools Development

Activity Title	Activity Begins	Activity Completed
Evaluate predictive models for concrete behavior at high temperatures	1Q08	1Q11
Radiation effects on concrete	2Q10	3Q12
Assess existing analysis tools for calculating earth pressures on foundation walls	1Q11	1Q13
Perform parametric study to identify key parameters that affect calculated earth pressures	1Q13	1Q15
Provide recommendations for licensing criteria for concrete evaluation and earth pressures	1Q11	4Q16

Plan for Digital Instrumentation and Control

Safety-Significant Issues and Phenomena

The combination of high temperatures and potentially corrosive process fluids and environments in VHTR plants can impose significant challenges to the design of plant instrumentation. The severe environmental conditions could significantly impact instrument reliability and accuracy. Three-dimensional mapping of core temperature and flux distribution will be challenging because of restrictions on locations for sensors. Meeting this challenge is likely to require the use of new kinds of sensors as well as innovative approaches to mathematical modeling and to the physical design of the reactor. Other new types of sensors and instrument systems also are likely to be required such as a monitoring system to detect damaged or depleted fuel spheres in a pebble-bed reactor. Research related to instrumentation such as described here would be needed for reviewing the NGNP submittal.

Advanced reactor designs such as VHTR would employ advanced control concepts and advanced control system implementations and equipment. In addition, the particular control schemes implemented in VHTR will necessarily differ from those in LWR because of the fundamental differences in the physical processes. Some technical concerns that are not crucial in the laboratory experiments may become very challenging in full-scale production. Robust control and protection system design is crucial, and fault detection may need to be incorporated into control and protection system design. Research related to advanced control system implementation and configurations will be applicable to new reactors and, to a lesser extent, to existing reactors as well as to VHTR.

Advanced diagnostics and prognostics (AD&P) refers to provisions for the detection of incipient failures in mechanical equipment and for self-monitoring by digital equipment. Introduction of AD&P capabilities can increase some aspects of system unreliability as a result of increased complexity and as a result of potential failures related to the AD&P process itself. AD&P would be useful for any type of reactor, new or existing.

Evaluation Models/Tools

Modeling of phenomena related to digital instrumentation and control (DI&C) is not anticipated. However, tracking the development of NGNP and heat application process models suitable for control and protection system design will be performed. Understanding these models will help NRC to review the control and protection system design and to make sure that it has adequate safety/stability margins. The research efforts will produce information for use in the evaluation of DI&C aspects of license applications such as technical guidance and acceptance criteria for DI&C.

Development of software-based tools for D&IC is not presently anticipated. The research efforts will produce information on state-of-the-art technologies of the VHTR for hydrogen production, required protection and control systems likely to be used, requirements and potential issues of plant models in the design of the control and protection systems, advanced control and protection system design methods, and modeling error limits. This information will be used to develop technical guidance and regulatory acceptance criteria.

Data Needs

The R&D efforts will involve literature review and consolidation of already-published information and relevant data. The review will track the latest DI&C design development for NGNP and will cover details of potential control and protection system designs for VHTR, methods and procedures to be used in the design of these protection and control systems, critical parameters that need to be controlled, and the conditions to enable the protection systems. No experimentation is presently anticipated. The R&D efforts will take into account process and environmental condition information projected for the reference reactor designs.

Data Sources

The focus in this area will be on review of existing data and other information. Table 14 identifies the major milestones for these reviews.

Table 14. Major Milestones for DI&C

Activity Title	Activity Begins	Activity Completed
Advanced instrumentation	3Q09	3Q12
Advanced controls	3Q09	3Q12
Advanced diagnostics and prognostics	4Q09	3Q12 (tentative)

Plan for Human Factors Analysis Tools Development

Safety-Significant Issues and Phenomena

The next generations of nuclear power plants (NPPs), including the new designs that are available today (Gen III) as well as the advanced reactor designs (Gen IV), are different in several respects from the currently operating NPPs (Gen II). First, in terms of the reactor design, the “near-term” next-generation plants generally rely on passive rather than active safety features or, for later deployment, they may be non-light-water designs and may involve concurrent control of multiple modules from a common control room.

Second, the next-generation plants will be designed using DI&C technology as opposed to the predominantly analog instrumentation and control (I&C) technology used in the current fleet of plants. DI&C systems provide the capability to implement control algorithms that are more advanced than have been used in plants to date, including nonlinear control methods, fuzzy logic, neural networks, and adaptive control. These systems will provide the capability for increased automation that makes greater use of interactions between personnel and automatic functions. These capabilities could lead to more intricate control of plant systems and processes and greater complexity.

