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Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)

Volume 1: Main Report

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Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)

Volume 1: Main Report

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ABSTRACT

A phenomena identification and ranking table (PIRT) process was conducted for the Next Generation Nuclear Plant (NGNP) design. This design (in the conceptual stage) is a modular high-temperature gas-cooled reactor (HTGR) that generates both electricity and process heat for hydrogen production. Expert panels identified safety-relevant phenomena, ranked their importance, and assessed the knowledge levels in the areas of accidents and thermal fluids, fission-product transport and dose, high-temperature materials, graphite, and process heat for hydrogen production. This main report summarizes and documents the process and scope of the reviews, noting the major activities and conclusions. The identified phenomena, analyses, rationales, and associated ratings of the phenomena, plus a summary of each panel's findings, are presented. Individual panel reports for these areas are provided as attached volumes to this main report and provide considerably more detail about each panel's deliberations as well as a more complete listing of the phenomena that were evaluated.

FOREWORD

The Energy Policy Act of 2005 (EPAct), Public Law 109-58, mandates the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Department of Energy (DOE) to develop jointly a licensing strategy for the Next Generation Nuclear plant (NGNP), a very high temperature gas-cooled reactor (VHTR) for generating electricity and co-generating hydrogen using the process heat from the reactor. The elements of the NGNP licensing strategy include a description of analytical tools that the NRC will need to develop to verify the NGNP design and its safety performance, and a description of other research and development (R&D) activities that the NRC will need to conduct to review an NGNP license application.

To address the analytical tools and data that will be needed, NRC conducted a Phenomena Identification and Ranking Table (PIRT) exercise in major topical areas of NGNP. The topical areas are: (1) accident analysis and thermal-fluids including neutronics, (2) fission product transport, (3) high temperature materials, (4) graphite, and (5) process heat and hydrogen production. Five panels of national and international experts were convened, one in each of the five areas, to identify and rank safety-relevant phenomena and assess the current knowledge base. The products of the panel deliberations are Phenomena Identification and Ranking Tables (PIRTs) in each of the five areas and the associated documentation (Volumes 2 through 6 of NUREG/CR-6944). The main report (Volume 1 of NUREG/CR-6944) summarizes the important findings in each of the five areas. Previously, a separate PIRT was conducted on TRISO-coated particle fuel for VHTR and high temperature gas-cooled reactor (HTGR) technology and documented in a NUREG report (NUREG/CR-6844, Vols. 1 to 3).

The most significant phenomena (those assigned an importance rank of "high" with the corresponding knowledge level of "low" or "medium") in the thermal-fluids area include primary system heat transport phenomena which impact fuel and component temperatures, reactor physics phenomena which impact peak fuel temperatures in many events, and postulated air ingress accidents that, however unlikely, could lead to major core and core support damage.

The most significant phenomena in the fission products transport area include source term during normal operation which provides initial and boundary conditions for accident source term calculations, transport phenomena during an unmitigated air or water ingress accident, and transport of fission products into the confinement building and the environment.

The most significant phenomena in the graphite area include irradiation effect on material properties, consistency of graphite quality and performance over the service life, and the graphite dust issue which has an impact on the source term.

The most significant phenomena in the high temperature materials area include those relating to high-temperature stability and a component's ability to withstand service conditions, long term thermal aging and environmental degradation, and issues associated with fabrication and heavy-section properties of the reactor pressure vessel.

The most significant phenomenon in the process heat area was identified as the external threat to the nuclear plant due to a release of ground-hugging gases from the hydrogen plant. Additional phenomena of significance are accidental hydrogen releases and impact on the primary system from a blowdown caused by heat exchanger failure.

The PIRT process for the NGNP completes a major step towards assessing NRC's research and development needs necessary to support its licensing activities, and the reports satisfy a major EPAct milestone. The results will be used by the agency to: (1) prioritize NRC's confirmatory research activities to address the safety-significant NGNP issues, (2) inform decisions regarding the development of independent and confirmatory analytical tools for safety analysis, (3) assist in defining test data needs for the validation and verification of analytical tools and codes, and (4) provide insights for the review of vendors' safety analysis and supporting data bases.

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EXECUTIVE SUMMARY

Background

The Next Generation Nuclear Plant (NGNP) is currently in the conceptual design stage. DOE (Office of Nuclear Energy, Science and Technology) candidates funded for NGNP conceptual design development include a modular reactor using a direct-cycle gas turbine with a prismatic block helium cooled core. The candidates also include an indirect cycle prismatic core design and a pebble bed reactor (PBR) version. All of these candidate designs will rely heavily on taking credit for passive phenomena in the safety aspects of the design. The NGNP's primary product is electricity but also includes a process heat loop (utilizing an intermediate heat exchanger) coupled to the reactor for the production of hydrogen.

The Phenomena Identification and Ranking Table (PIRT) process is an effective tool for providing an expert assessment of safety-relevant NGNP phenomena and for assessing NRC's research and development needs. A nine step PIRT process was conducted by five panels of experts for the NGNP in the following topical areas: Accident and Thermal Fluids, Fission-Product Transport and Dose, High-Temperature Materials, Graphite, and Process Heat and Hydrogen Co-Generation Production. Phenomena important to safety systems and components were identified and figures of merit were established. The panels rated (as high, medium, or low) the importance and the associated knowledge level of the phenomena. Panel deliberations and rationale for the ratings were documented. The major panel findings are summarized below. Additional details and documentation can also be found in Volumes 2 through 6 of this report (respectively, for the panels listed above).

Accidents and Thermal Fluids (Including Neutronics) Panel Findings

The panel concentrated on the thermal fluid phenomena but also considered the neutronic phenomena where appropriate. Normal operations, loss-of-forced-cooling (LOFC) events (both pressurized and depressurized), air ingress, reactivity insertion events, and some phenomena associated with the process heat loop and intermediate heat exchanger were evaluated. The most significant phenomena identified by the panel include the following:

- Primary system heat transport phenomena (conduction, convection, and radiation), including the reactor cavity cooling system performance which impact fuel and component temperatures
- Reactor physics phenomena (feedback coefficients, power distribution for normal and shutdown conditions) as well as core thermal and flow aspects. These often relate to the power-to-flow ratio and thus impact peak fuel temperatures in many events; and
- Postulated air ingress accidents that, however unlikely, could lead to major core and core support damage.

Fission Product Transport and Dose Panel Findings

The panel found that at this early stage in the NGNP design, a wide range of transport options needed to be examined. The most significant phenomena identified were:

- Fission product contamination of the graphite moderator and primary circuit (including the turbine) which is not negligible for normal operation and constitutes an available source term
- Transport of fission products into the confinement building and the environment. This is primarily a building leakage (and/or filtering) problem, but depends on the gaseous and suspended aerosol inventory of fission products.

- Behavior of the fission product inventory in the chemical cleanup or fuel handling system during an accident. An overheat event or loss of power may cause release from this system and transport by some pathway into the confinement building or environment.
- Transport phenomena (such as chemical reactions with fuel, graphite oxidation) during an unmitigated air or water ingress accident.
- Quantification of dust in the reactor circuit (from several sources). This may be easily released during a primary boundary breach. The highest dust quantities are expected in the pebble bed core and the lowest in the prismatic core (at least an order of magnitude less).

High Temperature Materials

The major aspects of materials degradation phenomena that may give rise to regulatory safety concern were evaluated for major structural components and their associated materials. These materials phenomena were evaluated with regard to their potential for contributing to fission product release at the site boundary under a variety of event scenarios covering normal operation, anticipated transients, and accidents and the currently available state of knowledge with which to assess them. Key aspects identified by this panel were:

- High-temperature stability and a component's ability to withstand service conditions.
- Issues associated with fabrication and heavy-section properties of the reactor pressure vessel.
- Long-term thermal aging and possible compromise of reactor pressure vessel surface emissivity as well as the reactor cavity coolant system.
- High temperature performance, aging fatigue and environmental degradation of insulation.

Graphite

Much has been learned about the behavior of graphite in nuclear reactor environments since the first graphite reactors went into service. It is expected that the behavior of these graphites will conform to the recognized trends for near-isotropic nuclear graphite. However, the theoretical models still need to be tested against experimental data for the new graphites and extended to higher neutron doses and temperatures typical of Generation IV reactor designs. Significant phenomena noted by the panel were:

- Material properties (creep, strength, toughness, etc.) and the respective changes caused by neutron irradiation.
- Fuel element coolant channel blockage due to graphite failures.
- Consistency in graphite quality (includes replacement graphite over the service life).
- Dust generation and abrasion (especially noteworthy for pebbles, but of concern as well for the prismatic design).

Process Heat and Hydrogen Co-Generation

The panel found that the most significant external threat from the chemical plant to the nuclear plant is from a release of ground-hugging gases. Oxygen was determined to be the most important because (1) it is a significant by-product from all hydrogen production processes that start with water and (2) it may be released continuously as a "waste" if there is no local market. This is due to its combustion aspects, plume behavior, and allowable concentration, and is consistent with the chemical safety aspects and known risks of oxygen plants. Accidental hydrogen releases from the chemical plant were considered a lesser concern in terms of reactor safety because of the high buoyancy of hydrogen and its tendency towards dilution.

The panel was also concerned with the high importance of heat exchanger failures and associated phenomena for blowdown. These can have different types of impacts (such as pressure pulses and thermal consequences) on the primary system.

Conclusions

The NGNP philosophy is different from most currently licensed reactors in that it relies on a robust ceramic-coated fuel particle in a relatively chemically inert environment (helium), immobilization of the small fission product releases during normal operation, and passive heat dissipation to withstand design basis events with minimal fuel damage and source term generation. As such, the NGNP places a burden on the designer to provide validation of key passive safety phenomena (conduction, radiation from the vessel to the RCCS), as well as reliance on the coated-fuel-particle performance and a stable graphite core structure. Additionally, fission product release and transport behavior must be well understood (or at least bounded) if the vented confinement approach is part of the design and credit is to be taken for dose reduction by the intrinsic features of the reactor and associated structures and systems. The PIRT panel findings, taken as a whole, provide a broad perspective of the phenomena. The PIRT process for the NGNP completes a major step towards assessing NRC's research and development needs necessary to support its licensing activities.

ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ACTH	Accident and Thermal Fluids PIRT
ANL	Argonne National Laboratory
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing Metals
ATWS	anticipated transients without scram
AVR	Atomgeneinschaft Versuchs Reaktor
BDBA	beyond design basis accident
BOP	balance of plant
B&PV	Boiler & Pressure Vessel
C–C	carbon–carbon
CEA	Commissariat à l'Énergie Atomique
CFD	computational fluid dynamics
CFP	coated fuel particle
CHE	compact heat exchanger
CTE	coefficient of thermal expansion
D-LOFC	depressurized loss-of-forced circulation
DBA	design basis accident
DIDO	Harwell (U. K.) test reactor
DOE	Department of Energy
EPAct	Energy Policy Act of 2005, Public Law 109-58
FEM	finite-element method
FOM	figure of merit
FP	fission product
FPT	fission-product transport
G-LOFC	general loss-of-forced circulation
GRAPH	Graphite PIRT
GT-MHR	gas-turbine-modular helium reactor
H,M,L	high, medium or low (ratings)
HPTU	high pressure test unit (PBMR—RSA)
HT	high temperature
HTGR	high-temperature gas-cooled reactor
HTMAT	High-Temperature Materials PIRT
HTR-10	high-temperature reactor (10 MW—China)
HTTF	heat transfer test facility (PBMR—RSA)
HTTR	high-temperature engineering test reactor (Japan)
HVAC	heating ventilating air conditioning
HX	heat exchanger

IHX	intermediate heat exchanger
INL	Idaho National Laboratory
IRSN	L'Institut de radioprotection et de sûreté nucléaire
KL	knowledge level
LEU	low-enriched uranium
LOFC	loss-of-forced circulation
LWR	light-water reactor
MIT	Massachusetts Institute of Technology
MS	molten salt
NERAC	Nuclear Energy Research Advisory Committee
NDE	nondestructive examination
NGNP	next generation nuclear plant
NNSA	National Nuclear Security Administration
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PHHP	Process Heat and Hydrogen Co-Generation Production PIRT
P-LOFC	pressurized loss-of-forced circulation
PBMR	pebble bed modular reactor (RSA)
PBR	pebble bed reactor
PHX	process heat exchanger
PIRT	phenomena identification and ranking table
PMR	prismatic-core modular reactor
PWHT	postweld heat treatment
QA	quality assurance
QC	quality control
R&D	research and development
RCCS	reactor cavity cooling system
RPV	reactor pressure vessel
RSA	Republic of South Africa
SiC	silicon carbide
SG	steam generator
SNL	Sandia National Laboratory
SRNL	Savannah River National Laboratory
SSCs	structures, systems, and components
TAMU	Texas A&M University
THTR	thorium high-temperature reactor
VHTR	very high temperature gas-cooled reactor

1. INTRODUCTION

1.1 Background

The Next Generation Nuclear Plant (NGNP) is a very high temperature gas-cooled reactor (VHTR) for generating electricity and co-generating hydrogen using the process heat from the reactor. The Energy Policy Act of 2005, Public Law 109-58 (EPAct) mandates the U.S. Department of Energy (DOE) to develop an NGNP prototype for operation by 2021 and provides the U.S. Nuclear Regulatory Commission (NRC) with licensing authority, in accordance with Section 202 of the Energy Reorganization Act of 1974. The EPAct also mandates DOE and NRC to develop jointly a licensing strategy for NGNP and submit a report to the U.S. Congress by August 8, 2008, describing the strategy.

The elements of the NGNP licensing strategy include a description of the analytical tools that the NRC will need to develop to verify independently the NGNP design and its safety performance and a description of other research and development (R&D) activities that the NRC will need to conduct to review an NGNP license application. The Phenomena Identification and Ranking Table (PIRT) is an effective tool for providing an expert assessment of safety-relevant NGNP phenomena and for assessing NRC's R&D needs for NGNP licensing.

1.2 The Phenomena Identification and Ranking Table (PIRT)

NRC, in collaboration with DOE, conducted multiple PIRT exercises using panels of technical experts covering five major topical areas relevant to NGNP safety and licensing: (1) accident and thermal fluids analysis (including neutronics); (2) fission-product transport and dose; (3) high-temperature materials; (4) graphite; and (5) process heat for hydrogen co-generation. The formal PIRT process, as applied to the NGNP, is described in Sect. 3.

The PIRT is a structured expert elicitation process designed to support decision making. The process consists of nine distinct steps as follows:

- **Step 1**—define the issue that is driving the need for a PIRT;
- **Step 2**—define the specific objectives for the PIRT;
- **Step 3**—define the hardware and the scenario for the PIRT;
- **Step 4**—define the evaluation criterion;
- **Step 5**—identify, compile, and review the current knowledge base;
- **Step 6**—identify plausible phenomena, that is, PIRT elements;
- **Step 7**—develop importance ranking for phenomena;
- **Step 8**—assess knowledge level (KL) for phenomena; and
- **Step 9**—document PIRT results.

1.3 Report Organization

Detailed documentation of each panel's deliberations and combined results are captured in supplemental Volumes 2 through 6 of this report. Volume 2 contains a full account of the Accident and Thermal Fluids (ACTH) PIRT and serves as the technical basis for summary information provided in this report. Volumes 3 through 6 contain, respectively, the Fission-Product Transport and Dose (FPT) PIRT, the High-Temperature Materials (HTMAT) PIRT, the Graphite (GRAPH) PIRT, and the Process Heat and Hydrogen Co-Generation Production (PHHP) PIRT. Each of these volumes is a stand-alone report prepared by the respective panels. Extensive bibliographies may be found in the supplemental volumes. The reader should also note that acronyms are used throughout this main report to refer to each specific topical panel's activities or areas.

The structured PIRT process produced a large body of materials in each of the topical areas. The individual panel analyses and deliberations are documented in this summary report and its supporting volumes. The summary report is organized into four sections. Section 1 provides background information. Section 2 provides a general description of the NGNP design concept and a brief description of high-temperature gas-cooled reactor (HTGR) technology. Section 3 provides an overview of the PIRT process, the objectives and scope of the topical areas, and a list of the PIRT panel members. Section 4 presents an analysis and summary of the major findings from each area and a brief discussion of the rationale for phenomenon importance and KL rankings. Section 5 enumerates and compares evaluations of important phenomena that were considered by more than one panel. Section 6 presents a summary and conclusions.

2. NEXT GENERATION NUCLEAR PLANT BASIC TECHNOLOGY DESCRIPTION

2.1 NGNP General Features

The NGNP reactor design features are based on the modular HTGR concept for Generation IV reactors. The modular HTGR is designed to meet fundamental safety objectives and requirements, as well as design requirements. The typical HTGR design features include the following:

- high-performance coated fuel particles (CFPs) with the capability of containing radioactive fission products for the full range of operating and postulated accident conditions, with a very low fuel failure fraction and subsequent release of fission products. The CFPs are embedded in either a rod compact inserted into a stacked prismatic block or a spherical compact that constitutes a pebble;
- an inert single-phase high-pressure coolant (helium);
- a graphite-moderated core with the characteristics of low-power density, large heat capacity, high effective core thermal conductivity, and large thermal margins to fuel failure;
- negative fuel and moderator temperature coefficients of reactivity sufficient to shut down, in conjunction with the negative reactivity feedback of the fission product xenon-135, the reactor in loss-of-forced circulation (LOFC) events. This aspect provides for stabilizing power-control feedback, for most reactivity insertion events (for both startup and power operation) for the entire fuel life cycle and for all applicable temperature ranges;
- a design basis accident decay heat removal system, typically a passive system utilizing natural-convection-driven processes (the Reactor Cavity Cooling System—RCCS); and
- a confinement-style reactor building structure (accommodates depressurizations dynamically and may be used instead of a leak tight sealed containment). The NGNP core design will be either prismatic or pebble bed. The balance of plant (BOP) will consist of an electrical power generation unit (most likely a gas turbine) and a high-temperature process heat component for production of hydrogen. The design power level will be between 400 and 600 MW(t), with approximately 10% of the total thermal power production applied to the hydrogen plant. Coupling of the reactor to the hydrogen plant will be via an intermediate heat exchanger (IHX) and a long heat transport loop, with various options for the transport fluid currently under consideration. Figure 1 shows a sketch of the NGNP concept highlighting the reactor, power conversion, and the hydrogen production units. Figures 2 and 3 show examples of the two types of NGNP reactor cores, prismatic, and pebble bed, respectively.

2.2 Description of NGNP Hardware

The NGNP is currently in the conceptual design stage, and DOE's selection of the design of both the reactor and process heat sectors is in progress. Reactor candidates funded for NGNP conceptual design development by the DOE Office of Nuclear Energy, Science and Technology include a prismatic modular reactor (PMR), which uses a direct-cycle prismatic block gas turbine HTGR [namely, the gas-turbine-modular helium reactor (GT-MHR) design by General Atomics and similar in configuration to that being co-funded by DOE/National Nuclear Security Administration (NNSA) and by Rosatom (Russia) for plutonium burning in Russia]. In addition, an indirect cycle prismatic core design by AREVA and a pebble bed reactor (PBR) version being developed by a consortium of Westinghouse and pebble-bed modular reactor (PBMR) Pty similar to the South African PBMR under development are also NGNP reactor candidates.

Prismatic fuel elements consist of fuel compacts inserted into holes drilled in graphite hexagonal prism blocks ~300 mm across the flats and 800 mm long (very similar to the Fort St. Vrain reactor fuel elements), interspersed with coolant holes. Pebble fuel elements are 6-cm-diam spheres containing a central region of TRISO fuel particles in a graphitized matrix material, surrounded by a 5-mm protective outer coating of graphitic material. The pebble bed concept was developed initially in the United States in the 1950s and later further developed in collaboration between Germany and the United States in the 1960s. The pebble bed concepts employ continuous refueling, with pebbles typically recycled ~6 to 10 times, depending on measured burnup.

A major component in the NGNP, the IHX, is required for coupling the primary high-temperature, high-pressure helium system to either the indirect gas-turbine system and/or the process heat component and must be designed to operate at very high temperatures. There is the potential for large pressure differences between IHX primary and secondary sides—at least in transients and perhaps for long-term operation.

