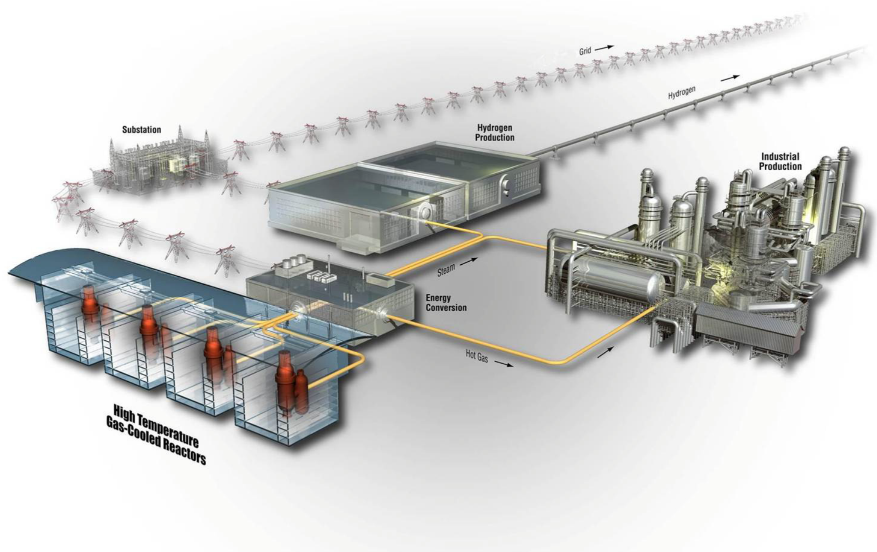
INL/EXT-11-21397

Assessment of NGNP Moisture Ingress Events

NGNP Moisture Ingress Assessment Committee (see Appendix A)

April 2011



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April 2011

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Next Generation Nuclear Plant Project

Assessment of NGNP Moisture Ingress Events

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April 2011

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ABSTRACT

A panel of experts in areas related to the U.S. next generation nuclear plant (NGNP) design assessed modular moisture ingress events for a high temperature gas-cooled reactor using a phenomena identification and ranking process. Consideration was given mainly to the prismatic core gas-cooled reactor configurations incorporating a steam generator within the primary circuit. Some aspects of ingress events and consequences peculiar to pebble-bed cores were also noted. Safety-relevant phenomena, importance, and knowledge-base concerns were assessed for normal operation and steam/water ingress accident scenarios.

The panel's judgment of the importance ranking of a given phenomenon (or process or event) was based on the effect it would have upon one or more figures of merit or evaluation criteria, including public and worker dose, fuel failure, pressure increases, and primary (and other safety) system integrity. The major phenomena and issues of concern that were identified, categorized, and generally agreed upon as being of high importance and requiring more attention were:

- Characterization of graphite properties and performance. In particular, the effects of long term graphite exposure to moisture levels in the primary coolant system should be investigated further. Long term structural damage is also a consideration as it may affect initial conditions in the evaluation of significant moisture ingress accidents.
- Investigation into the importance of the plate-out and resuspension of radionuclides in the primary coolant system is needed to determine if these phenomena are important to overall dose calculations to workers or the public, although exposures would occur only upon relief valve(s) opening. The panel identified the need for data and improved modeling for reactor building decontamination factors.
- Development of a systems accident code capable of simulating phenomena associated with moisture ingress, used in sensitivity studies to acquire a better understanding of the potential consequences of postulated moisture ingress event sequences, and to optimize the design of mitigation systems in the process.

As the plant designs mature, more scoping analyses need to be performed to further identify phenomena and sequences important to plant performance. These phenomena and sequences will enable the design of experiments to obtain the necessary data and the need for additional analytical tools.

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FIGURES

TABLES

ACRONYMS

AOO	Anticipated Operational Occurrence
AVR	Arbeitsgemeinschaft Versuchsreaktor (Germany; Association Experimental Reactor Ltd.)
BDBA	Beyond Design Basis Accident
DBA	Design Basis Accident
DBE	Design Basis Event
DF	decontamination factor
DOE	Department of Energy
FOM	figure of merit
FSV	Fort St. Vrain (High Temperature Gas Reactor, U.S.)
GA	General Atomics
HTGR	high temperature gas-cooled reactor
HTR	high temperature reactor
IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory
IPS	investment protection system
MHTGR	Modular High Temperature Gas-cooled Reactor
NA	not applicable
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
NUREG/CR	Nuclear Regulatory Commission/Contractor Report
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Reactor
PBR	pebble bed reactor
PIRT	phenomena identification and ranking table
PMR	prismatic modular reactor
PSID	Preliminary Safety Information Document
R&D	research and development
RCCS	reactor cavity cooling system
RPS	reactor protection system
RSS	reactor safety system
RPV	reactor pressure vessel
SAG	Senior Advisory Group
SCS	shutdown cooling system

SG	steam generator
SRDC	Safety-Related Design Condition
T/H	thermal-hydraulic
THTR	Thorium High-Temperature Reactor (Germany)
TRISO	tristructural-isotropic
V&V	verification and validation

Assessment of NGNP Moisture Ingress Events

1. INTRODUCTION

The *Energy Policy Act of 2005* (Public Law 109-58) required the Secretary of the U.S. Department of Energy (DOE) to establish the Next Generation Nuclear Plant (NGNP) Project to manage the research, development, design, construction, and operation of a prototype plant that would use process heat to generate electricity and/or produce hydrogen. The NGNP Project would be supported by the research and development (R&D) activities of the Generation IV Nuclear Energy Systems initiative.

DOE selected the high temperature gas-cooled reactor (HTGR) as the reactor concept to be used for the NGNP. Preconceptual designs for the NGNP were developed by three reactor suppliers. The characteristics of these designs are summarized in Table 1.

Table 1. Key operating parameters from preconceptual NGNP designs (INL 2007).

Condition or Feature	AREVA	General Atomics	Westinghouse
Power output in MW(t)	565	550 to 600	500
Reactor type	Prismatic block	Prismatic block	Pebble-bed
Core outlet temperature in °C	900	up to 950	950
Core inlet temperature in °C	500	490	325
Cycle Configuration	Indirect cycle[a]: parallel hydrogen process and power conversion	Direct power conversion cycle[b]: parallel indirect hydrogen process	Indirect cycle: series hydrogen process and power conversion

a. Indirect cycle uses an intermediate heat exchanger to isolate the radioactively contaminated primary fluid from the power or hydrogen generation processes.

b. Direct power conversion cycle uses the primary coolant in the power conversion unit.

At a meeting of the NGNP Senior Advisory Group (SAG) in October of 2008 (SAG 2008), it was agreed that two designs would be pursued:

- An indirect cycle configuration with a pebble bed reactor (PBR) and a gas-to-gas intermediate heat exchanger as shown in Figure 1
- An indirect cycle configuration with a prismatic block reactor and steam generator (SG) as shown in Figure 2.

The SAG also agreed that the reactor outlet gas temperature would be in the range of 750 to 800°C.

Additional studies performed in 2009 (Geschwindt 2009; Carosella 2009; WEC 2009) resulted in the operating parameters shown in Table 2. Although the latest 2009 design concept for the PBR employed an intermediate heat exchanger, the team was considering the use of an SG in the primary loop. This idea was presented to the SAG in July 2009 (SAG 2009).