The third key difference between the current and new and advanced plant designs is their human-system interface (HSI). Rather than using analog controls and displays, the next-generation plants are being designed with computer-based HSIs organized into sit-down workstations. Operators will be monitoring the plant through screen-based displays selected from networks of hundreds or even thousands of display pages. Control of plant equipment will be accomplished through soft controls that can be accessed through computer workstations, and computer-based procedures will be employed that allow control actions directly from the procedure display or they may be automated with the operator authorizing the procedure's embedded control functions to take actions.

Taken together, the advances in reactor design, I&C technology, and HSIs will lead to concepts of operation that are different from currently operating NPPs. Different qualifications and training will likely be required for the plant staff to maintain digital systems and to focus decisionmaking on monitoring and bypassing automatic systems rather than the active control that LWR operators now take. The trend toward increasing automation and fully computerized HSIs also will affect the tasks that plant personnel perform and the manner in which they perform them. Designers of advanced reactors will have to develop and validate concepts of operation that incorporate these new tasks as part of their design certification applications, and NRC staff will need to be prepared to evaluate both the concepts of operations and the adequacy of the human-system interfaces (HSIs) for supporting reliable human performance. In addition, the methods and tools used for the design and evaluation of next-generation plants have become more sophisticated, and the use of human performance modeling and virtual reality is now commonplace.

In anticipation of these changes, RES sponsored a study to identify human performance research needed to support staff reviews of the emerging technologies that will be found in new (Gen III) and advanced (Gen IV) NPPs. The study identified 64 potential human performance research issues associated with the introduction of new and emerging technologies. These 64 research issues are either (1) not addressed at all in NRC's current human factor regulatory review because of technological advances since the review guidance was last revised or (2) are not addressed in sufficient detail to support the desired level of consistency in staff reviews. The 64 issues were binned into 7 high-level human factors topic areas. Next, a Phenomena Identification and Ranking Table (PIRT) methodology was employed by 14 independent subject matter experts to prioritize the candidate research projects into 1 of 4 priority categories. A total of 20 different research topics were identified as being of the "highest priority" (see NUREG/CR-6947).

Identifying potential human performance issues associated with emerging technology in new and advanced NPPs was a challenging task. At present, only a few Generation III plants are in operation. Their operating experience is limited and not generally available in the literature. For NPPs that have yet to be designed and built, information concerning their operations or the design of their control rooms is limited at best, especially for designs of longer-term deployment such as the Gen IV advanced plants. Given the current state of knowledge, the differences between new and advanced NPPs exist more in the reactor design, and less in the design of the control room. Therefore, the research that is needed to address the human performance issues brought about by the emerging technology that utilizes DI&C and computer-based HSIs is equally applicable for both new and advanced reactor designs.

Although most of the research presented below is applicable to both new and advanced reactor designs, one area of research is directly needed to support the NRC staff reviews of advanced reactor applications, namely, advanced concepts of operation (CONOPS).

Human Factor Aspects in CONOPS of Modular Design. The concept of operating multiple reactors from one workstation introduces safety-critical performance considerations such as operation of multiple reactors by a single crew member. Moreover, it raises questions regarding situation awareness and workload in upset conditions that affect more than one reactor concurrently. Humans are more likely to make errors when they are required to allocate attention between different sets of systems. Additional research is needed to identify the impact on human performance of these changes to current concepts of operations.

The following research, although not advanced reactor specific, also is needed to support the review of licensing applications with new and advanced operational concepts and interfaces.

- *Automation and HSI Complexity.* The overall level of automation in new and advanced NPPs is expected to be much higher than in today's plants. The increased use of automation and reliance on passive systems will change human involvement in operations. Poorly designed automation can result in a loss of vigilance, low workload, complacency, and poor operator situation awareness. Thus, it is important to understand how automation impacts operators and whether the level of automation in a plant is acceptable from a safety perspective. The high levels of complexity inherent in the emerging technology, paired with the various levels of potential automation, have implications for the design of the HSIs. DI&C systems can provide many different types of dynamic information at a rapid pace. Information and their HSIs may be presented in complex and opaque ways that cause operators to miss or misunderstand critical information that is of particular concern to safety. Additional research is needed to assess the impact of automation systems on operator performance and human reliability and to identify the safety aspects of HSI complexity.