There are multiple methods to produce hydrogen using heat, heat and electricity, and electricity-only using nuclear energy. Candidate processes include steam reforming of natural gas, electrolysis, high-temperature electrolysis, and hybrid-sulfur or sulfur-iodine chemical extraction. There are also multiple markets for high-temperature nuclear process heat and hydrogen which can have a strong influence on the safety challenges associated with co-locating a nuclear plant and hydrogen plant. Several different types of chemical plants might be coupled to the NGNP reactor over its lifetime to meet different needs. This selection will depend on the currently identified potential applications for nuclear process heat and hydrogen production, with a consideration to demonstrate the reactor's safety features in tandem with various process heat configurations.

Several confinement and containment options have been investigated in the past, with the vented confinement option generally selected as a baseline (with or without filters). Any early fission product release in a depressurization accident is usually assumed to be small, requiring no holdup, while any delayed releases are assumed to be larger, but modest, with very little pressure difference to drive fission products out into the environment.

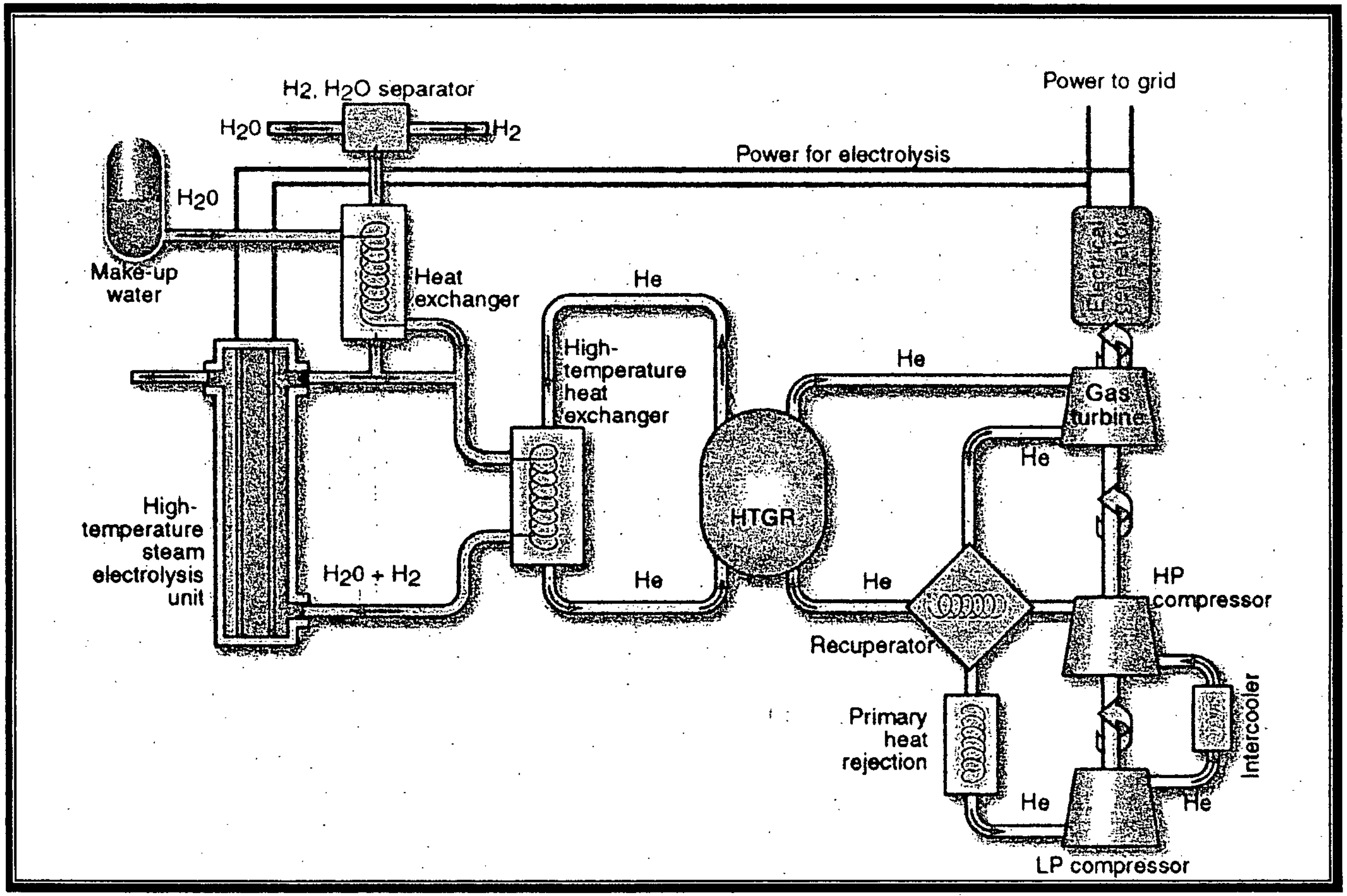


Fig. 1. Representative schematic of the NGNP.

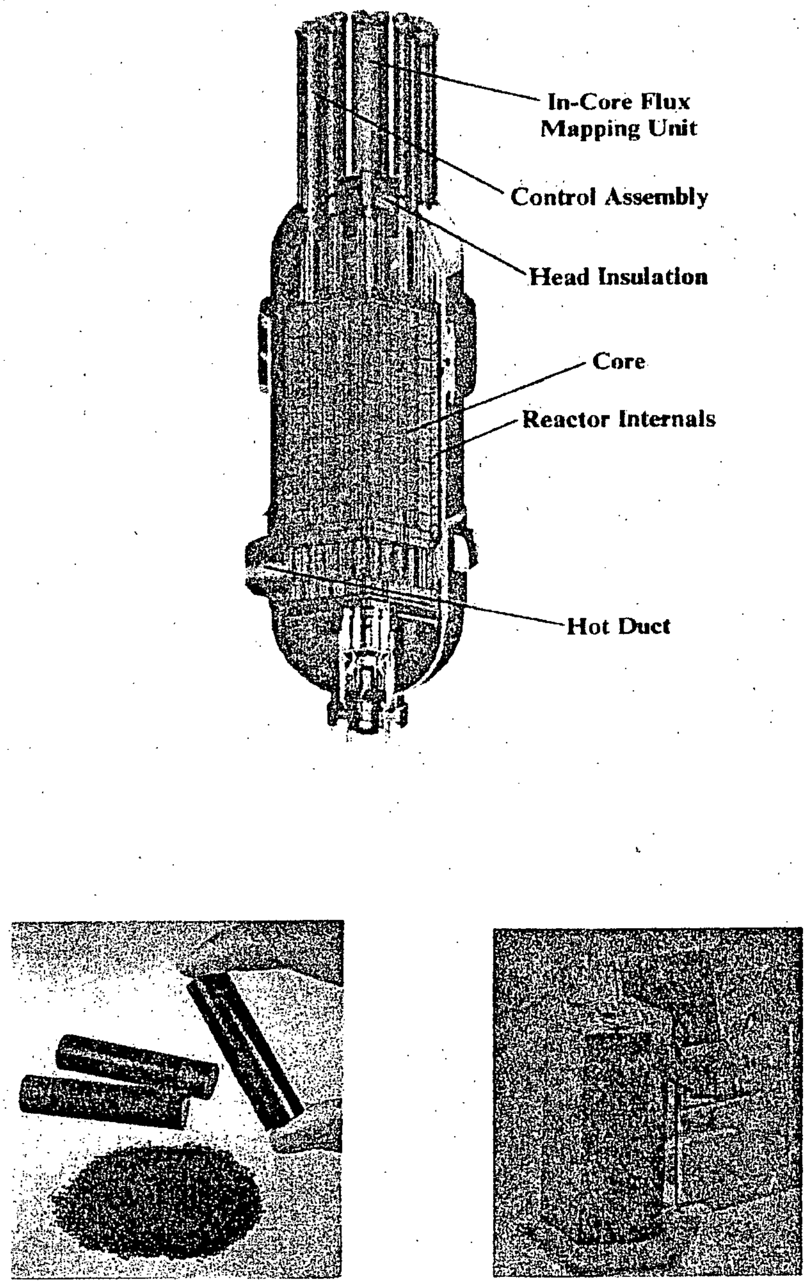
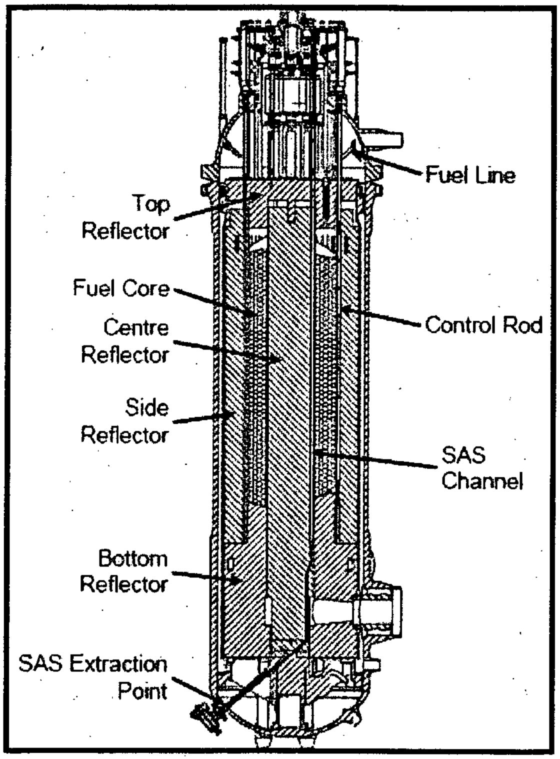
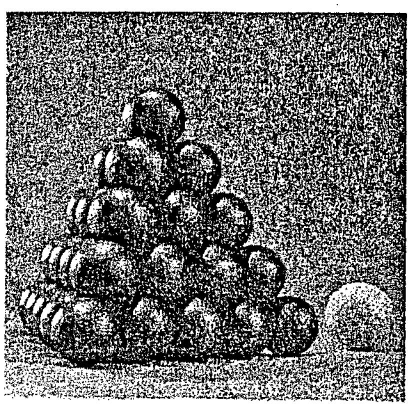


Fig. 2. NGNP prismatic core option.





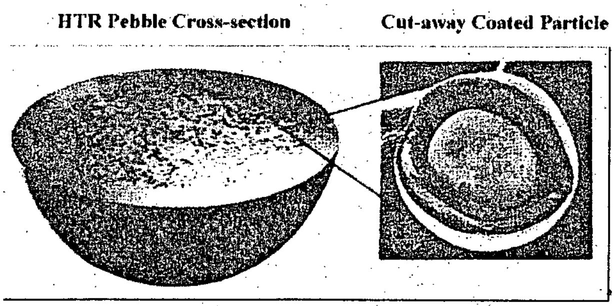


Fig. 3. NGNP pebble bed core option.

3. PIRT OBJECTIVES, PROCESS, SCOPE, SCENARIOS, AND PANEL COMPOSITION

3.1 Objectives

The overall objective is to identify safety-relevant phenomena associated with the NGNP (during normal operations, transients, and postulated accidents). The five panels utilized the nine step process described in Sect. 3.2 to meet this overall objective. A determination of the relative importance of these phenomena to the expected consequences [figures of merit (FOMs)], and an assessment of the knowledge level were performed for the five topical areas. As a result, the NRC will have an assessment of the phenomena important to the overall process of determining the R&D needs for NGNP licensing.

3.2 PIRT Process Description as Applied to the NGNP

As stated in Sect. 1, the PIRT process consists of nine steps. These steps are described below.

Step 1—Issue Definition

This step defines the issue that drives the needs for development of technical bases and analytical tools to perform safety analyses and regulatory reviews and other research and development needs to support NGNP licensing.

Step 2—PIRT Objectives

The panel-specific objectives of the NGNP PIRT were defined in this step and these are described below.

For the ACTH PIRT, the panel objectives were to identify safety-relevant phenomena for normal plant operation and postulated accident scenarios, ranking the importance of these phenomena relative to established evaluation criteria or FOMs, and assessing the existing knowledge base for its adequacy to investigate the safety significance of these phenomena.

For the FPT PIRT, the panel objectives were to identify and categorize potential sources and phenomena associated with fission product release for a few major scenarios, ranking these phenomena according to FOMs, and assessing the knowledge base.

The panel objectives for the HTMAT and GRAPH PIRTs were to (1) identify and rank potential degradation mechanisms for the HTGR materials under normal operating, transient, and accident conditions; (2) identify important parameters and dependencies that affect the degradation processes; (3) assess material performance requirements to ensure safety, including needs for additional codes and standards; and (4) assess material properties databases and identify new data needs, where appropriate.

The panel objectives for the PHHP PIRT were to focus on phenomena involved with coupling a hydrogen generation plant to the reactor, mainly those that could affect the reactor, not the hydrogen plant safety. The panel assessed the applicability of existing models and databases to safety analyses of coupled systems within the NGNP technology envelope.

Step 3—Hardware and Scenarios

This step involves identification of the hardware, equipment, and scenarios pertaining to the PIRT exercises. Generally, a specific hardware configuration and a specific scenario are considered before proceeding with the next step. The hardware may be divided into systems (primary system) and components (such as the reactor pressure vessel, graphite blocks, etc.). The scenario may be divided into phases, or in the case of the ACTH, a modular approach to scenarios can be taken. Further discussion on hardware and scenarios are provided later in Sect. 3.4

Step 4—Evaluation Criteria

This step specifies evaluation criteria or FOM for judging the relative importance of safety-relevant phenomena. The key evaluation criterion (FOM at the highest level) is dose to the public from fission product release, and it is common to all major topical areas. Subsidiary evaluation criteria or FOMs differ somewhat in the different topical areas. One of the ground rules for the PIRTs was for each panel to develop its own set of FOMs best suited to their topical area. These criteria are defined later in Sects. 3.5 through 3.9 in the context of individual PIRT topics.

Step 5—Current Knowledge Base

This step involves familiarization with the current knowledge base on HTGR technology, with particular focus on safety-relevant physical phenomena and/or processes associated with hardware and scenarios identified in Step 3 above.

Step 6—Phenomena Identification

This step involves identification of all plausible safety-relevant phenomena for hardware and scenarios identified in Step 3. This is accomplished by panels of experts in the respective topical areas, with individual panel members identifying relevant phenomena first, followed by the deliberations on the collection of phenomena identified. The objective is to develop a preliminary but comprehensive list of phenomena which, in the collective opinion of the panel, is relevant to safety. In developing the list, the panels considered in their deliberation a phenomenological hierarchy starting at the system level and proceeding through component and subcomponent levels, and so on. The panels' objective was to ensure that the lowest level of hierarchical decomposition be consistent with the data and modeling needs from a regulatory perspective.

Step 7—Importance Ranking

In this step, identified phenomena are ranked for their importance relative to the evaluation criteria adopted in Step 4. The rationale for the importance ranking is also provided. The process consists of individual and independent ranking by panel members, discussion of individual rankings considering the rationale, and collective ranking based on the discussion. A ranking breakdown of High, Medium, and Low (H, M, and L) proved to be sufficient in past PIRT exercises and was adopted for the present exercise.

Step 8—Knowledge Level

The level of knowledge regarding each phenomenon is assessed in this step by the panels. The process consists of individual and independent assessment, including the rationale and collective assessment based on the discussion. A qualitative ranking, that is, Known (adequate knowledge), Partially Known (incomplete knowledge), and Unknown (no or hardly any knowledge), or alternatively H, M, or L, was used in past exercises and was adopted for the present exercise.

Step 9—Documentation

The objective of this step is to provide sufficient coverage and depth in the documentation so that a knowledgeable reader can understand what was done (process) and the outcomes (results). The documentation includes PIRT objectives, tables of identified phenomena, importance and knowledge level ranking, and supporting text describing the process of phenomena identification and rationale of the ranking process.

3.3 Panel Composition

Five separate PIRT panels were convened to deliberate on five major topical areas mentioned previously. Members in any given panel were selected from a mix of researchers and subject matter experts in academia, national laboratories, and international organizations. The ACTH panel was comprised of 11 members covering three subtopics—accident analysis, thermal fluids, and neutronics. Two additional members participated in the ACTH panel's evaluations of reactor physics related phenomena. The FPT panel was comprised of five experts; the HTMAT and PHHP panels, four experts each; and the GRAPH panel, three experts. The five panel rosters and industry support participants are shown in Tables 1 through 6.

Table 1. Accident analysis and thermal fluids panel (ACTH)

Name	Affiliation	Relevant skill and expertise
S. Ball (Chair)	Oak Ridge National Laboratory (ORNL)	Graphite reactor severe accident analysis, international expert on HTGR technology and knowledge management
M. Corradini	U. Wisconsin	Reactor safety, thermal fluids, severe accident analysis, chair—DOE National Energy Research Advisory Committee (NERAC) review of NGNP
S. Fisher	ORNL	Reactor safety analysis, HTGR utility experience
R. Gauntt	Sandia National Laboratory (SNL)	Severe accident analysis, code development and assessment
G. Geffraye	Commissariat à l'Énergie Atomique (CEA)	Gas reactor thermal fluids, accident analysis, code development and assessment
J. Gehin	ORNL	Reactor physics
Y. Hassan	Texas A&M University (TAMU)	Thermal hydraulics, computational fluid dynamics (CFD) analysis, gas reactor R&D
D. Moses	ORNL	HTGR neutronics and reactivity feedback, gas reactor operational experience analysis
R. Schultz	Idaho National Laboratory (INL)	Accident analysis, thermal fluids, and gas reactor R&D
J.-P. Renier	ORNL	Reactor physics
T. Wei	Argonne National Laboratory (ANL)	Accident analysis, thermal fluids, and gas reactor R&D

Table 2. Fission-product transport panel (FPT)

Name	Affiliation	Relevant skill and expertise
M. Kissane	L'Institut de Radioprotection et de Sûreté Nucléaire (IRSN)	Fission-products transport research, gas reactor technology and safety analysis
R. Morris (Chair)	ORNL	Leading researcher in fission-products transport, lead member of TRISO fuel PIRT panel
D. Petti	INL	NGNP R&D Director at INL, HTGR technology development and associated R&D, fission-products research
D. Powers	SNL	International expert in fission products research, reactor safety analysis, and severe accidents, member—Advisory Committee on Reactor Safeguards (ACRS_
R. Wichner	Consultant	Leading researcher in fission products transport, member of TRISO fuel PIRT panel

Table 3. High-temperature materials panel (HTMAT)

Name	Affiliation	Relevant skill and expertise
R. Ballinger	Massachusetts Institute of Technology (MIT)	International expert in materials research including high-temperature materials for reactor applications
W. Corwin (Chair)	ORNL	National Director, Generation IV Reactor Materials Technology Program and leading researcher in high-temperature materials research including experiments and analysis and reactor safety applications
S. Majumdar	ANL	High-temperature materials research for reactor applications, mechanical properties of materials under accident conditions
K. Weaver	INL	NGNP Deputy Technical Director and researcher for nuclear engineering of advanced systems

Table 4. Graphite panel (GRAPH)

Name	Affiliation	Relevant skill and expertise
R. Bratton	INL	Researcher in graphite technology R&D; subject expert
T. Burchell (Chair)	ORNL	International expert on graphite technology R&D including experimental studies, analysis, standards development, etc.
B. Marsden	U. Manchester	International expert in graphite technology R&D including experimental studies, analysis, reactor applications, standards development, etc.

Table 5. Process heat and hydrogen co-generation panel (PHHP)

Name	Affiliation	Relevant skill and expertise
C. Forsberg (Chair)	ORNL	All-around expertise in reactor technology including BOP, process heat applications, and co-generation technology
M. Gorensek	Savannah River National Laboratory (SRNL)	Subject matter expert in process engineering and thermochemical hydrogen flowsheets
S. Herring	INL	Subject matter expert in hydrogen and high-temperature electrolysis
P. Pickard	SNL	Subject matter expert in process heat applications, hydrogen co-generation, and safety technology

C. Davis from INL participated in the first meeting of the ACTH PIRT representing R. Schultz. Likewise, M. Feltus from DOE represented D. Petti of INL at the first meeting of the FPT PIRT. J. Gehin and J.-P. Renier, both of ORNL, participated in the last meeting of the ACTH PIRT, contributing to deliberations on neutronics issues.

Besides the experts in the PIRT panels, subject matter experts from industry participated in various panels and provided additional resources pertaining to HTGR design concepts, industry R&D activities, and other related subjects. These experts, however, did not deliberate on the phenomena importance ranking exercise. The table below lists these additional experts and indicates the panels in which they participated.