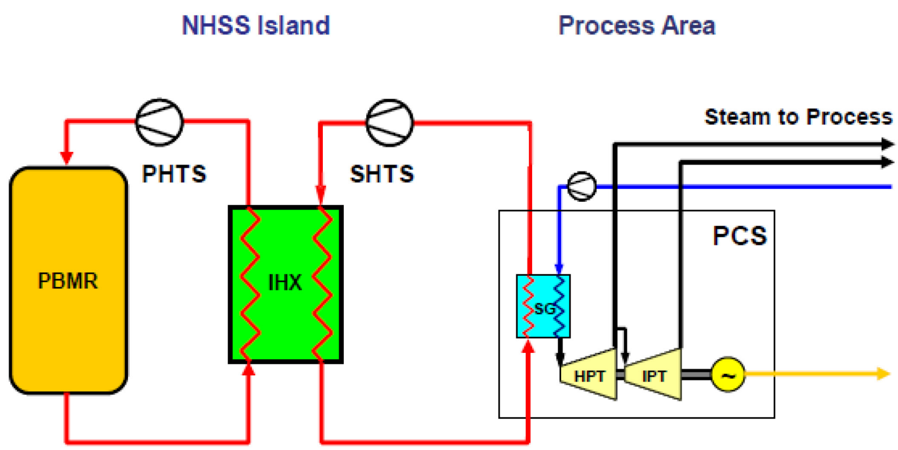

Figure 1. Pebble bed reference configuration (October 2008).

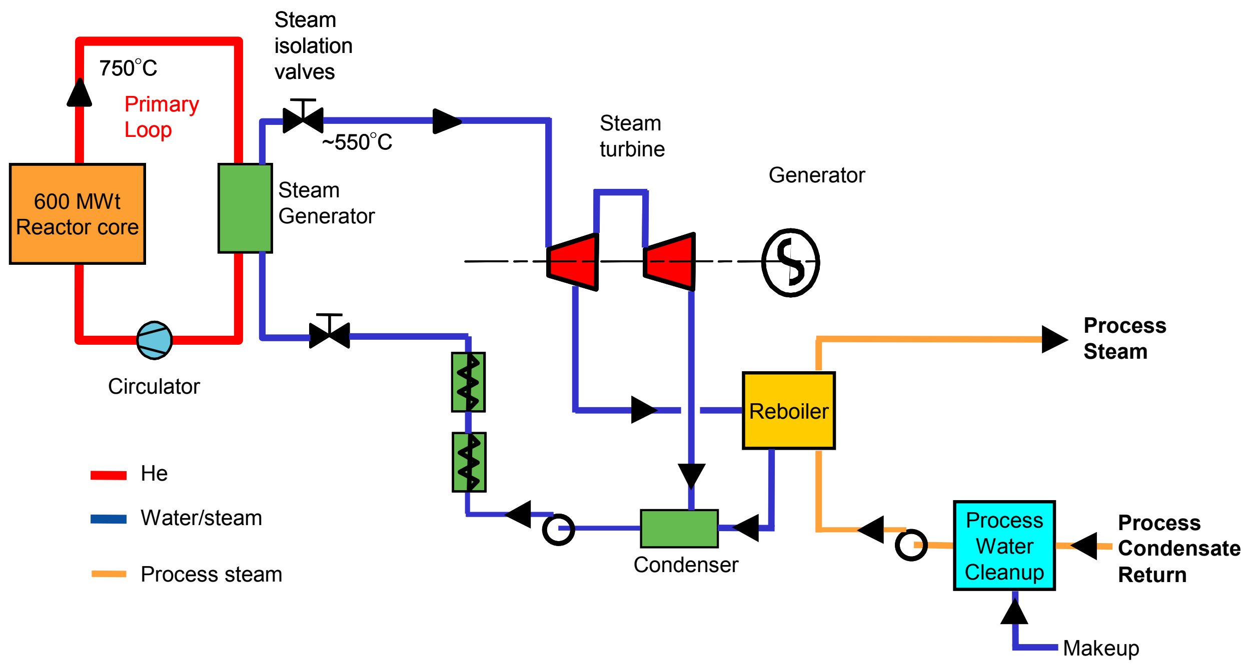

Figure 2. Prismatic block reference configuration (October 2008).

Table 2. Key operating parameters for the 750°C NGNP designs.

Condition or Feature	AREVA	General Atomics	Westinghouse
Power output in MW(t)	625	600	500
Reactor type	Prismatic	Prismatic	Pebble bed
Core outlet temperature in °C	750	750	750
Core inlet temperature in °C	325	322	280
Coolant pressure (MPa)	6	7	9
Cycle Configuration	Indirect Rankine (Steam)	Indirect Rankine (Steam)	IHX to Rankine (Steam)

As part of an effort to assess the safety performance of the NGNP and to identify the analytical tools and additional research that would be needed to support the safety analyses, design, and licensing efforts, the Phenomena Identification and Ranking Table (PIRT) process was applied to various aspects of the NGNP fuel in NUREG/CR-6844 (Morris, 2004), plus several areas of the NGNP design, including accident and thermal fluids analysis, fission product transport and dose, high temperature materials, graphite, and process heat for hydrogen co-generation in NUREG/CR-6944 (Ball, 2008). The NGNP design PIRT was conducted at about the same time as the preconceptual designs were being developed and was based on those configurations. Although these designs did not include an SG in the primary loop, there was some discussion of water ingress included in NUREG/CR-6944 (Appendix A of Volume 2). Given the current configurations, and the indications that the PBR concept might also employ an SG in the primary loop, the NGNP Project decided to develop an assessment of the impacts of postulated water/steam ingress events on the HTGR to better understand the needs for additional R&D, analytical tools, and experiments to validate the codes. Moisture ingress events have been recorded for some HTGRs, in particular the German Arbeitsgemeinschaft Versuchsreaktor (AVR) and Fort St. Vrain (FSV) reactors. However, no significant reactivity insertion events associated with water ingress were recorded in the experience base.

2. OBJECTIVES

The purpose of this assessment was to evaluate the effects of water or steam ingress into the NGNP primary coolant system and reactor core. Given that the maturity level of current preconceptual designs and subsequent design efforts are limited, this evaluation addresses the issues in a qualitative fashion. It is likely that a more formal PIRT-like effort will be performed when more design details and analyses are available. The objectives of this study are to:

- Identify causes and describe scenarios of water/steam ingress postulated events
- Assess the knowledge base for the effects of water/steam ingress on the core physics, fission product transport and release to the primary system and environs, and the long and short-term corrosion effects on graphite and other structural materials and components that could affect plant safety (the effects on fuel integrity were covered in a prior PIRT (Morris, 2004)
- Assess the capability and availability of analytical models and tools, along with sufficient data to support the models, to analyze water/steam ingress events
- Provide rankings for the phenomena involved according to importance and the current knowledge base
- Make recommendations, as appropriate, for additional R&D, code development, and any additional experimentation that may be needed to support the analytical work associated with postulated water/steam ingress events or long-term corrosion effects from low-level moisture in the primary system.

3. APPROACH

The nine-step PIRT process summarized in Section 5 was followed to the extent practical. Additional details on the PIRT process are provided by Wilson and Boyack (1998).

Based on previous experience with the PIRT evaluation process, a novel method for prioritizing recommendations was developed from the Pebble Bed Modular Reactor (PBMR). Some of the lessons learned from that exercise were incorporated into the current evaluation process. The recommendations were to specify, where applicable, more effort to be spent on R&D (e.g., to collect experimental data) and/or analysis (code development and use).