- *Computer-Based Procedures (CBPs).* The emerging trend of CBPs in new and advanced reactors introduces many human factors issues that may affect operator and plant performance and ultimately affect plant safety. Although they represent an important means of presenting integrated information to operators and providing them with guidance on the control actions required, CBPs fundamentally alter how operators process and share information and develop and maintain situational awareness. Additional research is needed to establish Human Factors Engineering (HFE) principles for acceptable CBP design.

- *Human Performance Impacts of Degraded I&C and HSIs.* Increased reliance on DI&C and computer-based HSIs increases the likelihood that crews will be faced with unplanned, unanticipated events involving degraded DI&C as well as HSI degradations and failures. Responding to such events may involve many aspects of HSI design, including alarm systems, CBPs, and other decision aids that are intended to support crews' situation assessments, decisionmaking, and response actions. Additional research is needed to develop a better understanding of how these systems should be integrated to ensure that operators are able to manage novel events and to identify the characteristics of the backup systems needed when the displays and controls normally available to operators are degraded or have failed.

- *Workload, Situational Awareness, and Teamwork.* The advanced technologies that are being developed with the intention of improving plant performance may have both positive and negative impacts on operator performance that may ultimately affect plant safety. For example, high levels of automation may increase task performance efficiency but reduce operators' awareness of plant situations; poorly designed human-system interfaces can increase operators' cognitive workload; and operations from remote locations with distributed teams may impact the effectiveness of teamwork. Therefore, additional research is needed to develop a knowledge base of the various human performance assessment methods needed to better understand their scope, feasibility, and limitations along with workload, situational awareness, and teamwork links to human performance indicators and safety aspects in NPP control rooms. Moreover, an analytic tool or method to assess those measurements in the NPP domain—as opposed to existing techniques that were developed for military, aerospace, and aviation domains—also is needed to assist NRC staff in conducting independent human performance evaluations for V&V purposes.

- *Considerations for Staffing Development and Validation in New and Advanced Reactors.* Staffing has been a protracted concern for NPP operations personnel and regulatory bodies. As staffing is central to any system, implications for the personnel such as the numbers, qualifications, responsibilities, and authorities have implications in the design and operation of an NPP. As part of staffing assessment, regulators responsible for the oversight of safe and secure operations have developed regulations and regulatory guidance to address these concerns. Recent advanced reactor concepts, however, have challenged the current paradigms in concepts of operation and staffing for LWR designs as those that are prescribed in 10 CFR 50, 52, 55, NUREG-0800, and NUREG-0711. Additional research is needed to (1) evaluate regulatory underpinnings for current regulations and guidance related to staffing and staffing plan validation, (2) collate and evaluate cognitive factors that are critical to the staffing evaluation and staffing plan validation with new and postulated advanced reactor designs, (3) identify and bound the critical parameters and indicators that provide a basis for staffing plan licensing reviews, and (4) update existing models and regulatory tools to ensure cohesive system integration considerations. The expected results of this effort are models that can be used to explore staffing from a regulatory perspective and updated tools for licensing staff to perform comprehensive reviews from construction and operating license applicants.

- *Integrated System Validation* (ISV). The principal purpose of ISV is to determine if new control rooms perform within acceptable limits and thereby support safe operation of the plant. Key challenges to successful ISV exist that represent primary knowledge gaps within the field. These knowledge gaps include the nonstandardized terminology and definitions; the amount of testing and evaluation needed to validate a control room design; the preferred process for selecting, classifying, and prioritizing the performance indicators and acceptance criteria; the lack of a standardized approach for ensuring a representative sampling of operators and task conditions and for anchoring performance acceptance criteria to safety criteria; and the methodological problem of determining criteria for validating new control room designs. Research is needed to address these major challenges including the establishment of trustworthy decision criteria for accepting or rejecting design solutions on the basis of human performance measurement.

Evaluation Models/Tools

NRC's *Standard Review Plan* (NUREG-0800) provides high-level guidance for the conduct of HFE reviews in Chapter 18, "Human Factors Engineering." Detailed design review procedures for evaluating the HFE programs of design certifications applicants are provided in *HFE Program Review Model* (NUREG-0711, Revision 2). The NRC staff also evaluates HSIs for conformance with the guidance contained in *Human-System Interface Design Review Guidelines* (NUREG-0700, Revision 2). NUREG-0711 and NUREG-0700 were last updated in 2004 and 2002, respectively, and do not address some of the human performance questions that are raised by new concepts of operations and more fully automated designs as discussed above. Therefore, the primary goal of the proposed research activities is to develop the technical bases for and update these staff review guidance documents.