Table 6. Industry experts providing additional resources

Name	Affiliation	PIRT panels
G. Brinkmann	AREVA	ACTH, FPT, GRAPH
C. Kling	Westinghouse	ACTH
M. Mitchell	PBMR Pty	GRAPH, HTMAT
L. Parme	General Atomics	ACTH, FPT, HTMAT
S. Penfield	Technology Insights	GRAPH, HTMAT, PHHP
P. Robinson	PBMR Pty	ACTH, FPT
F. Sharokhi	AREVA	ACTH, HTMAT, PHHP
W. Windes	INL	GRAPH, HTMAT

3.4 Scope of Panel Reviews

3.4.1 Major hardware and associated phenomena examined

Major NGNP systems and components were considered by the panels. The panels had to deal with the problem that the NGNP is in the high-level conceptual design stage, leaving many features undefined.

The panels, therefore, focused on the major systems for each specialty, analyzing phenomena related to hardware and systems at a top level.

The ACTH panel considered many phenomena associated with reactor systems, such as passive cooling of the reactor core via conduction and radiation, and cooling of the reactor pressure vessel (RPV) via radiation and convection utilizing the RCCS for all LOFC events. The panel had to consider both prismatic block and pebble concepts (and thus the aspects associated with continuous refueling when examining the phenomena). Material properties associated with hardware (such as effective core conductivity and reactor vessel emissivity) are key phenomena considered by the ACTH panel.

A variety of specific components and associated materials phenomena constituted major topics for the HTMAT panel. This panel focused on hardware such as the RPV, insulation, vessel supports, etc. The panel organized its assessments by components and included:

- nonmetallic and/or metallic materials for control rods;
- nonmetallic materials for other reactor internals and primary circuit components;
- metallic alloys for very high-temperature service for heat exchangers (HX) and turbomachinery;
- metallic alloys for high-temperature service for the RPV and vessel supports, as well as for other pressure vessels and components in the primary circuit;
- metallic alloys for secondary heat transfer circuits and BOP;
- materials for valves, bearings, and seals; and
- nonmetallic insulation materials.

The hardware of interest for the graphite panel was straightforward, encompassing fuel, core support, and reflector graphites with consideration given to the qualification of graphites for service temperatures and irradiations. The principal materials and structures covered include core graphite and both replaceable and permanent components.

The FPT panel focused on TRISO fuel particle performance, release and plate-out phenomena, and mitigation options. Thus, behavior of the actual kernel and coatings was of principle concern. The panel was concerned with phenomena of adsorbed fission products, transport, and plate-out of dust on the surfaces (reactor system, cavity, and confinement) that the source term may encounter.

The FPT, GRAPH, and HTMAT PIRT panels also analyzed phenomena related to both design and performance aspects of normal operations and accident situations. Many of the phenomena were chosen based on their impacts on source term generation, source term migration, and maintenance of fission product barriers.

The PHHP panel analyzed phenomena primarily that would broadly affect nuclear plant safety-related structures, systems and components (e.g., an external event from the process heat side impacting reactor plant equipment). This panel also considered aspects associated with failure of the intermediate heat exchanger. With regard to the possible working fluids for the IHX, both helium and molten salt (MS) applications were considered by the PHHP. However, the HTMAT panel did not evaluate any phenomena associated with molten salt because that panel concluded it was outside the envelope of likely configurations for the NGNP.

The identification of hardware components and reactor systems principally associated with the significant phenomena identified by each PIRT panel is covered in the phenomena tables in Sect. 4.

3.4.2 Accidents and thermal fluids

The scope of the ACTH PIRT addressed the need to identify phenomena associated with design and technology development areas that either influence safety or otherwise have relevance to regulatory requirements. The scope included both normal operations and a spectrum of accidents covering various cool-down events, reactivity events, and other scenarios related to aspects of a process heat loop as described in Sect. 3.5. The issues addressed by this PIRT are the importance of these phenomena to a FOM, and how well these phenomena can be characterized by existing data and analytical techniques.

3.4.3 Fission-product transport

The scope of the FPT PIRT included identification of the safety-relevant phenomena associated with the transport of fission products in an accident scenario such as a depressurization of the primary system. The phenomena were ranked in a way that can be used to help guide regulatory requirements and assessments. The FPT is often linked to ACTH areas, and some similar phenomena were assessed by both panels. The panel's scope included identification and ranking of the important FPT phenomena and assessment of the knowledge base, as well as the ability to model fission product behavior and transport from the fuel through the possible release paths.

3.4.4 High-temperature materials

The scope of the materials phenomena covered conventional material properties such as strength, creep, and fatigue as well as the associated aging in a potential 60-year lifetime for some of the plant components. The service conditions considered covered a range that included both chemical attack and thermal cycling; they also encompassed irradiated material properties for metallic and nonmetallic components in or near the core and the primary system. The maintenance of adequate safety margins over time was a major concern for these PIRTs.

3.4.4 Graphite

The scope of the materials phenomena covered conventional material properties such as strength, creep, stress, and fatigue as well as the associated aging in a potential 60-year lifetime for some of the plant components. The scope also included oxidation and the aspects of helium gas impurities and effects of gamma and neutron irradiation.

3.4.5 Hydrogen production and process heat

The scope of this PIRT was to identify potential safety concerns for the production and transport of high-temperature process heat (and electricity) for an adjacent hydrogen-production chemical plant. Because high-temperature heat can only be transported limited distances, the two plants will be in fairly close proximity. The scope did not include an assessment of the industrial chemical plant safety challenges. Rather, the scope covered releases of hydrogen and heavy gases and their potential impacts on the reactor. In addition, phenomena associated with the transport of high-temperature heat to the chemical plant are assessed.

3.5 Accident Scenario Selection

Postulated accident scenario and phenomena considerations were based in part on the ACTH panel's previous experience with HTGR plant operation and accident analysis. Prior studies and interactions with members from different PIRT panels helped to guide the ACTH panel's evaluations.

Normal operation is important (since a potential accident source term lies plated out in the primary system) in that it is the starting point after which the postulated accidents take place. "Normal Operation" was covered in the PIRT process because of its importance in providing initial and boundary conditions for postulated accidents. Consideration of normal operation was also important particularly for HTMAT and GRAPH PIRTs since these two PIRTs dealt with design and operational issues as well. Various PIRT panels recognized that one area of concern in normal operation is the possibility that maximum operating fuel temperatures may be significantly higher than expected, leading to fuel degradation that could cause premature failures when challenged in an accident.

For off-normal and accident situations, the following categorizations of three major event-frequency-based regimes (with typical ranges assigned to the frequency of occurrence) were used.

- *Anticipated Operational Occurrence (AOO)*: An AOO is a frequent event with an expected mean frequency of occurrence of 10^{-2} per plant-year or higher.
- *Design Basis Accident (DBA)*: A DBA is an infrequent event that might occur once during the collective lifetimes of a large number of plants. However, the plant is specifically designed to mitigate the event using only equipment classified as safety grade. DBAs are typically associated with events having a mean frequency between 10^{-2} and 10^{-4} per plant-year.
- *Beyond Design Basis Accident (BDBA)*: A BDBA is a very low-probability event not expected to occur within the collective lifetimes of a large number of similar plants. However, the plant design would mitigate the consequences, taking credit for the available safety-related equipment, operator actions, any existing nonsafety-related equipment, and accounting for long time periods available for corrective actions. BDBAs typically have a mean frequency between 10^{-4} and 5×10^{-7} per plant-year.

PIRT evaluations on specific accident scenarios were done using a matrix–building block format that allowed consideration of all the important phenomena or processes, minimizing repetition. Consideration of a wide range of postulated accidents was based in part on extensive review of operating experience, as well as on detailed and extensive accident analysis and licensing exercises for designs similar to NGNP (but without the process heat component). The scenarios selected for consideration by the ACTH PIRT were as follows:

1. the pressurized loss-of-forced circulation (P-LOFC) accident;
2. the depressurized loss-of-forced circulation (D-LOFC) accident;
3. the D-LOFC followed by air ingress;
4. reactivity-induced transients, including events involving anticipated transients without scram (ATWS);
5. steam-water ingress events; and
6. events related to coupling the reactor to the process heat plant.

3.5.1 The P-LOFC accident

The reference case P-LOFC assumes a flow coast-down and scram with the RCCS operating continuously. The natural circulation of the pressurized helium coolant within the core makes core temperatures more uniform, lowering the peak temperatures more than in a depressurized core, where buoyancy forces do not establish significant helium coolant recirculation flows. The chimney effect in P-LOFC events increases the core (and vessel) temperatures near the top. In P-LOFCs, the peak fuel temperature is not a concern, as it falls well within nominal temperature limits; the major concern is more likely to be the upper vessel and associated component temperatures.

3.5.2 The D-LOFC accident

The D-LOFC reference case assumes a rapid depressurization of the primary coolant and scram, with the passive RCCS operational, and without air ingress. This event for a PMR is known as a "conduction heat-up" (or "cool-down") accident since the core effective thermal conductivity is the dominant mechanism for the transfer of afterheat from the fuel to the reactor vessel. For the PBR, radiation heat transfer is dominant in the core during the heat-up. Typically the maximum expected fuel temperature would peak slightly below the limiting value for the fuel (by design), and the peak would typically occur ~2 days into the accident.

There are two primary parameter variations of interest for this accident, which is generally considered to be the defining accident for determining DBA peak fuel temperatures. The first is effective core graphite thermal conductivity (a function of irradiation history, temperature, orientation, and annealing) for the prismatic design and the effective pebble core thermal conductivity for the pebble-bed design. The second parameter is the decay-heat power distribution vs time after shutdown.

3.5.3 Air ingress following a D-LOFC accident

A more extreme case of the D-LOFC accident involves a significant and continuous inflow of air to the core following depressurization. The significant phenomena noted by the panels for these events are the following:

1. graphite structure oxidation to the extent that the integrity of the core and its support is compromised;
2. oxidation of the graphite fuel elements that leads to exposure of the TRISO particles to oxygen, with a potential for subsequent fission product release; and
3. release of fission products previously absorbed in the graphite structures.

The concern is about configurations and conditions that would support sustained (and large) flows of ingress gas and the long-term availability of oxygen in the gas. The characterization of air ingress accidents is made particularly difficult by the extremely large and diverse set of possible scenarios.

3.5.4 Reactivity events, including ATWS accidents

The most common postulated reactivity events assume a LOFC (either P- or D-) accompanied by a long-term failure to scram. These are extremely low-probability events, where the core heat-up transients are unaffected by a scram (or not) until recriticality occurs upon the decay of the xenon poisoning (typically in ~2 days). For this event to occur, a long-term failure of operation of two independent (safety-grade) scram systems plus a failure of the nonsafety control rods must be assumed.

Other potential reactivity events include the compaction of the pebble bed core during a prolonged earthquake (which can cause a significant reactivity increase), the potential for a positive reactivity insertion from a steam-water ingress event, and a "cold-slug" induced by a sudden decrease in core inlet coolant temperature.

3.5.5 Steam-water ingress events

The panel decided to eliminate this accident type from the current ranking process (see Sect. 4.1.3.8)

3.5.6 Other events: auxiliary and process heat plant-related accidents

The consideration of other events was influenced by difficulties in postulating any accidents relating to yet-to-be defined pertinent plant design features. As an example consideration for coupling to a process heat (hydrogen) plant, a scenario was postulated (by the ACTH panel) for an IHX failure involving a molten-salt heat transport loop coupling the reactor and the hydrogen plant. The process heat PIRT activity also encompassed a variety of scenarios based on the possible external event phenomena (chemical releases) emanating from the process heat plant. The process heat panel investigated scenarios associated with hydrogen, oxygen, and other gas releases with respect to the impact on the reactor.

Specific accident scenarios associated with maintenance and refueling modes, spent fuel storage and handling were not considered by the panels. The FPT panel did note that releases could come from cleanup and holdup systems. However, it was noted that such systems are only vaguely defined at this time. As the NGNP system design matures, these aspects can be considered.

4. PIRT PANEL ANALYSES

Each of the sections below covers the identification and ranking rationale associated with the FOMs developed by the respective panels. The analyses of some of the significant phenomena (high importance and low or medium knowledge base) that were identified by each PIRT panel are presented.* The panel findings are then summarized.

4.1 Accident and Thermal Fluids (ACTH) PIRT Panel

4.1.1 Phenomena identification and description

Phenomena identification in postulated accident sequences involved determination of factors important to the outcomes of the events. For modular HTGRs, which rely largely on inherent (passive) safety features, the important phenomena include physical characteristics (such as material thermal conductivity, radiation heat transfer aspects such as emissivity, temperature-reactivity feedback coefficients, etc.) rather than on the actuation of mechanical or electrical components to halt an accident progression. These phenomena involve combinations of several forms of heat transfer in various geometric configurations. Effective pebble-bed core thermal conductivity, for example, involves (primarily) radiation heat transfer, in addition to conduction, which is a function of pebble irradiation. A qualitative judgment of a phenomenon's importance is not always straightforward since for some specific scenarios it may be crucial to an outcome, while in other scenarios it may not be a factor.

4.1.2 Ranking rationale

Importance evaluations involve judgments of how certain phenomena would impact consequences (per FOM) during an accident. The PIRT panel concentrated on the thermal fluid aspects of the events but also considered neutronic behavior where appropriate. Each phenomenon's assessment and importance ranking was made relative to its importance to the FOMs established by the panel.

The four general FOMs selected by the ACTH were as follows:

- *Level 1*: dose at the site boundary due to radioactivity releases;
- *Level 2*: releases of radioactivity that impact worker dose;
- *Level 3*: fuel failures or conditions (e.g., high temperature) with the potential to impact fuel failure; and
- *Level 4*: includes the following:
    - fraction of the fuel above critical temperatures for extended time periods;
    - RPV, supports, core barrel, or other crucial in-vessel component service conditions;
    - reactor cavity concrete time at temperature; and
    - circulating (primary system) coolant radioactivity (including dust).

The panel members' evaluations of phenomena importance ranking and knowledge level were occasionally subject to different interpretations. For example, some phenomena were important for one postulated accident but inconsequential for another. Likewise for the KL, one view was that the KL should be based on a judgment of how much is known about the phenomenon independent of its

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*To have a comprehensive view of "significant" phenomena, the reader is encouraged to examine the relevant volume for the PIRT of interest. In this main report, only some of the phenomena are presented in the tables. The preparation of a "significant phenomena" table was not specifically discussed by each panel.

importance, while in the other view, the KL was judged as a relative, rather than absolute factor since it relates to a judgment of whether or not more work is needed.

4.1.3 Panel analysis

Because of the inherent safety features and design philosophy of modular HTGRs, the importance of some phenomena typically of concern in water-reactor accident sequences is not as great, or is not applicable, or may have a different role in NGNP accidents. There are both similarities as well as unique phenomena if one compares water-reactor accident phenomena with the NGNP. The panel evaluated thermal-fluid and neutronics phenomena and processes deemed pertinent to the NGNP's safety features. Four types of challenges were evaluated: challenges to heat removal, reactivity control, and confinement of radioactivity, and challenges to the control of chemical attacks. The complete composite tables and rationales documenting the panel's assessments are contained in Ref. 1.

The prospect of higher-than-expected core temperatures (in normal operation), the concern about RCCS performance in LOFC accident scenarios, peak fuel temperatures in D-LOFC events, and the uncertainties in postulated air ingress accident scenarios that could lead to significant core and core support damage were emphasized by the panel. The panel discussed potential accidents involving the high-temperature process heat (hydrogen) plant, but because that plant design has not yet been selected, the panel opted instead to evaluate one example event for a specific (molten-salt heat transport loop) design.

The more significant phenomena—those rated with high importance (H) and low or in some cases medium knowledge level (L, M)—are highlighted in Table 7. It is recommended that the reader refer to the detailed rationales and assessments in the ACTH report (Ref. 1), as there are many more phenomena identified by the panel that are not listed in Table 7.

As noted in Sect. 3, the ACTH panel organized their PIRT process by major accident scenarios. Accordingly, the following discussion is organized according to the phenomena evaluated for each scenario, starting first with normal operation.

Table 7. Significant ACTH phenomena (high importance and low or medium knowledge rankings)[a]

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
Cavity	Cavity filtering performance (air ingress LOFC) {Affects radioactive dust releases; dust can contribute to the source term for PBR}	H	M[c]	FOM—dose to public—Affects release to public. – Good knowledge base for HEPA filters, design dependent. – Dust filter options should be investigated and tested.
Core Support; Fuel	Molecular diffusion (air ingress LOFC) {Air remaining in the reactor cavity enters into RV by molecular diffusion, prior to onset of natural circulation}	H[c]	M	FOM—core support structure, fuel temperature, dose, fuel failure fraction—Low rate of transport of oxygen not important in driving fuel temperatures; process can occur over a period of days; local circulation may occur before large circulation; will determine onset of natural circulation, number of other factors—operator actions, initial conditions, where break occurs—can override diffusion; don't know how much circulation will be induced by oxidation vs diffusion; slow process will lag other phenomena. Uncertainties in circulation start time can affect severity of event. – Good agreement with calculations under idealized conditions. – Many other factors could influence processes leading to a significant ingress flow rate.
Core Support; Fuel	Core support structures oxidation (air ingress LOFC) {Low-temperature oxidation potentially damaging to structural strength}	H	M	FOM—core support structure, fuel temperature, dose, fuel failure fraction—Core structure area first seen by incoming ingress air. – Complex zone, mixing, heterogeneous, difficult to calculate boundary conditions. – Oxidation behavior of graphite well known.
Core Support Structures	Outlet plenum flow distribution (normal operation). {Affects mixing thermal stresses in plenum and down stream, outlet pressure distribution}	H	L[c]	FOM—worker dose, core support structures—Localized hot spots; excessive thermal gradients may lead to structural problems, and thermal streaking may lead to problems with downstream components such as a turbine or IHX. Problem led to failures in thorium high-temperature reactor (THTR). – Very complex turbulent mixing with incoming jets over large temperature spans. – PMR geometry contributes to the uncertainties in the pressure distribution.

Table 7 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
Fuel	Core coolant bypass flow (normal operation) {Determines active core cooling; affects $T_{max\ fuel}$}	H	L[c]	FOM—fuel time at temperature, fuel failure fraction – Bypass flow varies with shifts in block gaps, etc. – Results in uncertainties in fuel temperatures since there is no way to measure bypass flow. – Medium knowledge of bypass fraction (inferred) with good instrumentation. – Instrumentation in PBRs not practical, poor ability to model phenomena. – Bypass flows vary axially; difficult to measure in-core temperatures. – Test during initial startup for bypass flow cold gas will not leak into core; as a result, less uncertainty in bypass flow. Depend upon code validation; graphite shrink/swell effect on bypass flow. – Knowledge adequate for bounding estimates.
Fuel	Pebble-bed core wall interface effects on bypass flow (normal operation) {Diversion of some core cooling flow. Number of pebbles across impacts interface effects}	H[c]	L[c]	FOM—fuel time at temperature, fuel failure fraction—Combination of cooling anomalies and flux peaking leads to uncertainties. – Pebble-bed pressure drop equations: large uncertainty band with larger uncertainty in wall friction correlations, need experimental data. – Different packing fraction at wall. – Void fraction has large uncertainty. – Calculation tools improved recently. – Heat transfer coupling between flow regime; local values of heat transfer vary significantly from average heat transfer; close to wall there is laminarization of flow. – PBMR doing experiments with high-pressure test unit (HPTU)/heat transfer test facility (HTTF). – Heat transfer calculations in high-temperature regions are difficult.
Fuel	Reactivity-temperature feedback coefficients (normal operation)	H	L[c]	FOM—dose to worker, fuel failure fraction, fuel time at temperature, core support—Important for estimating control rod worth and power defect.

Table 7 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
	{Affects core transient behavior}			– Limited available experimental data for validation of reactivity temperature effects, particularly direct measurements of reactivity coefficients rather than using tests of overall transient response of the system. – Limited data for high-burnup fuels. – High temperature of HTR systems magnifies errors in differential feedback coefficients over that of relatively well-known systems. – Evidence of difficulty in prediction of power coefficients in recent startup experiments. – Physical phenomenon that may be important in accurate calculation of neutron capture in resonances is not accurately modeled in spectral codes; this may have a significant impact of reactivity coefficients (resonance scattering). – Lack of understanding of resonance capture phenomena at high temperatures; need for graphite reactor critical experiments with high burnup; evidence of miscalculation of power coefficients.
Fuel	Fuel performance modeling (normal operation and accidents) {Fuel type dependent. Crucial to design and siting; depends on performance envelope, quality assurance (QA)/quality control (QC), …}	H	L[c]	FOM—fuel failure fraction—Primary barrier. – Many unknowns; kernel migration; silicon carbide morphology relation to release. For D-LOFC, affects defining transient for rated power level.
Fuel	Core effective thermal conductivity (D-LOFC) {Affects T_{Fuel} max for D-LOFC}	H	M	FOM—dose, peak fuel temperature—Major parameter affecting peak fuel temperature in D-LOFC. – Core thermal conductivity uncertainties due to inherent difficulty with comprehensive measurements (both pebble and prismatic cores) – Number of models for effective conductivity exist; lack of consensus on which model is best. – Not all data are available. – Not important in P-LOFC. – More variability in PBR than PMR data.