Attention was focused on identifying the research efforts and analytical tools needed to support design confirmation and licensing, along with experiments that may be needed to support the analyses.

The specific phenomena considered included:

- Reactivity effects (increase for under-moderated core)
- Reduction of control/shutdown rod worth
- Pressure increase in primary helium system
- Pressure relief valve actions
- Graphite oxidation/corrosion
- Fission product release and transport
- Explosive gas mixtures within the reactor vessel or reactor building.

Even though both prismatic block and PBR HTGR concepts are being considered for the NGNP, the study was focused on the prismatic block configuration for the NGNP HTGR. The South African government's recent decision to cancel funding of its PBMR leads to some uncertainty as to the future design of the PBR. The prismatic block reactor design considered was based on the General Atomics' Modular High Temperature Gas-cooled Reactor (MHTGR), for which General Atomics and DOE submitted a Preliminary Safety Information Document (PSID) in the late 1980s, (DOE 1992) along with additional information provided by General Atomics. The MHTGR is a 350 MW(t) prismatic block HTGR with a single SG in the primary loop. The ingress-related similarities and differences between the PBR and the prismatic block reactor were also addressed briefly.

The scenarios considered included postulated accident transients such as pressurized and depressurized loss-of-forced convection coupled with water ingress, as well as long-term, steady-state operation considering the low-level moisture conditions in the primary system.

4. MOISTURE INGRESS ACCIDENT SCENARIO BACKGROUND

Typically, the grouping of HTGR accident scenarios is based on either the nature of the challenge to fundamental safety functions or on dominant phenomena occurring during the course of the event.

A typical grouping based on challenges to fundamental safety functions results in challenges to heat removal, reactivity control, confinement of radioactivity, and control of chemical attacks. The panel's listing of phenomena of interest was organized so as to cover these safety function categories.

The initiating event and ensuing event sequence for a postulated accident often challenges more than one safety function, as noted in the following two examples:

- Primary system pressure boundary breaks (challenge to confinement of radioactivity). The common feature of these events is that they result in a release of radioactivity from the primary system that may result in a dose to workers and/or the public. These include all leaks greater than normal operational leakage rates. Breaks with an accompanying loss of forced core cooling result in challenges to heat removal as well. Pressure boundary breaks may also lead to air ingress, which in turn challenges the control of chemical attack. They also present a challenge to the heat removal function, as in the following example.
- Primary system breaks in the interface with steam or cooling water systems (e.g., SG or heat exchanger tube breaks that result in steam or water ingress). Depending on the design, primary-to-water/steam system pressure differences, and pressure relief valve operation, there may be radioactivity releases resulting in worker and/or public dose. Such events therefore challenge reactivity control if steam in the core introduces a positive reactivity change and control of chemical attack as well as confinement of radioactivity.

There are a number of event sequences that may be postulated and accident states that could be encountered. The main objective here was to ensure that appropriate event phenomena were covered while avoiding duplication, if possible.

Both the normal operation and accident characteristics of modular HTGRs differ from those of other power reactor designs. Because of these differences, their specific passive safety features and the response of the plant systems and operators need to be considered appropriately. Because of the constraints put on the modular HTGR design (by the designers) and its passive safety features, traditional Design-Basis Accident (DBA) events such as loss of coolant do not result in large fission product releases from the primary coolant system, so the results of probabilistic safety analysis methods are dominated by low-probability initiating events. Safety margins are enhanced because of the passive features that accomplish safety functions without reliance on alternating current (ac) powered active safety systems, as long as it can be shown that the basic core configuration and systems that enable such passive cooling are not affected in the events. Furthermore, the plant response to low-probability, serious events can typically be modeled with greater assurance (e.g., no departures from nucleate boiling, no core melting, no need for core catchers, etc.).

The Nuclear Regulatory Commission's (NRC's) preapplication review of the MHTGR in the 1980s as documented in NUREG-1338 and the extensive supporting documentation provided by DOE in its report *Preliminary Safety Information Document* (*PSID) for the Standard MHTGR* (DOE 1992) thoroughly document a multiyear regulatory review of a 350-MW(t) prismatic modular reactor (PMR) plant similar to those currently under consideration for the NGNP. This applies to the current leading design for NGNP because a dominant risk is from steam/water ingress via SG tube leaks or breaks. Candidate NGNP PBR reactor designs, with power ratings on the order of ~200 to 400 MW(t), are similar to the German Module design of ~200 MW(t), but with an annular (and taller) active core utilizing a solid central reflector for the higher-power versions. Another difference may be the inclusion of the high-temperature process heat systems (such as hydrogen production) in proposed NGNP designs.

Tristructural Isotropic (TRISO) fuel safety issues were covered earlier in a previous NRC PIRT report dealing exclusively with TRISO-coated fuel particles (Morris, 2004). In that PIRT, the assumptions were made that the fuel kernels would be uranium dioxide (UO_2) and that the reactor was a PBR; however, the report authors maintained that the approach was more general and less plant-specific since "The information needed to develop more detailed specifications was not available to the panel." In that case, detailed PIRTs were prepared for fuel manufacturing, normal operation in a general sense, and four accident scenarios. Incremental PIRTs addressed importance rankings that would be altered for UCO fuel and prismatic fuel forms. The four accidents selected for the fuel PIRT emphasized those scenarios the panel thought presented the greatest challenge to fuel integrity and included:

1. Reactivity insertion based on the effect of rod ejection in the PMR, given excess reactivity representative of that in a PMR, but applied to conditions in the PBR
2. Power pulse of several seconds duration
3. Depressurized core heat-up followed by water ingress
4. Depressurized core heat-up followed by air ingress.

Major design and technology areas that either influence safety or have relevance to safety in the context of satisfying regulatory requirements would normally cover the following:

- Design, including design standards and the selection and qualification of materials, especially those materials used or relied upon in applications for safety-related structures, systems, and components.
- Fabrication, installation, preservice inspection and testing, maintenance, and in-service inspection and testing of materials and components, especially for "a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier" (Ball, 2008).
- Operation, including the safety functions of the operator, maintenance of the plant within technical specification limits based on reliable and adequately calibrated instrumentation, and potential risk from insider threat in an otherwise inherently safe reactor. Particular attention should be paid to instrumentation that is "used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary," or that is "used to detect and quantify a process variable, design feature, or operating restriction that is an initial condition of a design basis accident, or a transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier," or that is "used for post-accident monitoring" (Ball, 2008).
- Accident conditions, as affected by design selections, testing, and inspections of key materials and components to assure continued functionality and operability, operator or maintenance errors, and potential insider threat.

In view of some of the considerable differences in design philosophy and passive safety features of the modular HTGRs compared to those of conventional LWRs, studies identifying and characterizing the phenomena involved in the important postulated accident sequences are appropriate. The event selection process was based on the panel's study of these features.

5. WATER INGRESS ACCIDENT ANALYSIS EVALUATION PROCESS

The assessment panel adopted, to the extent practical, much of the NRC's standard nine-step process for implementing a PIRT as described in this section.

5.1 Step 1—Issues

In anticipation of future licensing applications for modular HTGRs such as the NGNP, work is needed in certain design and technology areas that either influence safety or have relevance to analyses satisfying applicable regulatory requirements. This is a multistep process, one of which is to identify phenomena that are characteristic of the NGNP designs. Certain phenomena come into play in influencing the response of the plant to initiating events and the postulated event sequences that follow. The issues addressed are the importance of these phenomena in potential eventual outcomes of the sequence, and how well these phenomena can be characterized by using existing data and analytical techniques.