In addition, just as the technologies available for new and advanced reactor designs are changing, so are the methods and tools used to analyze, design, and evaluate the HFE aspects of those designs. Emerging HFE methods and tools include software for performing traditional HFE analyses, rapid development engineering, and cognitive task analysis methods; use of virtual environments and visualizations; and applications of human performance modeling. NRC staff will be faced with evaluating the applicants' methods and tools used for advanced reactor designs as an important aspect of safety reviews. Another important goal of the human factors aspects of this plan is initiation of the development of a standardized "toolkit" that integrates these new HFE methods and tools with other more familiar methods and tools (e.g., existing measures of workload and situation awareness) and that can be adapted to address the safety issues that are relevant to various advanced reactor designs.

Data Needs

The following types of data are needed to address the expected safety significant human factors issues associated with advanced reactor designs and to be prepared to review multimodular designs.

- *Operating Experience.* Although a number of Generation III and III+ plants have been operating for many years (e.g., advanced boiling-water reactors in Japan, the N4 in France), very little information is generally available pertaining to their operating experience. The same can be said for the many NPPs around the world that have undergone modernization programs using many of the technologies that will be used in advanced reactor designs. This information is very important to the development of future research and as an input to the development of regulatory approaches to the safety review of the impacts of advanced technologies on human performance. The information needed includes:

    - The types of automation implemented and the characteristics of their HSIs
    - Characteristics of HSIs dedicated to disturbance management
    - Characteristics and functionality of CBPs and other computerized operator aids
    - Experience related to performance of tasks such as maintenance, equipment tagout, and testing using computer-based interfaces
    - Impacts on human performance of software upgrades and modifications
    - Operator-modifiable features such as set point adjustment, temporary alarms, and temporary displays
    - Experience with events
    - Identification and treatment of risk-important personnel actions

- *Laboratory and Simulator-Based Research Data.* Many research questions applicable to advanced reactors have been addressed, at least in part, in other settings including military, aerospace, and the oil and pipeline domains. However, some issues are unique to the nuclear industry. Moreover, although applicants will submit the research results needed to support their design decisions, it may be necessary to independently validate the information submitted. To do independent validation, the NRC staff requires access to laboratory and part-task or, in some cases, to full-scope simulation capabilities to validate the metrics to be included in the standardized "tool kit" for evaluating safety significant human factors issues in advanced reactor designs.

Data Sources

Data for the human factors analyses will be obtained from a variety of sources including data in the open literature, international cooperative agreements, vendor data, and NRC-funded programs. The simulation access capability is being addressed through participation in the

Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA)/Halden Reactor Project research, participation in the OECD/NEA/ Committee on the Safety of Nuclear Installations/Working Group on Human and Organizational Factors, collaboration with EPRI, development of contracts with national laboratories and universities, and the planned development of simulation and human performance modelling capabilities at NRC Headquarters. Table 15 identifies the major milestones for human factors analysis tools development.

Table 15. Major Milestones for Human Factors Analysis Tools Development

Activity Title	Activity Begins	Activity Completed
Human performance impacts of degraded I&C and HSIs	4Q07	2Q13
HF aspects in CONOPS of modular designs	1Q10	2Q13
Considerations for staffing development and validation in new and advanced reactors	2Q11	2Q13
Automation and HSI complexity	1Q10	4Q11
Computer-based procedures	1Q10	4Q12
Workload, situational awareness and teamwork	2Q11	2Q13
Integrated system validation	2Q11	2Q13
Publication of updates to NUREGs 0700 and 0711 and other review guidance	4Q09	4Q13

Plan for Probabilistic Risk Assessment (PRA) and Risk-Informed Licensing Infrastructure

Safety-Significant Issues and Phenomena

In comparison with current operating reactors in the United States, the potential HTGR designs have significant technological differences that could lead to different behaviors and accident scenarios. To support the staff's review of HTGR PRA (including HRA) models, research is required to identify and understand potentially risk-significant failure modes (including dependencies between systems and components and effects of human failures) and the consequences of accidents. NRC also has given consideration to revising the existing licensing regulations to include an expanded role in the use of PRA and risk-informed perspectives during the design, design certification, licensing, and operating phases of advanced non-LWRs. This possible expanded role will require additional research to determine the proper scope and level of detail for a technically acceptable risk assessment model.