Table 7 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
Fuel	Power and flux profiles (initial conditions for accidents) (normal operation) {Affects fuel potential for failures in accident conditions due to long-term exposures}	H	M[c]	FOM—dose to public, fuel failure fraction—Major factor in fuel accident performance models. --- – Need for code validation with newer designs—annular core, higher burnup, core reflector interface, fuel location.
Fuel	Decay heat (temporal and spatial) (general LOFC) {Time dependence and spatial distribution major factors in T_{Fuel} maximum estimate}	H	M	FOM—fuel failure fraction—Dependent on fuel type and burnup; major factor in peak temperatures in the D-LOFC accidents but not important for P-LOFC. --- – Spatial dependence calculation is difficult for annular core, axial, and radial peaking factors, inner reflector, higher burnups; need for validation. – Standard correlations appear to be conservative (vs experiments).
Fuel	Fuel performance with oxygen attack (air ingress LOFC) {Consideration for long-term air ingress involving core (fueled area) oxidation; fission product (FP) releases observed for high-temperature exposures}	H	M[c]	FOM—fuel temperature, dose, fuel failure fraction—Low probability; fueled core area of exposure probably at temperatures less than critical for FP release. --- – Uncertainties in accident calculations due to wide variety of possible conditions. – Fuel qualification. – Active R&D. – Much oxidation data based upon fresh fuel; need more data on irradiated fuel.
Fuel	Phenomena (various accident conditions) that affect cavity gas composition and temperature with inflow (air ingress LOFC) {Provides gas ingress and cold-leg conditions; needed to calculate ingress flow rate and properties. Possible entrainment through relief valve, etc.}	H[c]	M	FOM—fuel temperature, dose, fuel failure fraction, core integrity—In terms of overall damage to reactor core, it is a question of total oxygen available over course of accident, not specific composition; and impact on corrosion, conservative assumptions would result in less importance of phenomena. --- – Very complicated; various phenomena; difficult to know composition and temperature at inlet. – Link transient to opening of vent valve; pulses can affect phenomena. – Bounding calculations can define limits within large uncertainties. – How much air is carried out with valve break (size dependent; large break with vent valve more important).

Table 7 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
Fuel	Confinement-to-reactor cavity air ingress (air ingress LOFC) {Determines long-term oxidation rate if accident unchecked}	H	M	FOM—fuel temperature, dose, fuel failure fraction, core integrity—Defines long-term damage. – Lack of data on pressure differential between confinement and cavity. – Performance criteria provided by confinement vendor.
Fuel	Reactivity temperature feedback coefficients (fuel, moderator, reflectors) [reactivity (ATWS)] {Affects passive safety shutdown characteristics}	H	M[c]	FOM—fuel failure fraction, time at temperature—Inherent defense against reactivity insertions; major argument for inherent safety design. – Lack of understanding of resonance capture phenomena at high temperatures; need for graphite reactor critical experiments with high burnup; evidence of miscalculation of power coefficients. – Calculations of absorber worths can have large differences based on fixes to diffusion theory approach. – Control rod worths impacted by core axial power distribution, which may be difficult to predict because of temperature and burnup distributions. – Measurement of control rod worths generally performed as part of reactor startup procedures.
Fuel and Core	Core oxidation (air ingress LOFC) {Determination of "where" in core the oxidation would take place, graphite oxidation kinetics affected by temp oxygen content of air, irradiation of graphite}	H	M	FOM—fuel temperature, dose, fuel failure fraction, core integrity—Oxidation might occur at the top of the core, depending upon break location. – Data needed on effects of radiation damage on graphite. – Existing data from experiments varies with geometries and manufacturers. – Need to reduce uncertainties in graphite oxidation data.
Primary System; Secondary Side	Fission product transport through IHX loop (part of confinement bypass) [IHX failure (molten salt)] {Deposit/removal of FP, dust, scrubbing of molten salt, adsorption, plate-out}	H	M	FOM—public and worker dose—Determines activity released out of IHX relief valve, and residuals in IHX loop. Lack of scrubbing data applicable to countercurrent helium–MS flow, yet bounding models may be able to reduce uncertainties. [This postulated event was a "sample consideration" by the ACTH panel for possible accidents related to the process heat plant. A molten salt heat transport loop design was arbitrarily assumed.]

Table 7 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
Vessel	Vessel and RCCS Panel emissivity (general LOFC) {Radiant heat transfer from vessel to RCCS affects heat transfer process at accident temperatures}	H	M[c]	FOM—vessel integrity—maintain coolable geometry; limit vessel temperature—Change in inner surface vessel emissivity based on degraded environment; T^4 (radiant) heat transfer dominates (85–90%) in LOFC transients; and scoping calculations: large temperature differences between vessel and RCCS reduce emissivity importance. – In-service steel vessel emissivities are fairly well known. – Emissivities not well known during accidents as a function of time, dust on surface, optical transparency, etc., as a result of disturbances from a depressurization. – Knowledge of inner emissivity 0.5–>0.3, change nature of surface coating; e.g., from loss of oxide film. – Emissivities are fairly well known for steel, once oxidized (in air cavity). – Complex geometries involved—difficult to calculate for transient cases, especially in upper head region with control rods (standpipes) in between vessel and RCCS.
Vessel	Reactor vessel cavity air circulation and heat transfer {Affects upper cavity heating}	H	L	FOM—vessel and vessel support integrity—Affects RCCS performance; skewed (toward top) heat distribution; generation of hot-spots.
Primary and secondary system hardware	Ag-110*m* release and plate-out	H[c]	L	FOM—worker dose—Large uncertainty level; a function of fuel type, burnup, and temperature. Could be a maintenance (dose) problem for gas turbine maintenance (if direct cycle). FP release mechanism (from TRISO particle) not understood.

[a]Events selected from Tables 2.1 through 2.7 in Ref. 1.
Text in parentheses { } in the Phenomenon column is added to elaborate on the phenomenon.
[b]H, M, or L (high, medium, or low).
[c]Average or consensus ranking involved diverse opinions.

4.1.3.1 Normal operation

A major safety-related concern is the uncertainty in the core coolant bypass flow, which is very difficult or impossible to measure or even infer in HTGRs. Core coolant bypass flow was ranked as high importance (H), with the knowledge level low (L*), or overall an (H, L*) ranking, indicating suggested further study. (The "*" indicates that the average or consensus ranking involved diverse opinions.) Other mechanisms related to core-coolant-flow distributions and their variations were ranked (M, L) or (H, M).

Power/flux profiles in PBRs (H, L) were of concern to the panel due to the history of problems with prediction of pebble operating temperature, particularly in the Atomgemeinschaft Versuchs Reaktor (AVR), and also due to the lack of operating experience with tall annular cores. Uncertainties are exacerbated by the flux's tendency to peak sharply at pebble-reflector wall interfaces.

The panel ranked the reactivity-temperature feedback coefficients as (H, L) due to the lack of experimental data for this specific core configuration and the eventual large plutonium content, which increases with burn-up due to the use of low-enriched uranium (LEU). Tests run at experimental reactors [the Japanese high-temperature engineering test reactor (HTTR) and Chinese high-temperature reactor (HTR-10)] have shown good agreement with predictions (for low burn-ups), but concerns about higher burn-up were expressed.

Other phenomena characterized as (H, L) by the panel included the outlet plenum flow distribution. While not considered to be a primary safety issue, this phenomenon raises concerns about the effects of possible hot steaks in the helium on stresses in the plenum and outlet duct (and the downstream gas turbine, where applicable).

Fuel performance modeling was also ranked as very important (H, L) by the panel since such performance is a crucial factor in determining the source terms.

Another (H, L)-ranked phenomenon relates to fission product release and transport of silver (Ag-110*m*), where, for example, the potential for deposition on turbine blades for direct-cycle gas-turbine (balance of plant) is a maintenance or worker dose concern. Silver is released from intact silicon carbide (SiC) coating layers on TRISO particles by a yet-to-be-understood mechanism, primarily at very high operating temperatures and high burn-ups. The problem is likely to be greater for plutonium-bearing fuel since the fission product silver generation from ^{239}Pu fissions is ~50 times greater than for ^{235}U fissions.

4.1.3.2 General LOFC

The building block approach led to the creation of a general LOFC table (G-LOFC) that included common elements for the variations on the LOFC theme. It also enabled adding air ingress phenomena to the D-LOFC PIRT or a reactivity event to any LOFC case. RCCS behavior is generally very important in LOFC events since the RCCS becomes the only effective means of removing afterheat from the core and vessel.

One phenomenon ranked as (H, L) was the emissivity estimate for the RPV surface and RCCS panel, particularly due to uncertainties from aging effects. Emissivities are key factors in the ultimate heat sink performance in LOFCs since at high temperatures most of the heat removal (~80–90%) is by thermal radiation from the RPV to the RCCS, the rest being by thermal convection in the RPV cavity. Steels have been shown to have high emissivities (~0.8) at high temperatures given that an oxide layer (typically formed in most service conditions) is intact; however, there was concern that this layer might be compromised, resulting in significantly lower emissivities.

The other phenomenon given (H, L) ratings was the RPV cavity air circulation and heat transfer. While this typically provides a small fraction of the total heat removal in LOFCs, it is crucial to temperature distributions within the RPV cavity.

4.1.3.3 P-LOFC

In the P-LOFC case, the main concern shifts to the tops of the core and vessel, which become the hottest, rather than the coolest, areas. While no phenomena were given (H, L) rankings, several concerns rated (H, M) related to the convection and radiation heating of the upper vessel area and the design of the special insulation inside the top head. High-temperature insulation development is typically an important issue in HTGR designs due to considerations such as behavior during rapid depressurization events, which may tend to dislodge it.

4.1.3.4 D-LOFC

Following a depressurization, the effective core conductivity and afterheat become the two major influences on peak fuel temperatures. The D-LOFC accident is typically the design determinant for reactor maximum operating power level (for a given vessel size).

Although no phenomena received (H, L) rankings, considerable attention was given to the uncertainties in core effective conductivity and afterheat (for fuel temperatures) and to RCCS performance (for vessel temperature).

Fuel performance modeling, as it applies to heat-up accidents, was also ranked (H, M), noting its importance and the need to adapt it to each fuel design. The fuel quality assurance and control aspects of fuel manufacture in tandem with operating conditions, in addition to heat-up temperature trajectories, were noted by the panel.

Dust suspension in the RPV cavity (caused by depressurization) could impede the radiant heat transfer from the vessel to the RCCS. The radioactive dust in the primary circulating gas released to the confinement, along with other dust that becomes loose, is typically considered to be a major source term factor for PBRs (see evaluation by the FPT PIRT).

4.1.3.5 Air ingress following depressurization

Events involving significant air ingress, while of very low probability, add many complications and potential degrees of severity to the already complex D-LOFC event. The two primary factors are the rate of ingestion of "air" into the core area and the oxygen content of that gas.

For single-break scenarios, there can be long delays before a significant air ingress flow occurs, allowing major shifts in core temperature profiles before the onset of oxidation. The process of air encroaching into the space originally occupied by helium (molecular diffusion) is typically a very slow process, and as long as the helium "bubble" in the top region of the vessel is intact, substantial ingress flow is inhibited.

The scenario in which forced convection augments the air ingress process was not specifically considered. This scenario has the potential to increase considerably the net graphite oxidation rates (clearly a more bounding event of concern).

There are also wide variations in the possible composition of the ingress gas, depending on the location of the break in the RPV cavity, in- and out-leakages in the confinement, and many other design-dependent attributes. Factors such as gas density and stratification affect the predictions. As a result, bounding calculations with very conservative assumptions are seen as a possible approach, especially until more design details are available.

The possibility for a double break that exposes both the reactor upper and lower plenums to the confinement cavity atmosphere was also considered, even though any double-vessel break would be of extremely low probability. A chimney effect results in larger ingress flows with minimal delay. However, total long-term graphite oxidation damage is more dependent on total oxygen availability in the confinement building.

The integrity of the graphite core support system also depends on design details as well as the conditions for oxidation, where oxidation at lower temperatures tends to result in more structural damage. This phenomenon was ranked as (H, M) by the ACTH PIRT panel and was considered as well by the GRAPH PIRT panel. Potential damage to the fuel from oxidation, ranked as (H, M*), was a concern; however, it was noted, based on experimental data, that the SiC coating layer in TRISO fuel retains fission products well when exposed to air in the temperature ranges expected in the ingress scenarios.

4.1.3.6 Reactivity (including ATWS) events

The most commonly postulated ATWS events are those accompanying LOFCs (either P- or D-), and they are typically extremely low-probability events, usually falling outside the BDBA envelope.

A reactivity insertion from pebble-bed core compaction due to a severe, prolonged earthquake event was not seen as a major concern since the reactivity increase would likely occur over a relatively long time period (minutes), and that even without a scram, the negative temperature-reactivity feedback mechanisms would prevent fuel failures from over-temperature.

The possibility of significant positive reactivity insertions from steam/water ingress was seen as unlikely due to the lack of credible mechanisms for significant ingresses (the conclusion being predicated on the designs not including a steam generator in the primary circuit).

The temperature-reactivity feedback coefficients for the fuel, moderator, and reflectors were ranked as (H, M*) since negative feedback is essential to the inherent defense against reactivity insertions. Other panel concerns were associated with complex and untested core design features such as the very tall annular core, particularly for high-burn-up conditions in the core.

4.1.3.7 IHX failure, assuming MS as the transport medium

Since very large uncertainties remain in the process heat plant design, the panel decided to evaluate an example case of a failure in the IHX and heat transport pipeline. MS was chosen as the intermediate heat-transport coolant. The postulated event led to primary system helium penetration into the heat transport loop and possible release of part of the helium's circulating activity to the environs, followed by back-flow of MS into the reactor primary system, and eventually into the reactor core.

There were no (H, L) panel rankings in this category, although some concerns were raised about possible doses from the initial release of activity from the primary circuit.

4.1.3.8 Water-steam ingress events

Originally the intent was to cover events including potential design options for a steam generator (SG) in the primary loop. In this case, steam in-leakage from a high-pressure SG would be a dominant risk factor. Otherwise, primary water-cooled heat exchanger secondary systems (in Brayton cycle designs) would run at lower operating pressures and present minimal risks of any substantial water-steam ingress. Hence, the panel decided to eliminate this accident type from the current ranking process.

4.2 Fission Product Transport and Dose (FPT) PIRT Panel

4.2.1 Phenomena identification and description

The analysis of FPT phenomena must involve all three phenomenological levels: the system level to define the specific scenario; the component level to determine the overall fluid flow and temperatures; and finally the local level to determine relevant material properties, chemical interactions, and fission product mass (and dust) fluxes. The system and component levels may be thought of as setting the fluid flow and thermodynamic environment, while the local level determines the fluxes into and out of the components and surfaces. The knowledge base is detailed in the next section as it is an intrinsic part of evaluating the transport path. The FPT panel considered the operational modes as described Sect. 3.3 and also analyzed normal operation. With regards to accidents, the panel focused much of the discussion on the P-LOFC and D-LOFC accidents.

4.2.2 Ranking rationale

The four FOMs selected by the FPT were:

- Level 1 (Regulatory): Dose to control room and offsite location
- Level 2 (System): Release to confinement
- Level 3 (Component): Release into primary system
- Level 4 (Subcomponent): Release from graphite in fuel form

These FOMs were used as the basis to determine the source term release from the fuel. The movement into the reactor system and the confinement, and subsequent release into the environment, was also considered. Four major areas of concern implied by the FOMs are as follows:

1. The inventory of the fission products outside of the fuel. This inventory is due to coating manufacturing defects, uranium contamination outside the coated particles (which upon fissioning increases circulating activity), and in-service failures. Fission products that are released due to accidents are also considered.
2. Total curies released into the confinement, including the fission products of radiological (dose) interest.
3. Total fission product and transuranic curies which penetrated all the boundaries and are released into the environment and affects offsite dose.
4. Timing aspects of the release(s), including the history of the release(s).
5. Panel analysis of Accident and Thermal Fluids phenomena.

4.2.3 Panel analysis

Generally, panel convergence is seen on most issues, but different approaches to the specific physics and transport paths shade the answers accordingly. One item of particular interest is the final approach to the ranking process. Two methods are apparent—the identification of the phenomena in either a general or a path-dependent way. The general identification method allows one to collect all the items of interest without specifically outlining a transport path within the ranking table. This method avoids forcing a specific transport path model on the analyses but may not clearly identify the relative importance of particular phenomena along a specific path. The path-dependent approach allows the reader to see the importance of the particular phenomena along a path but requires the identification of the transport subpaths. These paths were based on historical work because of the lack of a specific NGNP design but should be relevant unless some truly unique design is proposed. Even with these two approaches to the PIRT table layout, the results are very similar. The complete composite tables and rationale documenting the panel's assessment are contained in Ref. 2.

Significant phenomena [those with high (H) importance and low or medium (L, M) knowledge level] and the associated rationale are highlighted in Table 8. Selecting phenomena based on high importance and low or medium knowledge level may not constitute a complete assessment of the situation. It is recommended that the reader refer to the panel's report (Ref. 2) for more information.

Depending on the design of a confinement or containment, the impact of a primary system pressure boundary breach can be minimized if modest, but not excessively large, fission product attenuation factors can be introduced into the release path. This exercise has identified a host of material properties, thermal fluid states, and physics models that must be collected, defined, and understood to evaluate such attenuation factors. Because of the small allowable releases during a depressurization from this reactor type (into a vented confinement), dust and aerosol issues are important to quantify even though the amounts of fission products involved may be modest [compared to potential aerosol generation in a severe light-water reactor (LWR) accident]. The initial fission product contamination of the reactor circuit is of great importance because the most powerful driving term, helium pressure, will most likely act during the earliest stages of the accident. If an air ingress accident occurs with an unimpeded flow path, larger fission product releases can occur later in the accident.

Another issue of importance is the approach to modeling graphite properties. Technically, this issue is beyond the scope of the FPT PIRT, as the panel was to focus only on phenomena, but it does impact how one approaches the collection of data for the models. Briefly, one approach is basic physics in nature, and the other is more empirical. The basic physics approach would have the advantage that measured graphite and fission product properties can be related to transport over a wide range of situations, but the physics may be very challenging. The empirical approach offers less theoretical complexity but may be limited by the cost of experiments and the range of accidents that can be covered. In any event, this issue would have to be resolved by a review of the state of the art in graphite and transport theory and would be influenced by the specific safety approaches taken by the reactor designers.

Finally, one phenomenon that was rated as important and may not have been explored in the past was the effect of mechanical shock and vibration in a D-LOFC on the transport and re-entrainment of dust and spalled-off oxide flakes. A failure of a large pipe would generate large mechanical forces (vibration, shocks, and pipe whip), and the resulting flow can generate a large amount of acoustic energy, both of which can launch dust and small particles into the existing gas flow as well as cause additional failures. Much of the literature is concerned with changes in temperature and flow velocity during an accident, but these impulsive and vibratory mechanical effects should also be considered, especially if the reactor internal surfaces are required to retain fission products during an accident to meet safety requirements. The internal surfaces will then take on many of the qualities of a safety system since they will have the formal function of retaining fission products during the course of an accident.