5.2 Step 2—Assessment Objectives

For the case of this assessment, the objectives are to:

1. Identify safety-relevant NGNP phenomena associated with moisture ingress
2. In each case, establish evaluation criteria or figures of merit (FOMs)
3. Rank the importance of the phenomena applicable to plant operation and/or postulated accident scenarios
4. Identify and rank the knowledge base associated with safety-relevant phenomena
5. Provide recommendations, if applicable, for additional efforts to be spent on R&D and/or analysis
6. Provide references (a bibliography) for use in subsequent reviews and evaluations.

5.3 Step 3—Hardware and Scenario

5.3.1 Hardware

The NGNP is currently in the conceptual design stage, and DOE's selection of the reactor design and process heat sectors is in progress. "Hardware," in this context, would refer mainly to the leading NGNP candidate reactor core and primary system design, which is expected to be similar to the DOE MHTGR developed in the 1980s.

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 FSV reactor fuel elements). The fuel compacts, in turn, are composed of fuel particles bonded by a carbonaceous matrix. The fuel particle itself is a TRISO coated fuel particle that provides the primary barrier against release of radionuclides.

Other barriers against release of radionuclides are the primary coolant system pressure boundary and, to a lesser extent, the reactor building itself. Several confinement and containment options for the reactor building 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 is usually assumed to be very small, requiring no holdup, while any later releases are assumed to be modest with little or no pressure differential driving force.

5.3.2 Moisture Ingress Accident Scenarios

While classification of plant events is not within the scope of this PIRT, judgments of the importance of phenomena were affected by risks posed by the accidents being considered and the potential frequency of occurrence of those events. A typical set of event classifications are:

- *Anticipated Operational Occurrence (AOO):* An AOO is an expected event that may occur one or more times during the life of a plant. AOOs typically have a mean frequency of occurrence of 10^{-2} per plant year or higher.
- *Design Basis Accident (DBA):* A DBA is an infrequent event not expected within the lifetime of one plant, but perhaps occurring once during the collective lifetimes of a large number of plants. Plants are designed to mitigate the effects of a DBA using only equipment classified as safety grade. DBAs typically have a mean frequency between 10^{-2} and 10^{-4} per plant year.
- *Beyond Design Basis Accident (BDBA):* A BDBA is a rare event that is not expected to occur even within the collective lifetimes of a very large number of similar plants. However, the plant is designed to mitigate their consequences, taking credit for available safety-related equipment, operator actions, any existing or ad hoc nonsafety-related equipment, and accounting for long time periods potentially available for corrective actions. BDBAs are usually associated with events having a mean frequency between 10^{-4} and 5×10^{-7} per plant year. Typically, the lower frequency limit is considered a cut-off frequency below which consideration and analyses are not required.

The scenarios selected for consideration in this assessment were:

1. Steam-water ingress events, primarily caused by SG tube leaks or breaks, but also possible leaks in water-cooled heat exchangers in the primary system such as in the shutdown cooling system (SCS)
2. Effects of long-term moisture presence in the primary system during normal operation that could cause structural damage. Such normal operation was also considered because it can affect the plant's vulnerability and its ability to achieve passive cooldown with a known structural condition in subsequent postulated events.

There were two steam water ingress events selected from the MHTGR PSID (DOE 1992) for discussion by the panel. These are designated DBE-6 (an SG tube rupture, classified as a Design Basis Event), and SRDC-6 (an SG tube rupture with only safety system mitigation, classified as a Safety-Related Design Conditions Event) and are described in more detail in Section 7.

5.4 Step 4—Evaluation Criteria or FOMs

The panel discussed evaluation criteria that would be appropriate to moisture ingress events and derived the following list, with criteria ranked (approximately) in order of importance:

1. Top level: dose at the site boundary or radioactive release from the reactor building
2. Second level: worker dose
3. Loss of structural integrity of the graphite (or composite, if applicable) reactor internals
4. Release of primary system contaminants to the reactor building:
   a. Does the PRV open?
   b. How many valves open?
   c. When and how often?
5. Fuel temperature
6. Configuration changes (e.g., dimensional changes resulting in different flow distributions) impact on subsequent restart or operations
7. Explosive gas concentrations
8. Fission product mobilization.

There is also the need to consider chronic degradation of graphite because of low-level concentrations of moisture in the primary system during normal operations. This chronic degradation may result in loss of material that could affect the integrity of structural supports.

5.5 Step 5—Knowledge Base

The panel compiled and somewhat reviewed the contents of a knowledge database that included:

- Recent design information available for both prismatic and PBR core types
- Relevant operational experience from FSV, the Thorium High-Temperature Reactor (THTR-300) in North Rhine Westphalia, Germany, and the AVR in Jülich, Germany
- The findings from the NRC preliminary safety evaluation of the steam-cycle MHTGR (NUREG-1338) and the MHTGR's PSID (DOE 1992)
- A database of extensive and comprehensive U.S. and international reports, many of which are available for downloading from the International Atomic Energy Agency (IAEA) website (www.iaea.org).

Extensive references are listed in the Bibliography (Appendix B).

5.6 Step 6—Identify Phenomena

As in the TRISO-coated particle fuel PIRT effort, the panel members first identified and then refined the phenomena lists. The term *phenomena* also includes events, processes, and characteristics.

Accident phenomena are typically classified by their challenges to the safety functions noted previously. The challenges to the designer-operator and the regulator are to ensure and confirm that the defense-in-depth features provided will reduce the probability and risks of serious accidents to acceptable levels. PIRT-like activities are part of a larger effort that lead to a comparison of the requirements with the existing (or developing) capabilities determining the analytical tools and data needed for confirmatory analyses. The applicability of confirmation activities, such as proving code capability via benchmarking (both code-to-code and code-to-experiment), is subject to varied interpretations because low probability serious accidents are not simulated experimentally in their entirety, but rather rely on a compilation of separate effects testing results that support a computational model and framework.

It is clear that both technological and regulatory perspectives will be needed to provide essential importance rankings to the elements involved.

Phenomena identification involves the listing of potentially significant situations and sequences, characterizing them, for example, with respect to their effect on core cooling, reactivity control, and radionuclide confinement, for the three classifications of events noted previously. For example:

1. Normal operation—peak fuel temperatures, fission product plateout (e.g., Ag-110m maintenance dose), loss of SCS
2. DBAs—long-term accidents where single-failure criterion applies
3. BDBA—multiple failures of safety-grade and/or passive systems, failure to maintain subcriticality, inadequate defense for a major earthquake, inability to limit water or air ingress, loss of all core heat sinks, etc.

In addition to equipment successes and failures, operator actions (both positive and negative) are to be considered, accounting for the typical very long accident response times.

5.7 Step 7—Importance Ranking

The panel ranked applicable phenomena in each table relative to one or more evaluation criterion or FOM, for example, "structural integrity." Each phenomenon was assigned an importance rank of "High," "Medium," or "Low," (H, M, L) accompanied by discussions where appropriate. Definitions associated with each of these importance ranks are given in Table 3.

Table 3. Importance ranks and definitions:

Importance Rank	Definition
Low (L)	Small influence on primary evaluation criteria
Medium (M)	Moderate influence on primary evaluation criteria
High (H)	High or controlling influence on primary evaluation criteria

Plant designs include various lines of defense to mitigate the consequences of postulated accident sequences. The panel considered the importance of the phenomenon or process to these sequences. Characterizations vary depending on plant design features such as pebble or prism core, process heat plant type, and loop design, as well as on the sequence assumptions such as break sizes and locations.