Evaluation Models/Tools

In addition to assessing the possible development of a risk-informed infrastructure, the following research tasks are planned to support the PRA needs for HTGR licensing activities:

- A planning study will be performed that will identify gaps in guidance and tools needed to support PRA technical acceptability review. This study will identify specific research programs that should be put into place to support advanced reactor licensing activities associated with review and approval of the NGNP PRA model. The planning study will include the following activities: review of available literature to identify unique design and safety issues associated with HTGR designs and highlight any issues that could potentially impact the development of a plant PRA, review of existing PRA standards and guidance to determine applicability to HTGRs, assessment of the use of existing PRA methods and data for HTGR PRA, assessment of the role of PRA during all phases of the HTGR plant life cycle including pre-operational phases, and assessment of needs for research on HRA for HTGRs. This work is being pursued in response to a user need request for research from the user office (NRO). The research proposed by the planning study will be consistent with the needs of the user office and the extent that PRA/HRA will be used in the risk-informed licensing approach.

- A feasibility study for developing a scoping-level PRA model will be performed. A scoping-level PRA is proposed to be developed to support the identification, prioritization, and selection of R&D topics using appropriate risk metrics that support agency policy goals. A major aspect of developing this type of PRA will be documenting the assumptions that are made to ensure development of a complete PRA. The main output from the feasibility study is an assessment of the schedule and resources needed to develop a scoping-level PRA for staff use during the review of the NGNP licensing application. The feasibility study is being pursued in response to a user need request for research from the user office. Additional research to develop a scoping-level PRA model may be pursued as directed by the user office and based on the outcome of the feasibility study.

- After completing the planning study, and in coordination with appropriate NRO staff, a guidance document will be developed to support staff review of an HTGR PRA submitted by the applicant. This guidance will focus on the gaps that arise as a result of the use of PRA with HTGR technology, as well as those due to the use of PRA in the design stage. This guidance will assist the reviewer in assessing the relative importance of the various assumptions in the PRA model, understanding the sensitivity of the PRA model to important assumptions, and identifying and understanding PRA uncertainties.

Also, a baseline probabilistic system analysis tool for NRC use (e.g., Standardized Plant Analysis Risk [SPAR] model) is planned to be developed. Potential uses of this analysis tool include: prioritization of review and inspection activities, and support for the reactor oversight process. The development of this detailed risk model will likely begin during the NGNP licensing review. While preliminary versions of the model may inform review activities, the model is not planned to have a formal role in the licensing review, and the final version of the model may not be available to support licensing activities.

Data Needs

The planning study, discussed above, will determine the specific data that are needed to support the technical acceptability review for an HTGR PRA/HRA model. The data needs may include reliability data for potentially risk-significant plant systems and components as well as plant and human performance data that may influence the technical elements of the PRA (e.g., success criteria, accident sequence analysis).

Data Sources

The planning study will consider the possible data sources that may be used to support an HTGR PRA/HRA model. The data sources may include relevant reliability data from operating LWRs and/or data that can be integrated from the research programs for the other technical disciplines discussed in this plan. The planning study also will consider the appropriate use of an HTGR PRA/HRA model if the proper reliability data is not be available. Table 16 identifies the major milestones for PRA and risk-informed licensing infrastructure.

Table 16. Major Milestones for PRA and Risk Informed Licensing Infrastructure

Activity Title	Activity Begins	Activity Completed
Develop feasibility of scoping-level PRA	3Q09	1Q11
Develop draft regulatory guidance for NGNP licensing strategy	3Q09	1Q10
Advanced reactor PRA research planning study	3Q09	1Q11
Develop guidance for PRA technical acceptability[77]	1Q11	4Q12
Develop detailed technical guidance for HTGR PRA[78]	1Q11	4Q12
Develop tools/methods/data for HTGR PRA technical acceptability review[79]	1Q11	4Q12

[77] Development of regulatory guidance and supporting guidance/tools/methods/data for advanced reactor PRA technical acceptability will be coordinated and directed by the user office.
[78] Same as in footnote 77.
[79] Same as in footnote 77.