Table 8. Significant FPT phenomena (high importance and low or medium knowledge rankings)[a]

System[b] or component	Phenomena	Importance	Knowledge level	Rationale
Confinement	Radiolysis effects in confinement	High	Medium	FOM—dose to control room and off-site location—FP (e.g., I, Ru, Te) chemistry, paint chemistry. Dose will be dependent on confinement radiation level (Trans.). LWR experience and data applicable to some extent.
Confinement	Combustion of dust in confinement	High	Medium	FOM—dose to control room and off-site location—Source of heat and distribution of FPs with in confinement. Data from international Tokomak (magnetic confinement fusion) experiment (ITER) development may be applicable. Contingent on specific design knowledge.
Confinement	Confinement leakage path, release rate through penetrations	High	Medium	FOM—dose to control room and off-site location—Cable/pipe penetrations, cracks, holes, heating ventilating air conditioning (HVAC) provide potential leakage paths (Trans.). Building leakage experience, design specific.
Confinement	Cable pyrolysis, fire	High	Medium	FOM—dose to control room and off-site location—Soot generation and changes to iodine chemistry. LWR experience.
Core	Recriticality (slow)	High	Medium	FOM—release from graphite in fuel form—release into primary system—release to confinement—dose to control room and off-site location—Additional thermal load to fuel. Increases source but not expected to affect transport path. Heat load easily computed with existing tools; effect on fission products not completely known.
Fuel	Fuel-damaging RIA	High	Medium	FOM—release form graphite in fuel form—release into primary system—release to confinement—dose to control room and off-site location—An intense pulse could damage fuel. Increases source but not expected to affect transport path. Some data exist, but outside of expected accident envelope.
Fuel and Primary Coolant System	Dust generation	High	Medium	FOM—release from graphite in fuel form—release into primary system—pathway for FP transport; possibility of high mobility. Limited experience; lack specific system information.
Graphite and Core Materials	Matrix permeability, tortuosity	High	Low	FOM—release from graphite in fuel form—Needed for first principle transport modeling provides initial and boundary conditions for transient and accident analysis (IC and Trans.). FP holdup as barrier, release as dust; expected from material PIRT.

Table 8 (continued)

System[b] or component	Phenomena	Importance	Knowledge level	Rationale
Graphite and Core Materials	FP transport through matrix	High	Low	FOM—release from graphite in fuel form—Effective release rate coefficient (empirical constant) as an alternative to first principles (IC and Trans.). FP holdup as barrier, release as dust; expected from HTMAT PIRT.
Graphite and Core Materials	Fuel block permeability, tortuosity	High	Medium	FOM—release from graphite in fuel form—Needed for first principle transport modeling (IC and Trans.). Depends on specific graphite; expected from HTMAT PIRT.
Graphite and Core Materials	FP transport through fuel block	High	Medium	FOM—release from graphite in fuel form—Effective release rate coefficient (empirical constant) as an alternative to first principles (IC and Trans.). Depends on specific graphite; expected from HTMAT PIRT.
Graphite and Core Materials	Sorptivity of graphite	High	Medium	FOM—release from graphite in fuel form—release into primary system—Historical data, need specific information on graphite and radiation effects. Depends on specific graphite; expected from HTMAT PIRT.
Graphite and Core Materials	Fluence effects on transport in graphite	High	Medium	FOM—release from graphite in fuel form—release into primary system—Influences transport, chemical reactivity. Historical data; need specific information on graphite and radiation effects.
Graphite and Core Materials	Air attack on graphite	High	Medium	FOM—release from graphite in fuel form—release into primary system—release to confinement—dose to control room and off-site location—Graphite erosion/oxidation, Fe/Cs catalysis liberating FPs (Trans.). Historical data largely applicable.
Graphite and Fuel	FP speciation in carbonatious material	High	Low	FOM—release from graphite in fuel form—release into primary system—Chemical form in graphite affects transport (IC and Trans.). Uncertain and/or incomplete.
Graphite and Fuel	Steam attack on graphite	High	Medium	FOM—release from graphite in fuel form—release into primary system—If credible source of water present; design dependent (Trans.). Historical data largely applicable.

Table 8 (continued)

System[b] or component	Phenomena	Importance	Knowledge level	Rationale
Graphite in Primary System	*(De)Absorption on dust*	*High*	*Medium*	FOM—*release from graphite in fuel form—release into primary* system—Provides copious surface area for FP absorption. Limited experience; lack specific details. Historical data from Peach Bottom HTGR largely applicable.
Primary Coolant System	Material/structure properties (critical initial and/or boundary condition)	High	Medium (graphite)	FOM—release into primary system—Density, viscosity, conductivity, etc., important parameters in calculations (IC and Trans.). Properties are well-known for steel and concrete, but graphite type not yet selected; data expected from HTMAT PIRTs. Well known for IC.
Primary Coolant System	Thermal-fluid properties	High	Medium	FOM—release into primary system—release to confinement—Temperature, pressure, velocity computations (IC and Trans.). Well known for helium; uncertainty in composition of gas mixtures makes gas property calculation more difficult; expected from ACTF PIRT.
Primary Coolant System	Gas composition	High	Medium	FOM—release into primary system—Oxygen potential and chemical activity. Central issue for chemical reaction modeling, FP speciation, scenario dependent.
Primary Coolant System	Gas flow path prior, during and post accident	High	Defer to ACTF PIRT	FOM—release into primary system—release to confinement—dose to control room and off-site location—Information needed to model accident (IC and Trans.). Need to coordinate with other groups; expected from ACTF PIRT.
Primary Coolant System	FP speciation during mass transfer	High	Medium	FOM—release from graphite in fuel form—release into primary system—Chemical change can alter volatility. Historical data; need specific information. Good for metals, oxides. Uncertain for carbides and carbonyls.
Primary Coolant System, Cavity, Confinement	Ag-110*m* generation, transport	High (O&M) Low (release)	Low	FOM—release from graphite in fuel form—release into primary system—Radioisotope, significant as potential O&M dose on cool, metallic components. Not significant as a potential dose to public from releases. Limited data, unknown transport mechanism.
Primary Coolant System, Cavity, Confinement	Aerosol growth	High	Low	FOM—release into primary system—release to confinement—dose to control room and off-site location—Low concentration growth can lead to high-shape factors and unusual size distribution. Regime has not been studied previously.

Table 8 (continued)

System[b] or component	Phenomena	Importance	Knowledge level	Rationale
Primary Coolant System, Cavity and Confinement	Resuspension	High	Low	FOM—release into primary system—release to confinement—dose to control room and off-site location—Flow/vibration induced, saltation; mechanical forces can release FPs from pipe surface layers films (Trans.). Lack of data and models for anticipated conditions.
Primary Coolant System, Cavity and Confinement	Aerosol/dust deposition	High	Medium	FOM—release from graphite in fuel form—release into primary system—release to confinement—dose to control room and off-site location—Gravitational, inertial, thermophoresis, electrostatic, diffusional, turbophoresis (Trans.). Reasonably well-developed theory of aerosol deposition by most mechanisms except inertial impact in complex geometries; applicability to NGNP unclear. Theory, data, and models lacking.
Primary Coolant System/Fuel	FP plate-out and dust distribution under normal operation	High	Medium	FOM—release from graphite in fuel form—release into primary system—Starting conditions. Theory and models lack specifics.
Reactor Coolant System and Confinement	FP diffusivity, sorbtivity in nongraphite surfaces.	High	Low	FOM—release into primary system—Determines FP location during operation; acts as a trap during transient (IC and Trans.). Little information on surface materials (and operating conditions) of interest.
Reactor Coolant System and Confinement	Coolant chemical interaction with surfaces	High	Medium	FOM—release into primary system—release to confinement—Changes oxygen and carbon potential which can affect nature and quantity of sorbed species (IC and Trans.). Surface properties are critical; need alloy data.

[a]From Table 10 in Ref. 2.

[b]It should be noted that in many cases, fission product transport phenomena can involve many components and systems, so this column should not be construed to be all-inclusive.

Notes: *IC—initial condition, the result of long-term normal operation.*
Trans—transient and accident condition.

Design aspects and accident scenarios were used as the basis for this exercise; so rather than focusing on the actual details of the scenario, the panel focused on the results of the scenarios that would significantly impact the release of fission products:

- *Large and small pressure boundary breaches.* These breaks and leaks were assumed to have the potential to release not only the material entrained in the gas during normal operation, but also material such as dust and fission products on metal surfaces.
- *Releases from the cleanup and holdup systems.* Breaks and leaks in these systems can release fission products to the confinement. These systems are only vaguely defined at the present time, but, in addition to the historical inventory of inert gases and perhaps iodine, newer designs may include a facility for removing dust.

Implicit in the needs of the FPT transport analysis are the models for determination of the fission product distribution in the core and reactor circuit during normal operating conditions since this is the starting point for the accident (and of course is very design specific). Simulation of the accident will require the addition of dust entrainment models and chemical reaction models. The description below covers accident scenarios that were analyzed by the FPT panel.

4.2.3.1 P-LOFC fission product transport

The major concern with the P-LOFC is how it may change the distribution of fission products prior to a pressure boundary breach since the event itself does not release fission products to the confinement. If the P-LOFC results in a pressure relief valve opening, with or without sticking, a fission-product transport path will be generated. This path is design specific since a filter may be incorporated into the exhaust circuit.

4.2.3.2 D-LOFC-fission product transport

The D-LOFC event has a two-part impact on FPT. The first is the initial depressurization, which releases fission products from the primary circuit via the blow-down/depressurization, any system vibrations, and source term entrainment by the discharge flow. This event can be the most important since some conceptual reactor building designs do not include a provision for filtering this rapid high-volume flow. Combustion of dust may add heat and more completely distribute the fission products in the confinement volume.

The second item of interest occurs after the depressurization and the heat-up of the core and reactor system. The higher temperatures (which are calculated by the accident codes) can cause the redistribution of fission products (and perhaps some limited fuel failure, depending on the design margins and quality of the fuel). However, the driving force for the release of fission products to the environment is only the very weak thermal expansion of the gas. In addition, at this point in the accident, the building filters are expected to be operational in most designs.

4.2.3.3 D-LOFC-with air ingress

The more extreme version of the D-LOFC accident is the significant and continued flow of air into the core, which is only possible with a major reactor building and reactor system fault that establishes a convective air path between the reactor vessel and the environment. In this case, high fuel temperatures are possible, high fission product release is unlikely but possible, and a convective path is available for the transport of material out of the building. Three mechanisms are then available for the enhancement of fission product releases and transport.

1. Locally increased temperatures due to graphite oxidation can drive the movement of the volatile fission products such as cesium and, if high enough, increase the amount of failed fuel and subsequent fuel releases.

2. The destruction of the graphite and matrix material could release the trapped fission products, which can then be carried along with the flow as particles, vapors, or aerosols.
3. The increased oxygen potential of the reactor environment may change the chemical forms of the fission products and surfaces with which they interact.

Graphite oxidation with core consumption (and possible partial or total collapse) is a complex process highly dependent on the particular design, structural materials, accident scenario, and the design safety margins. The key features are the flow path, the temperatures, and the amount of oxidizer available. The free flow of oxidizer may need to be stopped early in the accident to prevent serious fission product releases from the core.

4.3 High-Temperature Materials

4.3.1 Phenomena identification and description

The HTMAT (Ref. 3) and GRAPH (Ref. 4) PIRTs were performed by separate panels. Phenomena identification for the HTMAT PIRT focused on material strength, ductility, toughness, effects of radiation, material compatibility with the coolants (and associated impurities), material thickness, and joining methods. Key components considered include the low alloy steel for the reactor pressure vessel and piping, core barrel, and various components of the turbomachinery. Creep-fatigue properties were also of concern, as well as the aspects of flaw assessment and crack propagation.

4.3.2 Ranking rationale

The panel established FOMs related to each system or component. These were the criteria against which importance of phenomena is judged. While these are often derived from regulations (e.g., dose limit, siting criteria) at top levels and related to the issue being addressed, and scenario and component selected at subsidiary levels, in all cases the FOMs provided guidance with regard to the likelihood of radiation release at the site boundary.

The process by which the panel developed the FOMs is described since it is important to understand the relationship between the reactor system or component being considered, the FOM itself, and the potential development of a pathway for the release of fission products at the site boundary. The first step the panel took was to identify the major reactor system or structural components that were felt to have the potential to contribute to fission product release, such as the RPV, the piping, etc. Criteria were then established by which the significance of individual phenomenon could be evaluated with regard to their contribution to release at the site boundary (e.g., maintaining the integrity of the pressure boundary in the RPV or piping, limiting the peak temperature that the fuel might see, maintaining the geometry of core support structures and their related nuclear characteristics, etc.). These criteria were the FOMs. The component-specific phenomena were then evaluated against each FOM for its contribution to fission product release via a specific pathway (e.g., breach of piping or pressure vessels, excessive deformation of core supports, coolant flow blockage from debris, or component passage collapse).

Hence, it is important to understand that each phenomenon identified is ranked for its importance and knowledge base with respect to a particular component, FOM, and pathway to release. The FOMs and the associated phenomena (shown in parenthesis) were categorized by component or reactor system as below:

- Control Rods (both nonmetallic and metallic)
    - FOM—Insertion Capability (failure to insert)

- Reactor Pressure Vessel (RPV)
    - FOM—RPV Integrity (breach, excess deformation)
    - FOM—Peak Fuel Temperature (inadequate heat transfer)
- RPV—Metal Internals
    - FOM—Maintain Heat Transfer Capability (inadequate heat transfer)
    - FOM—Structural Geometry (excess deformation and fracture/failure)
    - FOM—Core Barrel Integrity (failure)
    - FOM—RPV Integrity (failure)
- RPV—Nonmetallic Internals
    - FOM—Structural Geometry (core restraint and support failure)
    - FOM—Insulation Capability (fibrous insulation degradation)
- Power Conversion and Turbomachinery
    - FOM—Primary System Integrity (breach of vessel, turbine failure)
    - FOM—Rotating Equipment (breach of vessel, turbine failure)
- Circulators
    - FOM—Primary System Pressure Boundary Integrity (oil-bearing failure, impeller failure)
    - FOM—Integrity of Rotating Equipment (oil-bearing failure, impeller failure)
- Piping
    - FOM—Primary System Integrity (breach, failure to insulate)
    - FOM—Rotating Equipment (insulation debris generation)
- Intermediate Heat Exchanger Vessel
    - FOM—Integrity of IHX (breach to ambient)

      FOM—Integrity of vessel (breach to ambient)
- Intermediate Heat Exchanger
    - FOM—Integrity of IHX (breach to secondary system or breach—secondary to primary)
    - FOM—Secondary Loop Failure/Breach (breach to secondary system)
    - FOM—Integrity of Hot Duct and Other Systems (breach from secondary to primary)
    - FOM–Integrity of IHX (catastrophic loss of function)
- Reactor Cavity Cooling System
    - FOM—Emergency Heat Removal Capability (inadequate heat removal)
- Auxiliary Shutdown System
    - FOM—Primary System Pressure Boundary Integrity (water contamination of primary coolant)
- Valves
    - FOM—Primary System Pressure Boundary Integrity (malfunction, failure to operate and breach)

4.3.3 Panel analysis

This NGNP is similar to another HTGR design for which the NRC had been requested to perform a preapplication evaluation in previous years. However, there are a few notable differences, such as a higher outlet gas temperature and a direct-cycle turbine plus the use of an IHX. Phenomena evaluations were made considering these differences and their impacts on core components. Table 9 lists selected phenomena that the panel considered to be of particular significance, with combinations of high importance and a low to medium knowledge ranking. The complete composite tables and rationale documenting the panel's assessment are contained in Ref. 3.

These tables describe the selected phenomena that the panel considered to be of particular significance due to their combination of a high ranking of importance (H) and a low or moderate knowledge ranking (L, M). Selecting phenomena based on high importance and low or medium knowledge level may not constitute a complete assessment of the situation. It is recommended that the reader refer to the panel's report (Ref. 3) for more information.

Table 9. Significant HTMAT phenomena (high importance and low or medium knowledge rankings)[a]

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
Control Rods (nonmetallic)	Composites structural design methodology limitations for new structures (lack of experience)	H	L	FOM—maintain insertion ability—Carbon–carbon (C–C) composites are prime candidates but need approved method of designing, proof testing, model testing, testing standards, design methods, and validation tests. Some code work is being developed by American Society of Mechanical Engineers (ASME), and American Society for Testing Metals (ASTM). Extensive aerospace industry design and usage can be assessed for applicability.
Intermediate Heat Exchanger (IHX)	Crack initiation and propagation [due to creep crack growth, creep, creep-fatigue, aging (with or without load), subcritical crack growth]	H	L	FOM—integrity of IHX—secondary loop failure/breach—Environmental effects on subcritical crack growth, subject to impacts of design issues, particularly for thin section must be addressed. Stresses on IHX (both thin and thick sections) can lead to these failure phenomena; thermal transients can cause toughness concerns; carbide redistribution as a function of thermal stress can change through-thickness properties, loading direction. More is known about Alloy 617 from HTGR and industry usage than for Alloy 230. Both environment and creep play significant roles in initiation and cyclic crack growth rate of 617 and 230. Mechanistic models for predicting damage development and failure criteria for time-dependent phenomena have to be developed to enable conservative extrapolation from short-term laboratory test data to long-term design life.
Intermediate Heat Exchanger (IHX)	Primary boundary design methodology limitations for new structures (lack of experience)	H	L	FOM—integrity of IHX—secondary loop failure/breach—Time-dependent design criteria for complex structures need to be developed and verified by structural testing. ASME Code-approved simplified methods have not been proven and are not permitted for compact IHX components. No experience for the complex shape IHX. No experience for designing and operating high-temperature components in the (safety) class 1 environment. Difficulties of design and analyses of compact IHX are discussed in Refs. 16–25 of Volume 4–High-Temperature Materials PIRT (Ref. 3).
Intermediate Heat Exchanger (IHX)	Manufacturing phenomena (such as joining)	H	L	FOM—integrity of IHX—secondary loop failure/breach—Compact heat exchanger (CHE) cores (if used) will require advanced machining, forming, and joining (e.g., diffusion bonding, brazing, etc.) methods that may impact component integrity. Must assess CHE vs traditional tube and shell concepts. However, these phenomena are generic and extend beyond the compact HXs to all the very high-temperature HXs.

Table 9 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
				Compact HXs have not been used in nuclear applications; the candidate alloys and their joining processes not adequately established in nonnuclear applications.
Intermediate Heat Exchanger (IHX)	Inspection/testing phenomena	H	L	FOM—integrity of IHX—secondary loop failure/breach—Traditional nondestructive evaluation (NDE) methods will not work for CHEs because of geometrical constraints. Proof testing of some kind will be required (maybe leak testing with tracer). Preservice testing will be difficult, and in-service testing will be even harder. Condition monitoring may be useful. Preoperational testing, preservice inspection, fitness for service, issue with leak tests, have very little knowledge here. Uncertainties in the margins
Piping	Aging fatigue, environmental degradation of insulation	H	L	FOM—peak fuel temperature—Concern is about insulation debris plugging core cooling channels, causing damage due to chunks of internal insulation falling off (ceramic sleeves or C–C composites would be most likely source of problems). Little system-relevant information about insulation failure mechanism is available.
RPV Internals (metallic)	Change in emissivity	H	L	FOM—maintain heat transfer capability—To ensure passive safety, high emissivity is required to limit core temperatures—(affect coolant pathway, high emissivities on both surfaces of the core barrel, formation and control of surface layers, consider under helium environments). Limited studies on SS and on Alloy 508 show potential for maintaining high emissivity.
RPV Internals (metallic)	Radiation-creep	H	L	FOM—maintain structure geometry—Irradiation creep and dimensional changes particularly for Alloy 800H at moderately low-dose should be assessed. Little information on irradiation creep is available for Alloy 800H.
RPV Internals (nonmetallic)	Composites structural design and fabrication methodology limitations for new structures (lack of experience)	H	L	FOM—maintain structure geometry—C–C composites are prime candidates but need approved method of designing, proof testing, model testing, testing standards, design methods, validation tests, scalability issues, fabrication issues, probabilistic methods of design. Must address large-scale (meters in diameter) structures as well as smaller ones. Extensive experience within the aerospace industry; applicability must be assessed.