5.8 Step 8—Knowledge Level Ranking

Panel members assessed and ranked the current knowledge level for applicable phenomena in each case. High, medium, and low designations were assigned to reflect knowledge levels and adequacy of data and analytical tools used to characterize the phenomena, using definitions shown in Table 4.

Table 4. Knowledge levels and definitions.

Knowledge Level	Definitions
H	Well known: a state of knowledge and understanding sufficient to satisfy the requirements
M	Partially known: potentially an incomplete knowledge and understanding, with doubt as to its sufficiency
L	Very little known: incomplete knowledge and understanding (DKS)

5.9 Step 9—Documentation of the Assessment—Summary

The lists and tables that were generated (presented in the following sections) document the panel's discussions of phenomena identification plus the importance and knowledge level rankings, with accompanying rationales and recommendations, as appropriate. The resulting charts document the collective assessments. In cases where the collective assessment differed significantly from that of an individual panel member, that member's views are noted in the table (with the panel member initials) or elsewhere in the report.

6. SUMMARIES OF GENERAL MOISTURE INGRESS ASSESSMENT CONCERNS BY DISCIPLINE

6.1 Accident Sequences (SJB)

If one adopts the assumption (made by the panel) that the only means for fission product escape to the environs in a moisture ingress accident is via operation of the primary system pressure relief valves (thus impacting the primary FOM "…dose at the site boundary or radioactive release from the reactor building"), then a crucial parameter in the accident sequences is primary system pressure. Simultaneous depressurization events caused by breaks in the primary system, considered to be independent of water ingress, were judged by the panel to be beyond the scope of this assessment.

Pressure is affected by the amount of steam (or water) injected into the primary system, reaction gas generated, system temperatures (e.g., core heat up), and the timing of the sequence. For cases where relief valves open on high pressure, cycling of the valves would impact the amount of gas (and dose) involved.

For the DBE-6 scenario, it is assumed that nonsafety grade moisture monitors operate and successfully activate the SG isolation valves, thus cutting off the water ingress source. This results in a modest amount of total water ingress, with very little effect on reactivity, core temperature, water gas generation, and graphite oxidation. On the other hand, scenarios that assume failure of the moisture monitor trip action (as in SRDC-6) ultimately rely on a high-pressure trip (and potentially a subsequent safety valve actuation), which also activates the SG isolation. In this case, the safety valve actuation maintains the system pressure within design limits, but at the cost of releasing fission products to the environment. Repeated openings of the relief valve(s) would continue to maintain a safe pressure, but continue to release fission products.

In the case of a controlled response of the reactor to a trip, there were two assumptions discussed by the panel regarding the core cooldown. One is that upon SCRAM, the main cooling system functions to quickly cool the core down to temperatures at which the reactions of the steam/water with the graphite would be markedly reduced, thus quickly limiting the chemical effects of the ingress. The other assumption is that the reactor control system attempts to balance the power reduction (from a SCRAM) with a simultaneous cooling flow reduction to avoid a rapid cooldown that could result in damage from transient thermal stresses in the block fuel, reflectors, and support structures. In this case, potential thermal stress problems would be avoided, but the higher temperature (for a longer time) core would be more prone to chemical attack.

Events involving moisture ingress during shutdown and refueling modes of operation could have more of an impact on reactivity than those during normal operation. In liquid form, there could be more moderation occurring than with high-temperature steam where the density would be much lower. Moisture monitors may not be able to detect the ingress and conditions associated with the presence of moisture in the liquid form (e.g., saturation levels do not indicate the status of water concentrations).

Multiple evaluations may apply in certain cases, such as those where a phenomenon is important in one accident sequence but unimportant in another. In the events and situations noted, there are cases where changes in operating conditions (or model assumptions) could have significant effects on predictions of total water ingress and total fission product releases. For long-term accident sequences, possibilities for operator actions also need to be factored in (where intuitive interventions may be beneficial or not).

Because of these uncertainties in scenario outcomes (and other factors), it appears to be imperative to acquire or develop a systems accident code capable of simulating phenomena associated with moisture ingress, and use it to do sensitivity studies to acquire a better understanding of the potential consequences of moisture ingress events and to optimize the design of mitigation systems in the process.

It is the panel's understanding that a comprehensive systems code that could simulate the interactive mechanisms involved in these and other moisture ingress scenarios is not generally available to DOE or NRC. Codes of this ilk may include the TINTE code from Germany (licensed to the PBMR project) and perhaps the Institute for Nuclear Engineering and Technology in China, which may be available but is under license (Sikik 2008). TINTE has much of the chemistry already installed.

Integrated code features should include:

- Reactivity effects of steam/water ingress (increased reactivity with under-moderated core)
- Reduction of control/shutdown rod worth
- Moisture detection instrumentation, protection system logic and equipment actuation
- Core and primary coolant temperatures and flow distributions versus time
- SCRAM action (response; interference from corrosion?)
- All factors affecting pressure increase in the primary system
- Reactor pressure vessel (RPV) pressure relief system activation and response
- Plant protection system (safety and nonsafety) response
- Confinement/containment release characteristics
- Operating or startup or shutdown conditions
- SG isolation and dump system operation
- Graphite oxidation/corrosion products for the graphites qualified for in-vessel use.
- Fission product releases from fuel and graphite
- Explosive gas mixtures in both the RPV and the reactor building

6.2 Plant Design and Safety Analysis (LJL)

The design concept for the 350 MW(t) MHTGR provides the context for the evaluation of water ingress events and the phenomena relevant to the analysis of HTGR water ingress scenarios.

6.2.1 Plant Design

The MHTGR is a single loop steam cycle HTGR system. It has a single primary heat transport loop which carries heat from the reactor core to the SG where it is transferred to secondary coolant producing steam. The system configuration is shown in Figure 3. The reactor is in the upper steel vessel which is uninsulated to allow for passive decay heat rejection in the event that both the main heat transport loop and the SCS are not available for active heat removal. The SG is in the lower vessel. The vessels are connected by a cross vessel that contains the concentric flow path between the vessels.

Figure 3 also illustrates the main sources of water ingress in the MHTGR configuration. The main source of water in the steam cycle system is a possible SG leak. The SG contains a large quantity of water under very high pressure, so it is has the potential to inject significant quantities of water into the primary circuit under normal operating conditions. The water entering from an SG leak will be steam, water, or a mixture of the two, depending on where in the SG the leak occurs. Figure 4 shows one of the hundreds of tubes which form the SG tube bundle.