Table 9 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
RPV Internals (nonmetallic)	Environmental and radiation degradation and thermal stability at temperature	H	L	FOM—Maintain insulation capability—Relatively low dose and exposure is expected, but LOFC can result in temperatures high enough to challenge stability of fibrous insulation such as Kaowool. Need to assess effects on microstructural stability and thermophysical properties during irradiation and high-temperature exposure in impure helium. Limited commercial information available for conditions of interest.
Reactor Pressure Vessel	Crack initiation and subcritical crack growth	H	L	FOM—RPV integrity—9 Cr–1 Mo steel (grade 91) must be assessed for phenomena due to transients and operationally induced—thermal loading, pressure loading, residual stress, existing flaws (degradation of welds, cyclic loading, low-cycle fatigue). There is a limited database from fossil energy applications at these temperatures. Low-cycle fatigue data in air, vacuum, and sodium (ANL unpublished data) at >482°C show life is longest in sodium, followed by vacuum and air. Aging in helium (depending on impurities) will most likely be greater than in air. Aging in impure helium may perhaps depend on impurity type and content.
Reactor Pressure Vessel	Compromise of emissivity due to loss of desired surface layer properties	H	L	FOM—RPV integrity—peak fuel temperature—To ensure passive safety, high emissivity of the RPV is required to limit core temperatures—must maintain high emissivities on both inside and outside surfaces. Formation and control of surface layers must be considered under both helium and air environments. There are limited studies on SS and on Alloy 508 that show potential for maintaining high emissivity. Some studies currently being conducted on emissivity but NOT on materials of concern.
Reactor Pressure Vessel	Field fabrication process control	H	L	FOM—RPV integrity—Fabrication issues must address field fabrication because of vessel size [including welding, postweld heat treatment (PWHT), section thickness (especially with 9 Cr–1 Mo steel) and preservice inspection]. Fossil energy experience indicates that caution needs to be taken. On-site nuclear vessel fabrication is unprecedented.
Reactor Pressure Vessel	Property control in heavy sections	H	L	FOM—RPV integrity—Heavy-section properties are difficult to obtain because of hardenability issues. Adequate large ingot metallurgy technology does not exist for 9 Cr–1 Mo steel. Maintaining fracture toughness, microstructural control, and mechanical properties in through-thickness of heavy sections, 9 Cr materials must be maintained. (Concerns in utilities regarding P91, >3-in. piping heat treatment a challenge.) Excess deformation was listed because of the emphasis on minimizing changes in

Table 9 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
				core geometry.
				Very limited data; not much over 3 to 4 in. thickness. Few data available for specimens from 300-mm-thick forgings show thick section properties lower than thin section.
Reactor Pressure Vessel	Thermal aging (long term)	H	M	FOM—RPV integrity—Uncertainty in properties of 9 Cr–1 Mo steel (grade 91), especially degradation and aging of base metals and welds for a critical component like the RPV must be addressed for 60-year lifetimes. Although it was not discussed in our meeting, Type IV cracking has been observed in operating fossil plants at 545°C after 20,000 h. Although unlikely, is Type IV cracking at NGNP operating temperatures possible for very long time (60 years) exposure?
				It is assumed that grade 91 is the prime candidate for NGNP, and no back up material is considered in this report for designs without active cooling. This is beyond experience base for conditions of interest, extensive fossil energy experience and code usage, though significant aging data exist at high temperatures (>500°C). Need is for long-term aging data at NGNP relevant temperatures.
Valves	Isolation valve failure	H	L	FOM—primary system pressure boundary integrity—Isolation valve failure (includes categories such as self-welding, galling, seizing) is possible. Concerns about isolation valves are similar to "breach to secondary" issues on IHX since they would provide barriers to secondary heat transport system.
				Information possibly available from previously constructed HTGRs, but relevance needs to be assessed. State of knowledge about helium-leak-tightness in large valves is unknown.
Valves	Valve failure (general)	H	L	FOM—primary system pressure boundary integrity—Concerns about a variety of valve failure mechanisms that will be design-dependent (includes categories such as self-welding, galling, seizing) will need to be assessed once design-specific details are available. Helium-tribology issues must be considered. Allowable identified and unidentified coolant leakage must be established.
				Information available from previously constructed HTGRs, but relevance needs to be assessed.

[a]From Table 6 in Ref. 3.
[b]H, M, or L (high, medium, or low).

Of all the NGNP high-temperature metallic components, the one most likely to be heavily challenged is the IHX. That is because the NGNP requires the use of a secondary loop process heat application and perhaps for electric power generation as well. The IHX's thin internal sections must be able to withstand the stresses associated with thermal loading and pressure differences between the primary and secondary loops, which may be quite substantial. Additionally, since these sections must operate at the full exit temperature of the reactor, metallurgical stability and environmental resistance of the materials in anticipated impure helium coolant environments must be adequate for the anticipated lifetimes. Several IHX materials-related phenomena were rated as an H importance for potentially contributing to fission product release at the site boundary and an L level of knowledge with which to assess their contribution to such a release. These included crack initiation and propagation due to creep crack growth, creep, creep-fatigue, and aging; the lack of experience with primary boundary design methodology for new IHX structures; manufacturing phenomena for new designs (including joining issues); and the ability to inspect and test new IHX designs.

Specific issues must be addressed for RPVs that are too large for shop fabrication and transportation. Validated procedures for on-site welding, PWHT, and inspections will be required for the materials of construction. For vessels using materials other than those typical of LWR construction required for operation at higher temperatures, confirmation of the ability to be fabricated (especially effects of forging size and the ability to be welded), and data on the effects of radiation will be needed. Three materials-related phenomena related to the RPV fabrication and operation were rated as an H importance for potentially contributing to fission product release at the site boundary and a L level of knowledge with which to assess their contribution to such a release, particularly for 9 Cr–MoV steels capable of higher temperature operation than LWR vessel steels. These included crack initiation and subcritical crack growth, field fabrication process control, and property control in heavy sections.

For the RPV, long-term thermal aging and a possible compromise of surface emissivity were identified as significant phenomena. Since the ability to reject heat passively and adequately during certain transients in the NGNP is dependent upon transmitting decay heat from the core and radiating it from the exterior of RPV, it is critical that emissivity of the various potential candidate materials for the RPV and core barrel remain sufficiently high over their lifetimes. Depending on the emissivity of the selected materials, it may be necessary to qualify and incorporate high emissivity, durable coatings on the surfaces of these components. Two materials-related phenomena for the RPV and core barrel emissivity were rated as an H importance for potentially contributing to fission product release at the site boundary and a level of knowledge with which to assess their contribution to such a release. These phenomena (emissivity degradations caused by loss of desired surface layer properties) were rated as H because of their potential impact on passive heat rejection ability.

Aging fatigue and environmental degradation of insulation with a possibility for plugging coolant channels was listed as a possible concern for fuel temperatures. The phenomena of high-temperature performance of the insulation during an accident were assessed by the panel. Large-scale core restraint phenomena were noted as well. Phenomena associated with control rod composites and structural designs indicate the need for approved design methods, validation tests, and design standards.

Other high-level issues for high-temperature metallic components that will require evaluation include the following:

- inelastic behavior for various materials, time-at-temperature conditions for very high temperature structures (e.g., creep, fatigue, creep-fatigue, etc.);
- adequacy and applicability of current ASME Boiler and Pressure Vessel (B&PV) Code allowables with respect to service times and temperatures for operational stresses;
- adequacy and applicability of the current state of high-temperature design methodology (e.g., constitutive models, complex loading, failure criteria, flaw assessment methods, etc.);

- effects of product form and section thickness;
- joining methods including welding, diffusion bonding, and issues associated with dissimilar materials in structural components;
- effects of irradiation on materials strength, ductility, and toughness;
- degradation mechanisms and ability to be inspected;
- oxidation, carburization, decarburization, and nitriding of metallic components in impure helium and helium-nitrogen;
- effects of short- and long-term operation on mechanical properties (e.g., tensile, fatigue, creep, creep-fatigue, ductility, toughness, etc.);
- high-velocity erosion/corrosion;
- rapid oxidation of graphite and C–C composites during air-ingress accidents;
- compatibility with heat-transfer media and reactants for hydrogen generation; and
- development and stability of surface layers on RPV and core barrel affecting emissivity.

Impurities in the helium and their long-term impacts on mechanical properties such as creep and the aging of components were identified. Overall corrosion aspects were noted in the PIRT review. Supporting data along with experimentally based constitutive models that are the foundation of the inelastic design analyses specifically required by ASME B&PV Section III, Division I, Subsection NH must be developed for the construction materials.

4.4 Graphite PIRT Panel

4.4.1 Phenomena identification and description

The GRAPH PIRT covered the qualification of nuclear-grade graphite and its material property characterization under various thermal and neutron-irradiation conditions. The phenomena include FP release from (or through) the graphite, degradation of thermal conductivity, structural properties, annealing, dust generation, and the aspects of creep and strain. Many of these property aspects serve as input into the ACTH PIRT analysis (conductivity being a prime example). Oxidation was also a concern, both in steady-state and in accident conditions, and the kinetics of that reaction and the associated phenomena were identified and evaluated. These important material characteristics provide the basis for safety margins in the design, as well as being important phenomenological aspects that impact accident scenarios and consequences.

4.4.2 Ranking rationale

The GRAPH PIRT panel identified three FOM levels. The top level was the requirement to maintain dose levels to the public within the regulatory requirements. The second level consisted of a set of three sublevels of FOMs which can affect the top-level FOM. These FOMs, in turn, are influenced by and through the third-level FOM (known as "component") and are listed below.

These levels and their respective FOMS are as follows:

- Level 1: Regulatory; dose at the site boundary due to radioactivity releases
- Level 2: System
  1. Increased coolant radioactivity
  2. Challenge to primary system integrity
  3. Ability (or degraded ability) to obtain and keep cold shutdown

- Level 3: Component
  1. Ability to maintain passive heat transfer
  2. 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
  7. Minimize radioactivity in the coolant

The panel identified and analyzed the impact that phenomena had on the FOMs and determined which of the component FOMs applied to these phenomena.

4.4.3 Panel analysis

The graphite single crystal is highly anisotropic due to the nature of its bonding (strong covalent bonds between the carbon atoms in the basal plane and weak van der Waals bonds between the basal planes). This anisotropy is transferred to the filler coke particles and also to the crystalline regions converted by graphitization in the binder phase. Thus, the mechanical and physical properties of graphite vary within a billet due to texture introduced during forming and thermal processing (graphitization). Moreover, there is a statistical variability in the properties between billets within the same lots, between lots, and between batches due to variations on raw materials, formulations, and processing conditions. Therefore, it is necessary to develop a statistical database of the properties for a given graphite grade. The variations in chemical properties (chemical purity level) will have implications for chemical attack, degradation, and decommissioning. Probabilistic design approaches are best suited to capturing the variability of graphite.

A significant challenge related to graphite for HTGRs is that the previous graphite grade qualified for nuclear service in the United States, H-451, is no longer available. The precursors from which H-451 graphite was manufactured no longer exist; furthermore, the present understanding of graphite behavior is not sufficiently developed to enable the H-451 database to be extrapolated completely to nuclear graphite grades currently available. Hence, it will be necessary to qualify new grades of graphite for use in VHTRs and, in doing so, gain a more robust understanding of irradiated graphite behavior to ensure that new theories and models have a sound, in-depth, scientific basis. It will be necessary to qualify the new graphite(s) with regard to nonirradiated and neutron-irradiated performance. In reactor designs that impose large neutron-irradiation damage doses (i.e., beyond volume change turn-around), it may become necessary to replace core components and structures during the lifetime of the reactor. There is also a need for associated in-service inspection and assessment of the structural integrity of these structures. Thus, the designers and operators will require data and an understanding of the phenomena so that decisions can be made on replacement and service life.

The panel noted the inherent variability in the physical, mechanical, and thermal properties of unirradiated graphite within billets and lots and rated the associated phenomena as high importance. In addition, the effects of reactor environment (temperature, neutron irradiation, and chemical attack) on the physical properties must be characterized when the graphite is qualified. Significant work is required to bring the existing graphite codes and standards to an acceptable condition. The proposed Section III Division 2, Subsection CE of the ASME B&PV Code (Design Requirements for Graphite Core Supports) was issued for review and comment in 1992, but only limited action has been taken on this code since that time and it must be updated and adopted. During 2006, a Special Group was commissioned under Section III of the ASME B&PV Code Committee to develop it.

Table 10 contains the group of selected phenomena that the panel considered to be of particular significance with a combination of a high importance ranking and a low or moderate knowledge ranking. The FOM is provided in the rationale (a numbered level is provided here; refer back to the previous section on FOM used by the graphite panel). The reader is cautioned that merely selecting phenomena based on high importance and low or medium knowledge may not constitute a complete assessment of the situation. It is recommended that the reader refer to the panel's detailed assessment in Ref. 4.

The panel noted several significant phenomena (stress, creep, and coefficient of thermal expansion) related to graphite properties and material characterization of these properties as functions of temperatures and neutron irradiation. Stress due to differential thermal strain and differential neutron-irradiation-induced dimensional changes would very quickly cause fracture in the graphite components if it were not for the relief of stress due to neutron-irradiation-induced creep. Currently, there are no creep data for the graphite grades being considered for use in the NGNP. A new model for creep is needed which can account for the observed deviations from linearity of the creep strain rate with neutron dose. Differential thermal strains occur in graphite components due to temperature gradients and local variation in the coefficient of thermal expansion (CTE). The variations in the CTE are dependent upon the irradiation conditions (temperature and neutron dose) and the irradiation-induced creep. Irradiation-induced changes in CTE are understood to be related to changes in the oriented porosity in the graphite structure. There are insufficient data available for the effect of creep strain on CTE in graphite. Moreover, none of the available data are for the grades being considered for the NGNP. For these three phenomena, an H/L (high importance and low knowledge level) assignment was made.

Mechanical properties such as strength, toughness, and the effect of creep strain were also identified by the panel. The properties of the graphite are known to change with neutron irradiation, the extent of which is a function of the neutron dose, irradiation temperature, and irradiation-induced creep strain. Local differences in moduli, strength, and toughness due to neutron fluence and temperature gradients must be accounted for in the design. The importance of this phenomenon is thus ranked high. Although data exist for the effect of neutron dose and temperature on the mechanical properties of graphite, there are insufficient data on the effects of creep strain on the mechanical properties. Moreover, none of the available data are for the grades currently being considered for the NGNP (thus knowledge level is L).

Several graphite phenomena leading to a blocked fuel element coolant channel (or in a blockage to reactivity control element insertion) were identified by the panel. Significant uncertainty exists as to the stress state of any graphite component in the core. Moreover, the strength of the components changes with neutron dose, temperature, and creep strain. The combination of these factors makes the probability of local failure, graphite spalling, and possible blockage of a fuel-element coolant channel difficult to determine. Consequently, the panel rated this phenomenon's importance as an H. Although the changes in properties of graphite have been studied for many years, there are still data gaps that make whole core modeling very difficult (e.g., effect of creep strain on properties). Moreover, data on the grades being considered for NGNP are not available. Therefore, the panel rated the knowledge level for this phenomenon as L.

Table 10. Significant GRAPH phenomena (high importance and low or medium knowledge rankings)[a]

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
Graphite	Irradiation-induced creep (irradiation-induced dimensional change under stress) {Could potentially reduce significantly internal stress}	H	L	FOM—ability to maintain passive heat transfer; ability to control reactivity; thermal protection of adjacent components; shielding of adjacent components; maintain coolant flow path; prevent excessive mechanical load on the fuel; minimize activity in the coolant. – Required for graphite finite-element method (FEM) stress analysis, acts to reduce stress. It is essential that irradiation creep is better understood; mechanistic understanding essential. There are interaction affects with the CTE and may be dimensional change and modulus. New models are needed along with data on new graphites.
Graphite	Irradiation-induced change in CTE, including the effects of creep strain	H	L	FOM—ability to control reactivity; thermal protection of adjacent components; prevent excessive mechanical load on the fuel; minimize activity in the coolant. – Essential input into irradiated graphite component stress analysis; also affected by irradiation creep. Extensive database, some microstructural/mechanistic studies required.
Graphite	Irradiation-induced changes in mechanical properties (strength, toughness), including the effect of creep strain (stress) {Tensile, bend, compression, shear (multiaxial), stress-strain relationship, fracture, and fatigue strength}	H	L	FOM—ability to control reactivity; thermal protection of adjacent components; prevent excessive mechanical load on the fuel; minimize activity in the coolant. – Essential input into irradiated graphite component stress analysis. Extensive database, some microstructural/mechanistic studies required. Better understanding of fracture process required.
Graphite	Statistical variation of nonirradiated properties {Variability in properties (textural and statistical); isotropic. Probabilistic approach use is prudent. Purity level; implications for chemical attack, degradation, decommissioning}	H	M	FOM—ability to maintain passive heat transfer; ability to control reactivity; thermal protection of adjacent components; Shielding of adjacent components; maintain coolant flow path; prevent excessive mechanical load on the fuel; minimize activity in the coolant. – Graphite has a significant spread in properties; therefore, a statistical approach is essential. That is within block, block to block within the same batch, and batch to batch.

Table 10 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
				This has to be known and understood. Statistical methods need to be in agreement, improved upon, and validated. Standards need establishing.
Graphite	Consistency in graphite quality over the lifetime of the reactor fleet (for replacement, for example)	H	M	FOM—ability to maintain passive heat transfer; ability to control reactivity; thermal protection of adjacent components; shielding of adjacent components; maintain coolant flow path; prevent excessive mechanical load on the fuel; minimize activity in the coolant. – Raw materials and manufacturing techniques may change with resultant change in properties and irradiation behavior. While there is a general understanding of graphite behavior for similar types of graphite, research is required to enable a reasonable prediction of irradiated graphite behavior to be made from knowledge of the microstructure of unirradiated graphite, thus reducing the need for large databases which may take many years to carry out.
Graphite	Graphite contains inherent flaws {Need methods for flaw evaluation}	M*	M	FOM—ability to control reactivity; thermal protection of adjacent components; shielding of adjacent components; maintain coolant flow path; prevent excessive mechanical load on the fuel; minimize activity in the coolant. – Available techniques need further development, demonstration, and confirmation. New improved component NDE techniques are desirable. New improved NDE methods require developing.
Graphite	Irradiation-induced dimensional change {Largest source of internal stress}	H	M	FOM—ability to maintain passive heat transfer; ability to control reactivity; thermal protection of adjacent components; shielding of adjacent components; maintain coolant flow path; prevent excessive mechanical load on the fuel; minimize activity in the coolant. – Required for graphite FEM stress analysis, main driver for stresses. Data available or can be measured, but better mechanistic understanding desirable.

Table 10 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
Graphite	Irradiation-induced thermal conductivity change {Thermal conductivity lower than required by design basis for LBE heat removal due to (a) inadequate database to support design over component lifetime and (b) variations in characteristics of graphites from lot to lot; potential is to exceed fuel design temperatures during LBEs}	H	M	FOM—ability to maintain passive heat transfer; thermal protection of adjacent components; maintain coolant flow path; minimize activity in the coolant. – Important input to loss of coolant accidents and used to define temperatures for FEM irradiated graphite component stress analysis. Low fluence data available and understanding adequate. High fluence data and understanding required. Methodology for temperature dependence requires validation.
Graphite	Irradiation-induced changes in elastic constants, including the effects of creep strain	H	M	FOM—ability to control reactivity; thermal protection of adjacent components; prevent excessive mechanical load on the fuel; minimize activity in the coolant. – Essential for input into irradiated graphite FEM stress analysis. Data available or can be measured; better mechanistic understanding desirable. Concept of increase in modulus due to "pinning" needs further investigation.
Graphite	Tribology of graphite in (impure) helium environment	H	M	FOM—maintain coolant flow path. – Depends on design. Impacts seismic assessments. Whole-core modeling needs this data. Limited data available.
Graphite Component	Blockage of fuel element coolant channel—due to graphite failure, spalling {Debris generated from within the graphite core structures}	H	L	FOM—maintain coolant flow path. – Two mechanisms: (a) component failure due to internal or external component stresses and (b) component failure due to very high irradiation and severe degradation of the graphite. Generic graphite codes available for the prediction of internal stresses in irradiated graphite components; however, they require validation. There are also whole-core models for component interaction; however, these are reactor specific. These codes will also require validation.
Graphite Component	Blockage of coolant channel in reactivity control block due to graphite failure, spalling {Debris generated from nongraphite components within the RPV}	H	L	FOM—ability to control reactivity; thermal protection of adjacent components. – Significant uncertainty exists as to the stress state of any graphite component in the core. Moreover, the strength of the component changes with dose, temperature, and creep strain. The combination of these factors makes the

Table 10 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
				probability of local failure, graphite spalling, and possible blockage of a coolant channel in a reactivity control block difficult to determine. Generic graphite codes available for the prediction of internal stresses in irradiated graphite components; however, they require validation. There are also whole-core models for component interaction; however, these are reactor specific; these codes will also require validation.
Graphite Component	Blockage of reactivity control channel—due to graphite failure, spalling {Debris generated from within the graphite core structures}	H	L	FOM—ability to control reactivity. – Two mechanisms: (a) component failure due to internal or external component stress and (b) component failure due to very high irradiation and sever degradation of the graphite. Generic graphite codes available for the prediction of internal stresses in irradiated graphite components; however, they require validation. There are also whole-core models for component interaction; however, these are reactor specific; these codes will also require validation.
Graphite Component	Degradation of thermal conductivity {Has an implication for fuel temperature limit for loss-of-forced cooling accident}	H	M	FOM—ability to maintain passive heat transfer. – Important input to loss-of-coolant accidents and used to define temperature for FEM irradiated graphite component stress analysis. Low-fluence data available and understanding adequate. High-fluence data and understanding required. Methodology for temperature dependence requires validation.
Graphite Component	Blockage of fuel element coolant channel—channel distortion {Deformation from individual graphite blocks and block assemblies. There is a link to the metallic core support structure}	M*	M	FOM—maintain coolant flow path. – Individual graphite component dimensional changes are normally significant but relatively small. However, in damaged components, dimensional changes can become quite large. The accumulation of dimensional changes in an assembly of components can result in significant overall dimensional changes and kinking (i.e., in a column of graphite bricks). Generic graphite codes available for the prediction of deformations in irradiated graphite components; however, they require validation. There are also whole-core models for component interaction; however, these are reactor specific; these will also require validation.

Table 10 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
Graphite Component	Graphite temperatures {All graphite component life and transient calculations (structural integrity) require time-dependent and spatial predictions of graphite temperatures. Graphite temperatures for normal operation and transients are usually supplied to graphite specialists by thermal-hydraulics specialist. However, in some cases, gas temperatures and heat transfer coefficients are supplied, and the graphite specialists calculate the graphite temperatures from these}	H	M	FOM—ability to maintain passive heat transfer; ability to control reactivity; thermal protection of adjacent components; shielding of adjacent components; maintain coolant flow path; prevent excessive mechanical load on the fuel; minimize activity in the coolant. – All graphite component life and transient calculations (structural integrity) require time dependent and spatial predictions of graphite temperatures. Graphite temperatures for normal operation and transients are usually supplied to graphite specialists by thermal-hydraulics specialist. Although, in some cases, gas temperatures and heat transfer coefficients are supplied, and the graphite specialists calculate the graphite temperatures from these. Justification for the use (or not of EDT-equivalent DIDO temperatures) requires validation.
Graphite Component	Tribology of graphite in (impure) helium environment	H*	M	FOM—ability to control reactivity; thermal protection of adjacent components. – Depends on design. Impacts seismic assessments. Whole-core modeling needs these data. – Limited data available.

[a]From Tables 5 and 6 in Ref. 4.
[b]H, M, or L (high, medium, or low).
Text in parentheses { } in the Phenomena column is added to elaborate on the phenomena.
Asterisk (*) by ranking indicates some disagreement in panel ranking.

The mechanical and physical properties of nonirradiated graphite vary within a billet due to texture introduced during forming and thermal processing. Moreover, there is statistical variability in the properties between billets within the same lots, between lots, and between batches due to variations in raw materials, formulations, and processing conditions. Therefore, it is necessary to develop a statistical database of the properties for a given graphite grade. Variations in the chemical properties (chemical purity level) will have implications for chemical attack, degradation, and decommissioning. Although other nuclear graphites have been characterized and full databases developed, allowing an understanding to be developed of the textural variations, only limited data exist on the graphites proposed for the NGNP. Therefore, the panel rated this phenomenon as H/M.

Neutron-irradiation-induced changes in dimensions, thermal conductivity, and elastic constants were noted by the panel. The amount of irradiation-induced dimensional change is a function of the neutron dose and irradiation temperature and is the largest source of internal stress. Consequently, gradients in temperature or neutron dose will introduce differential dimensional changes (strains). Thermal conductivity is also reduced by displacement damage caused by neutron irradiation. At very high irradiation doses, thermal conductivity decreases further, at an increased rate due to porosity generation due to large crystal dimensional changes. Values of thermal conductivity under all core conditions are therefore subject to some uncertainty. Irradiation-induced thermal conductivity changes have been researched for many years, and several conductivity change models have been proposed. However, there is a paucity of data for the conductivity changes of the graphites proposed for the NGNP. Neutron-irradiation induces changes in the elastic constants of graphite. Although the understanding of irradiation-induced moduli changes is well developed, there are no direct microstructural observations or sufficiently well-developed models of these mechanisms. For these three phenomena, the importance was rated H and the knowledge rating was M.

The need for consistency in fabricated graphite quality over the lifetime of a reactor fleet was noted by the panel. Graphite is manufactured from cokes and pitches derived from naturally occurring organic sources such as oil and coal (in the form of coal tar pitch). These sources are subject to geological variations and depletion, requiring the substitution of alternate sources. Therefore, the consistency of graphite quality and properties over the lifetime of a reactor, or a reactor fleet (for replacement, for example), is of concern. The panel ranked the importance of this phenomenon as H. The panel's understanding of this phenomenon is sufficient in that generic specifications should be able to be drawn up. However, this has not been proven, especially due to the lack of neutron-irradiated properties data. The panel assessed the knowledge base for this phenomenon as M.

The abrasion of graphite blocks on one another, or abrasion of fuel pebbles on the graphite moderator blocks, could produce graphite dust. Graphite is a lubricious material. Studies are needed to assess the effect of the helium environment on the friction and wear behavior of graphite. The possibility that fuel balls can "stick" together and cause a fuel flow blockage must be explored, although German pebble bed experience was positive in this regard (i.e., no significant blockages). The consequences of dust generation (possible fission product transport mechanism) and possible fuel ball interactions resulted in the panel ranking the importance of this phenomenon as H. Limited literature exists on this subject, mostly from the past German program. Consequently, the panel ranked the knowledge level as M.

Significant uncertainty exists as to the stress state of any graphite component in the core. The strength of the components changes with neutron dose, temperature, and creep strain. The combination of these factors makes the probability of local failure, graphite spalling, and possible blockage of a reactivity control channel in a reactivity control block difficult to determine. Consequently, the panel rated this phenomenon's importance as H. Although the changes in properties of graphite have been studied for many years, there are still data gaps that make the modeling of the whole core very difficult (e.g., effect of creep strain on properties). Moreover, data on the grades being considered for NGNP are not available. The NGNP designs are known to be capable of safe shutdown without control rod entry. Therefore, the panel rated the knowledge base for this phenomenon as M.

Theoretical models for the effects of neutron damage on the properties of graphite have been developed. However, these models need modification for the new graphites and will need to be extended to higher temperatures and/or higher neutron doses. Verification and validation of theoretical models can only come through the generation of experimental data on the effect of neutron irradiation on properties. Experimental data to fill the data gaps must be generated in a technology development program. The biggest gaps that have been identified are related to predicting the buildup in stress in graphite core components. Uncertainties in the temperature and neutron dose received by a component; the severity of temperature and neutron dose gradients in a components, the rate of dimensional change in the specific graphite used in a given design, the extent to which stresses are relieved by neutron-irradiation-induced creep, and the extent of changes in key physical properties such as elastic moduli, thermal conductivity, coefficient of thermal expansion, etc., are all compounded to make the prediction of component stress levels, and, hence, decisions regarding components lifetime and replacement schedules, very imprecise.

That said, the behavior of graphite in reactor environments in the 60 plus years since the first graphite reactors went into service has been extensively studied, and the current knowledge base is well developed. However, there is also no question that graphite for the NGNP will have to undergo a qualification process to obtain the required statistical data. Although data is lacking for the specific grades being considered for Generation IV concepts such as the NGNP, it is fully expected that the behavior of these graphites will conform to the recognized trends for near-isotropic nuclear graphite. Thus, much of the data needed is confirmatory in nature. Theories that can explain graphite behavior have been postulated and in many cases shown to well represent experimental data. However, these theories need to be tested against data for the new graphites and extended to the NGNP's higher neutron doses and temperatures. It is anticipated that current and planned graphite irradiation experiments will provide the data needed to validate many of the currently accepted models, as well as provide the needed data for design confirmation.

4.5 Process Heat and Hydrogen Co-Generation Production PIRT (PHHP) Panel

4.5.1 Phenomena identification and description

This PIRT considered a range of process chemical releases and consequences with respect as to how these may be a precursor to an external event at the reactor site. Chemical releases including hydrogen, oxygen, and toxic gases were considered. Additional aspects identified in the PIRT are temperature transients or a loss of heat sink. Both of these create thermal transients that can possibly feed back and influence transient or accident behavior at the nuclear plant. A heat exchanger failure and its various effects on the response of the nuclear plant were examined by the PIRT panel.

With respect to the VHTR, possible transients that could result in a dose or release pathway to the chemical process side of the plant were considered. This could be tritium or perhaps some fission product gases or aerosols.

4.5.2 The PHHP panel developed the evaluation using the following stepwise strategy

1. The types of accident events that were possible were identified and the qualitative result or direct consequence (challenge to the NGNP) of that event was estimated.
2. The next step was to examine the phenomena that controlled the severity of the potential impact on the NGNP. The characteristics of released materials, conditions associated with the release, magnitude of the thermal event, and potential timing all were considered in defining the magnitude of the potential threat to the NGNP.
3. The final step was to evaluate the potential impact on the NGNP (with an emphasis on safety-related or important-to-safety aspects) resulting from that event.

High (H), medium (M), or low (L) importance ratings were assigned according to the following criteria.

- If the material release or thermal event could potentially affect the likelihood or severity of core damage (meaning fuel integrity or core structural integrity), then it was considered high importance.
- If the event would impact operations or contribute to other safety-related events but not strongly impact the severity of an accident, then it was medium importance.
- If the event was considered to primarily impact operations but have limited effects on the reactor or workers, then it was low importance.

The knowledge base estimate was done on expert opinions regarding the status of the tools and data available for quantifying these accident sequences and consequences. If the tools and database were considered to be adequate and currently available, a high (H) rating was given. Incomplete tools or information resulted in a medium (M) rating. If significant R&D would be required to establish a basis, then a low (L) rating was given.

4.5.3 Panel analysis

The major phenomena of safety significance to the reactor that were considered by the panel are (1) chemical releases, (2) thermal events on the chemical-plant process side, (3) failures in the intermediate heat-transport system, and (4) reactor events that could provide a feedback path. This PHHP PIRT was conducted to identify the events and phenomena that must be considered in evaluating the safety of the NGNP.

Significant phenomena (those with H importance and L or M knowledge level) are highlighted in Ref. 5. Table 11 summarizes the phenomena and associated events that were judged to have high importance along with medium or low knowledge level. Selecting phenomena based on high importance and low or medium knowledge may not constitute a complete assessment of the situation. It is recommended that the reader refer to the panel's detailed assessment in Ref. 5

The hazards associated with various chemicals and methods to minimize risks from those hazards are well understood within the chemical industry and by the chemical plant regulators. This provides much but not all of the information that will be required to define conditions (separation distance, relative elevation, berms, other mitigation features) to ensure reactor safety when the reactor is coupled to a chemical plant. There is also some experience in the nuclear industry associated with various nuclear plants in several countries that have produced steam for industrial applications. In all cases, the specific characteristics of the chemical plant, the proposed site layout, and the maximum associated stored inventories of chemicals provide the starting point for the safety assessments.

Table 11. Significant PHHP phenomena (high importance and low or medium knowledge ranking)[a]

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
Primary System Components; Structures, Systems, and Components (SSCs)	Fuel and primary system corrosion [process heat exchanger (PHX) failure]	H	M	FOM—damage or impairment of SSCs—PHX failure would precipitate problems in IHX; more critical. It is a unique threat to IHX; ultimate impact would be on IHX. Novel PHX designs at this point do not yet exist; no experience base.
Primary System Components, SSCs	Blow-down effects, large mass transfer; pressurization of either secondary or primary side (IHX failures) {Fluid hammer. Thermal and concentration gradients can work against the D/P such that chemicals can diffuse toward the IHX}	H	M	FOM—damage or wear of SSCs—Failure modes are equally important in both IHX and PHX. IHX is important because it is a boundary between the core and the secondary loop; small helium purge of a hot core. Small leaks more worrisome. Consensus: If salt intermediate loop, then no massive pressurization. Have models available that can handle these problems.
Primary System Components, SSCs	Loss of main heat sink (hydrodynamic loading on IHX; cutting margins down by increasing D/P over IHX; decrease operating life of IHX) (loss of intermediate fluids) {Rapid pulse cooling of reactor during depressurization of intermediate loop and IHX. Very rapid event. Self-closing valves act faster than I&C system}	H	M	FOM—damage, wear, or impairment of SSCs—Loss of heat sink with all the blow-down effects. Potential for high probability in plant lifetime. Perhaps could occur in reactor lifetime? Uncertainty about IHX design. Good tools to work with currently, but design uncertainty exists.
Primary System Components, SSCs, TRISO Fuel Coatings	Reactivity spike due to neutron thermalization (mass addition to reactor: hydrogenous materials) {Power spike in fuel grains, could lead to TRISO-failure with prolonged high temperature}	H	M	FOM—damage, wear, or impairment of SSCs (TRISO layers; fission product confinement)—The importance of hydrogenous mass additions was considered high because of the reactivity potential with possible power increases leading to a more severe thermal scenario. The neutronic and thermal effects of hydrogenous material additions can be readily analyzed with available tools. The knowledge base was designated M because the configurations, flow paths, and pressure characteristics are not well defined at this point in time.

Table 11 (continued)

System or component	Phenomena	Importance[b]	Knowledge level[b]	Rationale
Primary System Components, SSCs, TRISO Fuel Coating	Chemical attack of TRISO layers and graphite (mass addition to reactor: hydrogenous materials) {Steam and graphite react; TRISO. More concerned with gases produced in core by the steam, rather than the chemical attack on fuel. Pressure relief valve would open in primary loop releasing hydrogen into confinement}	H	M	FOM—damage, wear, or impairment of SSCs (TRISO layers; fission product confinement)—Accidentally dumped water into core in AVR; had to boil water off, no chemical attack. Graphite attack and reformer gas production. Hydrogenous mass additions could lead to thermal and pressure transients and corrosion issues if the introduction were severe. Fission product panel should be aware of this. The knowledge base was designated M because the configurations, flow paths, and pressure characteristics are not well defined at this point in time.
Safety System Components, SSCs	Allowable concentrations (oxygen releases) {What oxygen levels cause damage?}	H	M	FOM—structures, systems, and components (SSCs)—High, partially over concerns of both accident and long-term elevated levels; is there a chance of locally high concentrations in NGNP that are higher than designed for? Are we changing chemical properties of equipment, I&C, and people if locally high O_2 concentrations? Importance of plume issue; want to know where the O_2 goes; worst case is low temperature, release. Small inventory but possibility of plume is important. – Question is really one of what flammable material is present and what are ignition sources? – The tools and knowledge are available; models do not have any new physics or considerations. – Such extensive experience working with O_2 in industry; understand effects on some equipment well.
SSCs	Spontaneous combustion (oxygen releases) {What levels cause spontaneous combustion?}	H	M	FOM—SSCs—May not easily disperse if released in large quantities. Importance if plume issue; want to know where the O_2 goes; worst case low-temperature release. Small inventory but possibility if plume important. – Question is really one of what flammable material is present and what are ignition sources? – The tools and knowledge are available; models do not have any new physics or considerations.

[a]From Table B-1 in Ref. 5.
[b]H, M, or L (high, medium, or low).
Text in parentheses { } in the Phenomena column is added to elaborate on the phenomena.

4.5.3.1 *Chemical release impacts on NGNP*

The key analysis results for the PHHP PIRT involving the collocated chemical plant are listed below. The following phenomena cover the aspects of an external event caused by the nearby chemical process plant. This includes releases of hydrogen, oxygen, and other heavy gasses. The differences between chemical safety philosophy and nuclear safety philosophy were documented by the panel. Particular phenomena were identified for their role in external event challenges to the reactor. The applicability of existing models and databases to safety analyses of coupled systems within the NGNP technology envelope were used in the knowledge ranking. The assessments were difficult in that the designs of the process heat plant and heat transport loop have not yet been selected.

NGNP vs. a commercial high-temperature reactor dedicated to process heat

The PIRT panel examined safety issues associated with the NGNP and a commercial plant. For the NGNP, only a small fraction of the heat is expected to be used to produce hydrogen or other chemicals, with most of the heat used to produce electricity. In contrast, for a commercial high-temperature reactor dedicated to a process heat application, all of the heat may be used for production of hydrogen or chemicals. Because the total chemical inventories determine the potential hazard to the nuclear plant from a chemical plant, the hazards of a small chemical plant associated with the NGNP may be significantly less than for a commercial high-temperature reactor coupled to a large chemical complex. The detailed safety assessments must consider actual inventories. There is one further complication associated with the NGNP. As a pilot plant facility, there may be multiple generations of hydrogen production and other chemical technologies that are tested; thus, one must either envelope the safety implications of the different technologies to be tested or update the safety analysis with time.

Chemical plant safety, regulatory strategy, and site layout

The safety philosophy for most chemical plants is fundamentally different from the safety philosophy associated with nuclear power plants. For many hazards, such as a hydrogen leak, the safety strategy is dilution with air to below the concentration of hydrogen that can burn in air. For example, a small amount of hydrogen in an enclosed room is an explosion hazard. However, a large release of hydrogen to the environment is a relatively small hazard when outdoors. As a consequence, most chemical plants are built outdoors to allow rapid dilution of chemicals with air under accident conditions. The reverse strategy is used for nuclear plants, where the goal is to contain radionuclides since their hazard does not disappear if diluted with air. The chemical plant safety strategy implies that the primary safety "devices" to prevent small events from becoming major accidents include outdoor construction (no containment structure), controlling the size of the chemical inventories, the site layout, and the separation distances between various process facilities and storage facilities. This different safety strategy must be recognized and understood when considering safety challenges to a nuclear reactor from coupled chemical plants.

Hydrogen

Accidental releases of hydrogen from a hydrogen production facility are unlikely to be a major hazard for the nuclear plant assuming some minimum separation distances. This conclusion is based on several factors: (1) if hydrogen is released, it rapidly rises and diffuses, thus making it very difficult to create conditions for a large explosion and (2) a hydrogen burn does not produce high thermal fluxes that can damage nearby equipment. In addition to laboratory and theoretical analysis of hydrogen accidents, there is a massive knowledge base in the chemical industry with hydrogen accidents and thus a large experimental basis to quantify this hazard based on real-world experience.

Heavy gases

Many chemical plants under accident conditions can produce heavy ground-hugging gases such as cold oxygen, corrosive gases, and toxic gases. Industrial experience shows that such accidents can have major off-site consequences because of the ease of transport from the chemical plant to off-site locations. If the chemical plant or the stored inventories of chemicals are capable of releasing large quantities of heavy gases under accident conditions, this safety challenge requires careful attention. Oxygen presents a special concern. Most proposed nuclear hydrogen processes convert water into hydrogen and oxygen; thus, oxygen is the primary by-product. Oxygen has some unique capabilities to generate fires. Equally important, these will be the first facilities that may release very large quantities of oxygen to the atmosphere as part of normal operations. There is a lack of experience. The phenomena associated with plume modeling and the effects of such plumes on the nuclear plant safety-related structures, systems, and components are of high importance.

4.5.3.2 Heat exchanger failure

The second major class of safety challenges with high importance is associated with the failure of the intermediate heat transport loop that moves heat from the reactor to the chemical plant. Several different heat transport media are being considered including helium, helium-nitrogen mixtures, liquid salt mixtures, and high-temperature steam. High-temperature steam is required as a process chemical for some processes, such as the production of hydrogen using high-temperature electrolysis, thus steam could be the intermediate heat transport fluid. For gas-phase intermediate heat transport systems, there are several specific phenomena of high importance. These safety challenges define a second group of phenomena with high safety importance and are described below.

Blow-down of the intermediate heat transport loop

If the pressure boundary of the intermediate heat transport system fails, the blow-down will accelerate fluid flow through the primary heat exchangers. Depending upon the failure location, this may result in accelerated fluid flow of the cold heat-transport fluid through the IHX during the blow-down and result in overcooling the reactor coolant because of enhanced heat transfer in the primary heat exchanger. After blow-down, there will be a loss of the heat sink.

Leak into the reactor primary system

The total gas inventory in the intermediate loop may be significantly larger than the total inventory of gas in the reactor primary system. A large or small leak from the intermediate heat transport loop into the reactor in accident scenarios where the primary system depressurizes could add large inventories of gas to the reactor, providing a sweep gas to move fission products from the reactor core.

Chemical additions to the reactor core

If steam or other reactive gases from the intermediate heat transport loop enter the reactor because of a heat exchanger failure, there is the potential for fuel damage—particularly given the much higher temperatures proposed for some applications of high-temperature reactors.

Hot fluids

If the heat transfer fluid escapes into the reactor building, the high temperatures can cause significant damage.