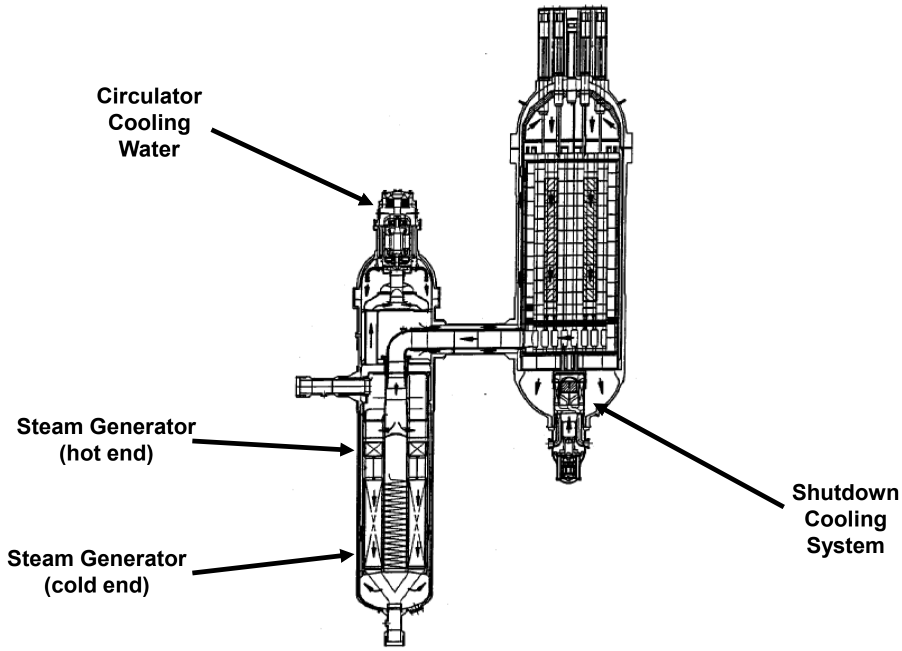

Figure 3. MHTGR arrangement and water ingress sources.

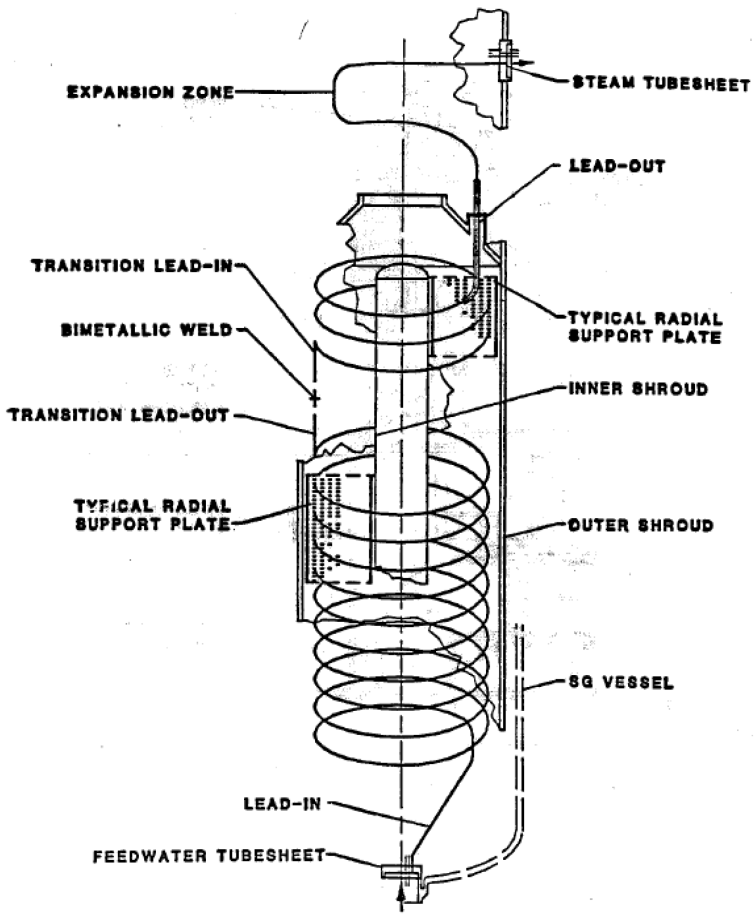

Figure 4. SG arrangement details.

Other sources of water include the circulator motor cooling heat exchanger and the SCS. However, these systems operate at much lower temperatures and pressures. Therefore, they are mainly of interest only for shutdown depressurized conditions.

The MHTGR design addresses both the safety and investment consequences of water ingress. As has already been mentioned, the safety considerations include pressure increase, reactivity effects, graphite oxidation (and combustible gas generation), and fission product mobilization. Pressure increase is of particular importance, since it can lead to opening of the primary relief valves, thus providing a path for radionuclides to escape from the primary circuit. Investment concerns include graphite oxidation and resulting component degradation and plant availability.

Of course, the first priority in designing for water ingress is to prevent such events. This is done through robust SG design and by specifying appropriate operating coolant parameters for the secondary and primary loops. However, the MHTGR concept also includes important features to mitigate the consequences of water ingress should such an event occur. In general, the approach is to detect and terminate the ingress as rapidly as possible in order to minimize the amount of water entering the primary loop. This is complemented by the inherent characteristics and large design margins, which can accommodate the residual consequences of the event.

Table 3 illustrates the basic MHTGR approach for detecting and mitigating the water ingress event. There are three main ways to detect water ingress:

- For a significant event in which moisture is rapidly transported to the core, the resulting reactivity transient will cause the reactor protection system (RPS) to trip the reactor because of a high power-to-flow ratio.
- If water ingress continues (e.g., reactivity transient was not large enough to trip on high power), the system pressure will gradually rise because of steam addition and potential graphite oxidation products. Eventually the RPS trips on high pressure, causing both reactor trip (using the reserve shutdown system) and main loop trip and isolation (trip circulator and close feedwater and main steam isolation valves).
- The most direct indication of water ingress is the measurement of high moisture concentration in the primary coolant. Exceeding the high moisture set point causes the investment protection system (IPS) to initiate reactor trip, main loop trip and isolation, and SG dump.

Table 5. MHTGR water ingress detection and mitigation.

		Reactor Trip	**Loop Trip (Circ trip and SG Isolation)**	**SG Dump**
Operator	Nonsafety Sensitive 30 min. delay	Not credited	Not credited	Not credited
RPS- High Power	For very rapid ingress Fast detection	X		
RPS-High Pressure	For very large ingress Slow detection	X (Reactor Safety System [RSS])	X	
IPS-High Moisture	Nonsafety Sensitive for small ingress	X	X	X
Shutdown Cooling Water System high pressure		Isolate the shutdown cooling heat exchanger and shut down the reactor		

No credit is taken for the operator in the safety analysis. Nonetheless, in a small leak that would take a long time to reach one of the trip set points, manual trip by the operator, based on gradually increasing moisture, is the most likely termination of the event.

Note that for the MHTGR design considered in the PSID analysis, the RPS trips (high power-to-flow ration and high primary pressure) are safety-related, while the IPS trip (high moisture) is nonsafety-related. The safety-related responses serve to trip the reactor to control power generation and to terminate the water ingress by isolating the SG. The investment protection serves to further reduce the consequences including subsequent oxidation, by draining residual water from the SG.

The steam and water dump system configuration is illustrated in Figure 5. If the high trip set point is exceeded, the feedwater and main steam isolation valves are closed and the dump valves are opened to drain the remaining water inventory from the SG. The dump valves are reclosed when the secondary pressure has dropped to the primary coolant pressure. The dump tank is partially filled with water to quench the hot water and suppress flashing. Draining the excess water from the SG prevents additional water from gradually entering the primary circuit via gravity drainage from the SG or vaporization of remaining water in the tube bundle.

The MHTGR has three main paths to remove residual heat from the reactor. Normally the main heat transport loop is used to cool down the reactor. However, if an SG leak is detected, the main loop will be shut down. When that happens, the SCS would be used for reactor cooldown and decay heat removal. On the other hand, if the SCS failed (or a water leak in the SCS occurred), then decay heat removal would be performed by the passive reactor cavity cooling system (RCCS).