5. PHENOMENA IDENTIFIED AND EVALUATED BY MULTIPLE PANELS

This section identifies similar phenomena that were evaluated somewhat independently by each PIRT panel. As such, these phenomena could generally be considered as cross-cutting. Not surprisingly, each panel had its own views regarding phenomena importance and knowledge level. Some panels are in general agreement for some cases, but in other cases, vastly different ratings for the same or similar phenomena were obtained. It is difficult to know if the exact same phenomena and its different aspects were actually being discussed by two or more panels because very few inter-panel discussions were held during the PIRT process. The phenomena (or in some cases it may be an issue identified by the panel) are paraphrased from the individual reports below. The panels that identified them (in any aspect—see individual PIRT reports) are shown in Table 12. Some of these phenomena were not necessarily found to be of high importance and/or of low or medium knowledge level.

The following is a general discussion of the items identified in Table 12 and a discussion of what was noted by the respective panels. A more complete picture can be obtained by referring to the individual PIRT panel reports and comparing the panel reports.

Temperatures and fission product transport phenomena

During the PIRT process, the linkage and cross-cutting aspects of the ACTH PIRT and the FPT PIRT assessments were acknowledged by both panels during the process. Table 6 of the FPT PIRT (Ref. 2) lists factors that impact major phenomena along the fission product transport paths from the fuel particle surface. It is noteworthy that temperature (of mutual concern to thermal fluids) is listed as at least one factor in all of the phenomena. There are other cross-cutting phenomena quoted in that table, such as lift-off and transport of aerosols and leakages through the confinement building

Bypass and core flows

For normal operation, the ACTH PIRT (Ref. 1) identified several flow phenomena affecting normal core bypass and core flow (see normal operation–rated high importance and medium to low knowledge). The phenomena identified were coolant properties, bypass flows resulting both from gaps between blocks and from gaps between the reflector and core barrel, and overall core flow distribution. This aspect applies for the prismatic and for the pebble bed (the core wall has interface effects on bypass flow). These phenomena's obvious importance is that they affect power-to-flow ratios in the core and thus impact fuel temperature and fuel performance.

Similarly the FPT PIRT (Ref. 2) identified graphite geometry (also, of course, covered by the graphite PIRT). Gas flow paths prior to and during accidents were rated by the FPT as high importance as well as a high knowledge level. In addition, temperature and pressure distribution phenomena needed for accident modeling were noted as important phenomena.

Table 12. Phenomena (that are similar) identified by multiple panels

Phenomena	ACTH	FPT	HTMAT	GRAPH	PHHP
Temperature and fission product transport phenomena	x	x			
Bypass and core flows	x	x			
Graphite dust and aerosols	x	x			
Reactivity insertion and fuel failure	x	x			
Fuel performance modeling	x	x			
Decay heat and distribution	x	x			
Graphite temperature profiles	x			x	
Graphite thermal conductivity	x			x	
Coolant flow	x			x	
Plenum structural collapse	x			x	
RPV and RCCS emissivity	x		x		
RCCS fouling	x		x		
Upper head insulation	x		x		
Graphite oxidation	x	x		x	
Insulation failures			x	x	
Heat exchanger failure/cyclic loading			x		x
Chemical attack (molten salt as example) to core	x	x		x	x

Graphite dust and aerosols

In the D-LOFC analysis, the ACTH PIRT (Ref. 1) identified the phenomena of dust suspension as high importance, medium knowledge level (H/M) phenomena. Similarly the FPT PIRT (Ref. 2) identified FP plate-out and dust distribution under normal operation as an H/M phenomenon. Many of the phenomena reported and evaluated in the FPT PIRT are all phenomena that play a role and impact the plate-out loading in the primary system.

The ACTH identified the phenomena of cavity filtering, aerosol transport, duct exchange flow, and molecular diffusion in D-LOFC and air ingress events. The FPT characterized some similar phenomena such as radiolysis in confinements, filtration, leak paths, and release rates in the confinement.

Reactivity insertion and fuel failure

The ACTH examined reactivity phenomena associated with ATWS events (such as reactivity insertions, reactivity feedback coefficients). The FPT identified phenomena of fuel damage from a reactivity insertion accident (characterized as an intense pulse on fuel). These cross-cutting phenomena will require a coupled neutron kinetics/thermal hydraulics model to supply the profile conditions to a fission product transport model or time-at-temperature limit that is applicable for modeling rapid fuel failure.

Fuel performance modeling

The ACTH PIRT (Ref. 1) identified fuel performance modeling broadly as an H/M phenomenon (listed under normal operation). In addition, heat-up accident fuel performance modeling (for D-LOFC) was noted to be a crucial factor (H/M rating). A more detailed treatment of the phenomena that make up a fuel performance model is covered in the FPT PIRT (Ref. 2) and deals with the various transport paths and a number of phenomena

Decay heat and distribution

The ACTH PIRT (Ref. 1) identified decay heat and its distribution over time for a D-LOFC since it affects peak fuel temperature. This is obviously linked to the neutronics aspects. The FPT PIRT (Ref. 2) also identified decay heat and transient power level as phenomena (the rationale was that it is an energy source).

Graphite temperature profiles

The GRAPH PIRT (Ref. 4) identified time and spatially dependent component temperatures as a needed input from the thermal analysis. These data are fed into the component life and transient calculations to confirm structural integrity. Obviously, the tools for this calculation are the thermal codes whose validity is driven by the appropriate treatment for the phenomena identified in the ACTH PIRT (Ref. 1).

Graphite thermal conductivity

As noted in the GRAPH PIRT (Ref. 4), neutron irradiation degrades the thermal conductivity and annealing of graphite during accident improves it. Statistical variations of this property and other intrinsic core material characteristics (such as specific heat) as a function of temperature and fluence levels are needed. A thorough understanding of these aspects were identified (all part of necessary graphite characterization) by the GRAPH PIRT (Ref. 4). Mention is made that these have the potential to threaten allowable fuel design temperatures during licensing basis events. Thus, good thermal conduction models [rated as H/M in the ACTH PIRT (Ref. 1)] are needed that account for such variations since thermal conduction plays an important role in passive safety.

Coolant flow

The GRAPH PIRT (Ref. 4) identified "Blockage of Reflector Block Coolant Channel" as a phenomenon with a number of different causes related to materials, graphite performance, distortion, and strain. These aspects are linked to ACTH identification of bypass and coolant path flow (normal operation). There are dimensional changes in graphite with neutron irradiation and temperatures over a variety of core conditions. This results in stress and strain profiles throughout the core and reflector. Increased bypass coolant flow channels due to channel distortion or cracking in blocks is also noted in the GRAPH PIRT (Ref. 4).

Plenum structural collapse

Plenum collapse due to a wide variety of phenomena was identified and evaluated by the GRAPH PIRT panel. The ACTH PIRT (Ref. 1) also mentions a reactivity insertion and evaluation of phenomena as a result of a core support failure.

RPV and RCCS emissivity

RPV emissivity and associated heat transfer phenomena were identified as linked to fuel temperatures and heat transfer by the ACTH (rated as H/M). The ACTH panel also identified vessel emissivity as an important factor. The HTMAT PIRT (Ref. 3) identified both the vessel and core barrel emissivity and identified concerns related to surface coating layers and compromise of surface layer properties.

The phenomenon of RCCS emissivity was identified by HTMAT PIRT (Ref. 3) as a H/L phenomenon. Similarly, the ACTH PIRT (Ref. 1) also identified RCCS panel emissivity in the general LOFC accident as an important phenomenon. There is no question that emissivities of both RPV and the RCCS are important cross-cutting phenomena due to their importance to passive safety.

RCCS fouling

Both ACTH (Ref. 1) and HTMAT (Ref. 3) PIRTs identified the phenomena of RCCS coolant fouling. However, the ACTH panel rated it as an (H/M), and the HTMAT panel rated it an (L/H).

Upper head insulation

The phenomena associated with heating the upper plenum and insulation in top head was identified by the ACTH PIRT (Ref. 1), and concern was expressed about the design of this region. The phenomena [plumes and radiant heat transfer, rated (H, M)] were of primary concern during P-LOFC cases where the heat rises to the top plenum area. The HTMAT PIRT (Ref. 3) also identified insulation capability, environmental, and radiation phenomena associated with thermal stability as important (H/L).

Graphite oxidation

The major phenomena of oxidation and chemical attack on graphite were identified in the GRAPH PIRT (Ref. 4). The ACTH panel also identified the phenomena of fuel performance with oxygen attack for the air ingress accident. This is a cross-cutting concern with the FPT due to the fission-product generation from the oxidation of the core graphite. There are a number of phenomena identified and evaluated by the FPT PIRT that lead to fission products escaping the particle, fuel matrix (pebble or rod), and the graphite.

Additionally, the ACTH panel identified air ingress accident and the need for mixed gas analyses and heat transfer correlations for those gases. The phenomena of fuel performance and graphite oxidation were noted as an H importance and M knowledge level. Similarly, the FPT panel identified gas composition as having H importance because of the oxygen potential and chemical activity and assigned an M knowledge level as well.

Insulation failures

The GRAPH PIRT (Ref. 4) identified "Foreign object (debris)" deposition for items such as insulation which fall onto a channel. This is linked to high-temperature materials and component failures. The HTMAT PIRT also identified environmental and radiation degradation and stability phenomena (H/L) for the insulation.

The HTMAT PIRT (Ref. 3) reported on phenomena for both metallic and nonmetallic components. For nonmetallic components, radiation-induced degradation and oxidation phenomena (medium importance) were evaluated. Composite structural design/fabrication and its effect on carbon composites were noted as well. A number of phenomena affecting the structural aspects of graphite blocks were identified in the GRAPH PIRT (Ref. 4).

Heat exchanger failure/cyclic loading

Process heat transport line and IHX failures were identified by two panels. The HTMAT panel focused on phenomena that may result in a failure of the IHX, whereas the PHHP panel focused on fuel and primary system corrosion phenomena resulting from a postulated PHX failure event and blow-down effects from IHX failure.

Cyclic loading and the temperature effects on SSCs was also identified by PHHP panel (Ref. 5) as they were concerned with the process heat and resultant cycling of the reactor. Thermal cycling and fatigue were also, of course, identified by the HTMAT panel for the IHX (again, it is noted that the HTMAT panel did not evaluate molten salt in the design). This raises the issues and questions associated with considering the additional thermal cycling considerations resulting from adding the process heat capability and associated IHX/PHX to the NGNP. Of course, more design details are needed to assess adequately the impacts of the process heat aspects of the NGNP.

Chemical attack (molten salt as example) to core

The ACTH identified molten salt and core support material degradation phenomena in the event of heat exchanger failure (if molten salt is used). The process heat team also identified leaks into the primary system and assessed possible damage to safety systems on the reactor side resulting from PHX and IHX failures, as well as unwanted chemical additions to the primary and the possibility of fuel damage. The GRAPH PIRT (Ref. 4) and the FPT PIRT (Ref. 2) identified the generic phenomena of chemical attack on the graphite.

It is important to note that the HTMAT panel did not consider molten salt within the scope of their PIRT (Ref. 3), as it was believed that such a process heat design was not in consideration for the NGNP, while other panels simply assumed failure of heat exchangers and assessed some molten salt aspects.

6. SUMMARY AND CONCLUSIONS

Currently licensed commercial LWRs rely more on active safety systems to prevent fuel melting and relocation and subsequent generation of large source terms. The NGNP philosophy is different in that it relies on a robust ceramic-coated fuel particle in a relatively chemically inert environment (helium), immobilization of the small FP releases during normal operation, and passive heat dissipation to withstand design basis events with minimal fuel damage and source term generation. As such, the NGNP places a burden on the designer to provide validation of key passive safety phenomena (conduction, radiation from the vessel to the RCCS), as well as reliance on the coated-fuel-particle performance and a stable graphite core structure. Additionally, FP release and transport behavior must be well understood (or at least bounded) if the vented confinement approach is part of the design and credit is to be taken for dose reduction by the intrinsic features of the reactor and reactor building. Some of the unique aspects of the technology require that various passive phenomena associated with the NGNP, with high importance ratings, should have a high knowledge level.

This report documents a structured assessment of the safety-relevant phenomena in each of five topical areas [accidents and thermal fluids (including neutronics), fission-product transport and dose, high-temperature materials, graphite, and process heat for hydrogen production]. The NGNP design (either pebble bed or prismatic core) employing a graphite-moderated gas-cooled reactor with a provision for process heat was analyzed by the five PIRT panels. The nine-step PIRT process was used as a methodology for providing expert assessments of safety-relevant phenomena. The key findings from each PIRT panel are briefly summarized below.

6.1 Accidents and Thermal Fluids (Including Neutronics)

The PIRT panel evaluated both normal operation and postulated accident scenarios, concentrating on the thermal fluid aspects of the events but considering the neutronic behavior as well where appropriate. Four types of challenges were evaluated: heat removal, reactivity control, confinement of source terms, and control of chemical attacks. The panel evaluated normal operations, LOFC events (both pressurized and depressurized), air ingress, reactivity events, and some phenomena associated with the provision of a process heat loop and intermediate heat exchanger.

The most significant phenomena identified by the panel include the following:

- a variety of primary system cool down phenomena (conduction, convection, and radiation), including RCCS performance;
- a variety of reactor physics phenomena (feedback coefficients, power distribution for normal and shutdown conditions) as well as core thermal and flow aspects. These often relate to the power-to-flow ratio and thus impact peak fuel temperatures in many events; and
- postulated air ingress accidents that, however unlikely, could lead to major core and core support damage.

Upon completion of the PIRT process, both the panel and industry observers noted that there were many phenomena with (H/L) and (H/M) ratings and few with low importance ratings. Also, many phenomena are shared by other panels (Table 12). Possible reasons for this "skew" in the ratings are that analysts would naturally tend to concentrate on important phenomena, and subsequently tend to judge them as high importance (until analyzed with validated models or proven otherwise). This seemed endemic to the nature of the PIRT phenomena selection process. Another observation to be made is that a systematic sensitivity analysis of important effects on the primary FOMs is needed. Such an approach will provide a better quantitative perspective and a much better foundation for the qualitative ratings and associated rationale.

6.2 Fission Product Transport and Dose

The panel evaluated both normal operation, which established the initial-condition fission product distribution in the primary circuit, and accident conditions that contributed to the release of fission products. The PIRT panel found that at this early stage in the NGNP design, a wide range of transport options needed to be examined. Since the FP release from normal operation is not negligible, and this material is potentially available for release during the accident, one must have assurances that it is immobilized in a manner that is not threatened by the accident, even if later releases are inconsequential.

Some of the significant phenomena identified were as follows.

- Fission product contamination of the graphite moderator.
- Fission product contamination of primary circuit surfaces incurred during normal operation, including the power conversion unit components.
- Transport of fission products into the confinement building as a result of various types of accidents involving depressurization.
- Transport of fission products from the confinement building to the atmosphere. This is also primarily a building leakage problem but depends on the gaseous and suspended aerosol inventory of fission products in the building and filtering provisions. In addition, chemical reactions of fission products in the building may affect their transport.
- Behavior of the fission product inventory in the chemical cleanup or fuel handling system during an accident. An overheat event or loss of power may cause release from this system and transport by some pathway into the confinement building or environment.
- Transport phenomena (such as chemical reactions with fuel, graphite oxidation) during an unmitigated air or water ingress accident.
- Quantification of dust in the reactor circuit (from several sources). This may be easily released during a primary boundary breach. Carbon-based dust is generally quite absorptive of fission products and, when combined with its high mobility, leads to an important path from the reactor core to the environment. The highest dust quantities are expected in the pebble bed core (~10–50 kg for a test reactor; perhaps much more for a power reactor) and the lowest in the prismatic core (at least an order of magnitude less).

Because of the dependence on diffusive, physiochemical, and aerosol behavior, the transport of fission products depends on a host of chemical, thermodynamic, fluid flow, and physical properties. At this point, much is not clear about the actual material properties, their exact environment, some physics issues, and secondary transport mechanisms for dust. Future design work and testing should clarify some of the open questions and allow the analysis to be more focused on the merits of the actual design.

6.3 High Temperature Materials

The major aspects of materials degradation phenomena that may give rise to regulatory safety concern were evaluated for major structural components and their associated materials. These materials phenomena were evaluated with regard to their potential for contributing to fission product release at the site boundary under a variety of event scenarios covering normal operation, anticipated transients, and accidents and the currently available state of knowledge with which to assess them. Key aspects identified by this panel are

- high-temperature material stability and the ability of this component to withstand service conditions;
- issues associated with fabrication and heavy-section properties of the RPV;

- long term thermal aging and possible compromise of RPV surface emissivity (emissivity of RCCS as well); and
- high-temperature performance, aging fatigue, and environmental degradation of insulation.

A number of other high temperature issues were identified by the panel for many other components. These include control rods, power conversion unit, circulators, RPV internals, and primary system valves. The analysis was summarized in Sect. 4.3.3, and more detail can be found in the panel's report.

6.4 Graphite

Much has been learned about the behavior of graphite in nuclear reactor environments in the 60 plus years since the first graphite reactors went into service. The current knowledge base is well developed. Although data are lacking for the specific grades being considered for the Generation IV concepts, such as the NGNP, it is expected that the behavior of these graphites will conform to the recognized trends for near-isotropic nuclear graphite. Some of the significant phenomena noted by the panel are

- material properties (creep, strength, toughness, etc.) and the respective changes caused by neutron irradiation;
- fuel element coolant channel blockage due to graphite failures;
- consistency in graphite quality (includes replacement graphite over the service life); and
- dust generation and abrasion (noteworthy for pebbles but of concern as well for the prismatic design).

Theories that can explain graphite behavior have been postulated and, in many cases, shown to represent experimental data well. Thus much of the data needed is confirmatory in nature. However, these theoretical models still need to be tested against data for the new graphites and extended to higher neutron doses and temperatures typical of Generation IV reactor designs. It is anticipated that current and planned future graphite irradiation experiments will provide the data needed to validate many of the currently accepted models, as well as provide the needed data for confirmation and validation of designs.

6.5 Process Heat and Hydrogen Co-Generation Production

This panel found that the most significant external threat from the chemical plant to the nuclear plant is from ground-hugging gases that could be released. Within this category, oxygen is the most important because (1) it is the by-product from all hydrogen production processes that start with water and (2) it may be released continuously as a "waste" if there is no local market. This is due to its combustion aspects, plume behavior, and allowable concentration and is consistent with the chemical safety aspects and known risks of oxygen plants. Accidental hydrogen releases from the chemical plant were considered a lesser concern in terms of reactor safety because of the high buoyancy of hydrogen and its tendency to be diluted by air.

Since there are no existing facilities that release large quantities of oxygen, this is new in the context of chemical plant experience and thus deserves special attention. The knowledge base is considered to be medium to high but depends upon what particular aspect of oxygen and its impact is evaluated.

This panel was also concerned with the high importance of heat exchanger failures and associated phenomena for blow-down and other impacts that failures may have on the primary system, be it chemical, nuclear, or a pressure pulse. IHX and PHX failures were noted as significant.

Multiple high-temperature reactors have been built to produce electricity, and there have been many reactor safety studies. Consequently, there is a large body of experience and analysis that supports the safety evaluations of the other PIRT panels. In contrast, very little work has been done to address the safety issues of collocating nuclear and chemical plants. The safety uncertainties associated with collocation of nuclear and chemical plants are significantly larger than the uncertainties associated with internal reactor safety challenges since there have been only limited studies in this area. While the safety-significant phenomena have been identified, the detailed studies to understand relative risks and consequences are at a much earlier state of development.

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10. SUPPLEMENTARY NOTES
S. Basu, NRC Project Manager

11. ABSTRACT *(200 words or less)*

A phenomena identification and ranking table (PIRT) process was conducted for the Next-Generation Nuclear Plant (NGNP) design. This design (in the pre-conceptual stage) is a modular high-temperature gas-cooled reactor (HTGR) that generates both electricity and process heat for hydrogen production. Expert panels qualitatively rated the importance and assessed the knowledge levels (KLs) of phenomena in the areas of accidents and thermal-fluids, fission product transport and dose, high-temperature materials, graphite, and process heat for hydrogen production. This main report summarizes and documents the process and scope of the reviews, noting the major activities and conclusions. The identified phenomena, analyses, rationales, and associated ratings of the phenomena, plus a summary of each panel's findings, are presented. Individual panel reports for these areas are provided as attached volumes to this main report and provide considerably more detail about each panel's deliberations, as well as a more complete listing of the phenomena that were evaluated.

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