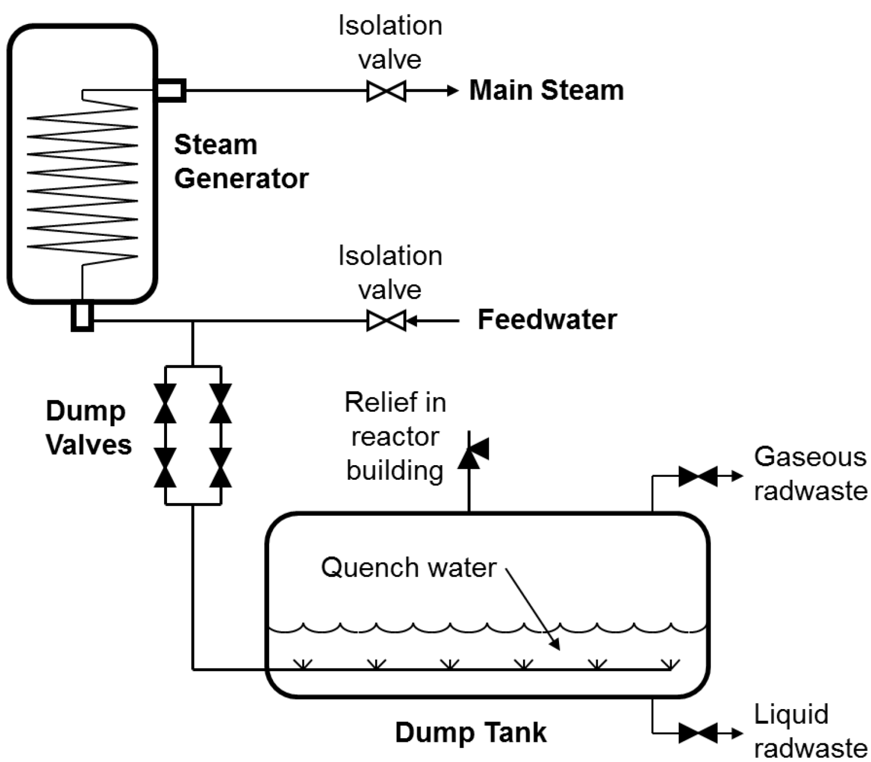

Figure 5. Steam water dump system configuration.

6.2.2 Water Ingress Safety Analysis

Analyses were performed for HTGR water ingress events to determine the specific system response to the scenario and to assess the consequences in terms of dose, impact on plant components, and to identify recovery sequences.

The analysis of a water ingress event in a conventional steam cycle HTGR such as the MHTGR typically includes the following steps (further explained below):

1. Identify leak size.
2. Determine ingress rate.
3. Evaluate water/steam transport.
4. Evaluate initial system transient.
5. Determine protection system response.
6. Incorporate protection system actions.
7. Evaluate the consequences (oxidation and fission product transport).

The first step is to identify the leak size and location as well as the plant operating state. These factors affect the rate of water ingress and the quality of the water/steam entering the primary circuit. This step is generally a result of the Licensing Basis Event selection process.

With the size and location of the leak known, the ingress rate can be calculated. The ingress rate is controlled by the thermal hydraulics of the secondary side. An equilibrium flow rate is quickly established following a brief initial ingress rate determined by the local conditions at the break location. The equilibrium flow rate is calculated based on the single or two-phase pressure drop in the leaking tube.

The next step is to evaluate how much of the water is transported to the core. This is determined by the quality of the water entering the primary circuit. Steam mixes readily with the primary coolant and is transported to the core. Liquid water entering the system may flash to vapor, but a significant fraction could remain as water. Some of the liquid water may collect in the bottom of the SG vessel, but some droplets might be transported through the system. Often it is assumed that all water is steam, both to be conservative and to simplify the analysis. This leads to step increase in water concentration at core every t_{loop} seconds where t_{loop} is the transit time around the primary circuit (on the order of 10 seconds).

An integrated transient model is used to calculate the overall system response based on calculated water ingress rate. Key results are the reactivity transient and resulting power response and the system pressure response.

The response of the plant protection systems can be determined from the initial overall system response. The system response will demonstrate which trip points are encountered. This determines which mitigation actions are initiated thus controlling the subsequent evolution of the event.

Once the overall response of the MHTGR system is determined, detailed evaluations are performed to assess the specific consequences. In particular, oxidation calculations determine the location and magnitude of the graphite oxidation. This supports the evaluation of structural margins, calculation of combustible gas generation, and potential mobilization of fission products caused by local oxidation. Fission product mobilization because of wash-off, oxidation, and other potential mechanisms is a key aspect of the consequence assessment as discussed in a subsequent section.

Generally, the major parts of the water ingress analyses are evaluated using an integrated system code or suite, since many of the resulting effects govern the subsequent evolution of the scenarios. For example, water transport to the core strongly impacts the transient power which determines when the reactor is tripped, oxidation determines combustible gas generation which in turn affects system pressure, potentially affecting the time of reactor trip and loop isolation, etc.

Previous analyses were generally preformed on a best-estimate basis. However, several conservatisms were introduced in the past MHTGR water ingress analyses in order to simplify the analyses and reduce the number of specific cases to be analyzed. For example, the assumed ingress rate of 5.7 kg/s was selected to bound, with margin, both the calculated rate for a main steam tube rupture and the rate for a feedwater tube rupture. Water is assumed to enter as 100% steam so that it is carried throughout the primary loop (corresponding to a leak at the main steam end of the SG), but in-leakage of a substantial fraction of the residual water in the SG is assumed (corresponding to a leak at the feedwater end of the SG). Various protection system actions are ignored on a case-by-case basis to expand the range of scenarios examined. For example in some cases, the high pressure trip is ignored, since if a reduced amount of water were to reach the core, the reactivity transient would be less severe and the high power-to-flow trip might not be reached. Nonetheless, the water is assumed to reach the core for the oxidation analysis. The specific scenario assumptions for the MHTGR analyses are discussed in Section 7.

6.2.3 Considerations for Other HTGR Designs

The general characteristics of water ingress events resulting from an SG leak are similar for all modular HTGR concepts. However, individual details will vary for each concept.

For example, the specific protection system logic varies between different concepts. The priority In the MHTGR is given to maintaining cooling capability, and the main loop is isolated only on strong indication of water ingress (e.g., high pressure or direct detection of moisture). In other concepts, the main loop is tripped for all abnormal indications.

Another difference is whether or not the concept includes a backup cooling system like the SCS. If the alternate concept does not include such a system, then isolation of the SG leads directly to loss of forced cooling. This will affect the subsequent outcome of the water ingress event, since higher core temperatures will affect oxidation rates.

Multiloop concepts, such as AREVA's steam cycle high temperature reactor (HTR) concept, may also have different water ingress scenarios, since the trip of one loop with an SG leak may not lead to the trip of all loops.

More significantly, PBR concepts may have other factors that affect the evolution of water ingress scenarios. First of all, the pebble fuel elements are not the same material as in standard nuclear grade graphite. Therefore, the oxidation characteristics during water ingress may differ somewhat. In addition, the reactivity balance in a PBR is more precise than in a prismatic reactor. The nominal excess reactivity, the degree of under-moderation in the core, and the balance of the control rod worths will all be somewhat different for each reactor design. These differences will affect the details of the resulting reactivity transient during a water ingress event.

Nonetheless, the primary features of a water ingress event and its mitigation are the same for all steam cycle modular HTGR concepts. When water ingress is detected, the reactor is tripped and the main cooling loop is isolated. Reactor margins accommodate the resulting consequences without violating fundamental design or safety limits.

6.3 Reactor Physics (GS)

6.3.1 Overview of HTR Core Physics Relevant to Water Ingress Events

The prismatic and pebble bed HTR designs are both under-moderated, and the fundamental core physics of the pebble bed designs will be very similar to that presented here for the MHTGR prismatic design. These under-moderated core designs imply that any additional neutron moderation will increase the system's reactivity. The positive reactivity change that occurs with water/moisture ingress is the combined effect of three phenomena:

- Less thermal neutrons are available for U-235 fission because of neutron absorption by hydrogen
- The neutron energy spectrum softens (less high energy neutrons), which increases the fission cross section and decreases resonance capture in U-238
- The reduced neutron leakage out of the core region decreases the ex-core control rod's effectiveness as well as having an effect on reserve shutdown system worth.

This reactivity increase can be offset by lowering the fuel to moderator ratio (e.g., lower heavy metal loading and U-235 enrichment), but fuel designers usually require higher loadings to lower fuel production per unit energy produced. The effect of steam ingress into the HTR-MODUL core for various heavy metal and enrichment loadings are shown in Figure 6.

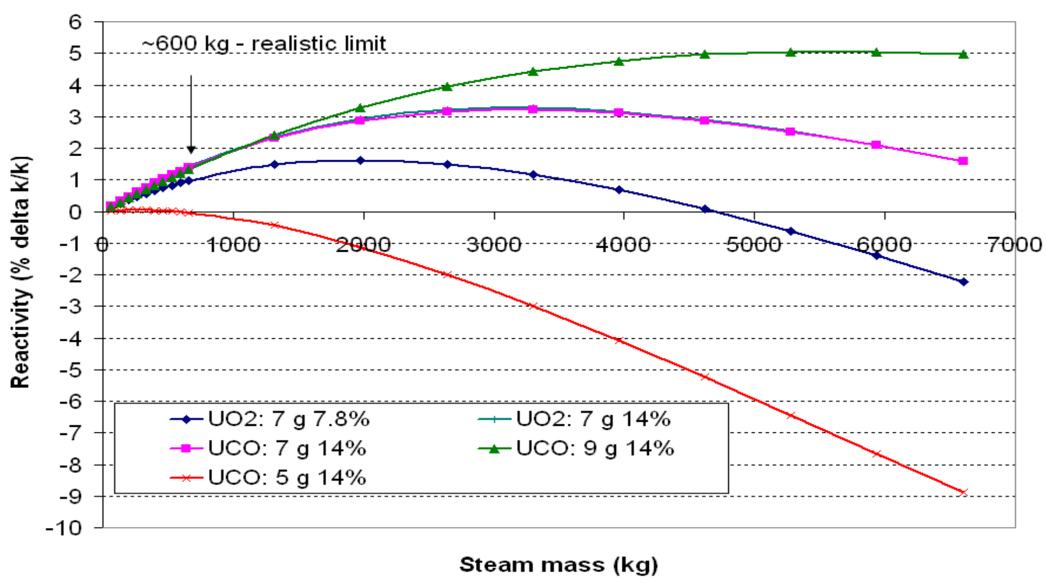

Figure 6. Reactivity change (%) vs. primary circuit steam inventory for five heavy metal and enrichment loadings in the HTR-MODUL (Strydom 2010).

The system reactivity behavior during water ingress events depends on several factors:

- System size, geometry, and the moderator material used
- Fuel (and to a lesser degree the moderator) temperature. This implies that the full spectrum between cold standby and hot operating core states should be investigated, or at least bounded.
- Fuel type (HEU, LEU, MOX, etc) and fissile/fertile mix.
- Burnup, fission product poisons, and reactivity control (burnable poison, control rods, RSS-as appropriate) status for a prismatic core reactor. The beginning of cycle MHTGR core with fresh fuel

and deep control rods will behave different than the same core at end of cycle when there is more plutonium present and the control rods are almost completely out of the core.

- Core thermal conditions under which the moisture ingress occurs.

These factors combine into the measurable temperature and density reactivity coefficients. Uncertainties in these coefficients are the main drivers for variations in the predicted power/temperature behavior during water ingress events. A specific problem area in HTR reactor physics is the validation of moisture ingress reactivity code predictions, since experimental data are not readily available at the elevated temperatures applicable to the HTR domain.

An important factor is the loss of control rod worth for HTR thermal systems. The decrease in shutdown margin needs to be taken into account, as well as ingress events at cold shutdown core states. This effect is shown in Figure 7 for the experimental HTR PROTEUS facility.

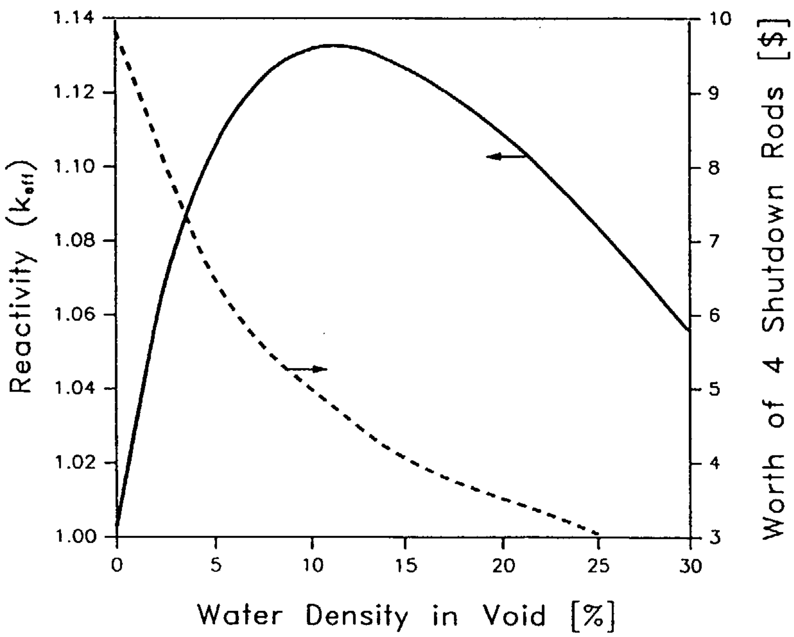

Figure 7. K_{eff} and rod worth changes vs. water density for the HTR-PROTEUS (IAEA 1983).

In safety studies, the dynamic effects of water ingress into the operating or shutdown core are usually of high importance. The 1986 MHTGR PSID (DOE 1992) included several DBA and BDBA versions of steam leaks and breaks, with and without safety equipment intervention (SCRAM, turbine trip, valve closures). The DBA scenario (a single SG tube rupture with reactor trip on a 1,200 ppm moisture detection, with a bounding leak rate of 5.7 kg/s) resulted in a peak reactivity increase of 0.196%, a power peak of 180% within 10s, and a maximum fuel temperatures rise of 48°C. The transient behaviors of reactor power and fuel temperatures are presented in Figures 8 and 9, respectively. Of the total 270 kg steam that entered the primary system, only 28% actually reacted with the core. It was also noted that uncertainties in the temperature and H_2 density reactivity feedback coefficients were responsible for most of the variation in the power behavior.

From a reactor safety point of view, the power and temperature increase for the typical HTR DBA water ingress event is therefore well within the material limits, and no fuel failure is expected to occur. Even the extreme BDBA scenarios' power density and rate of fuel temperature increase are bounded by TRIGA pulse experimental values, i.e., within the safety envelope of the TRISO fuel particle design.

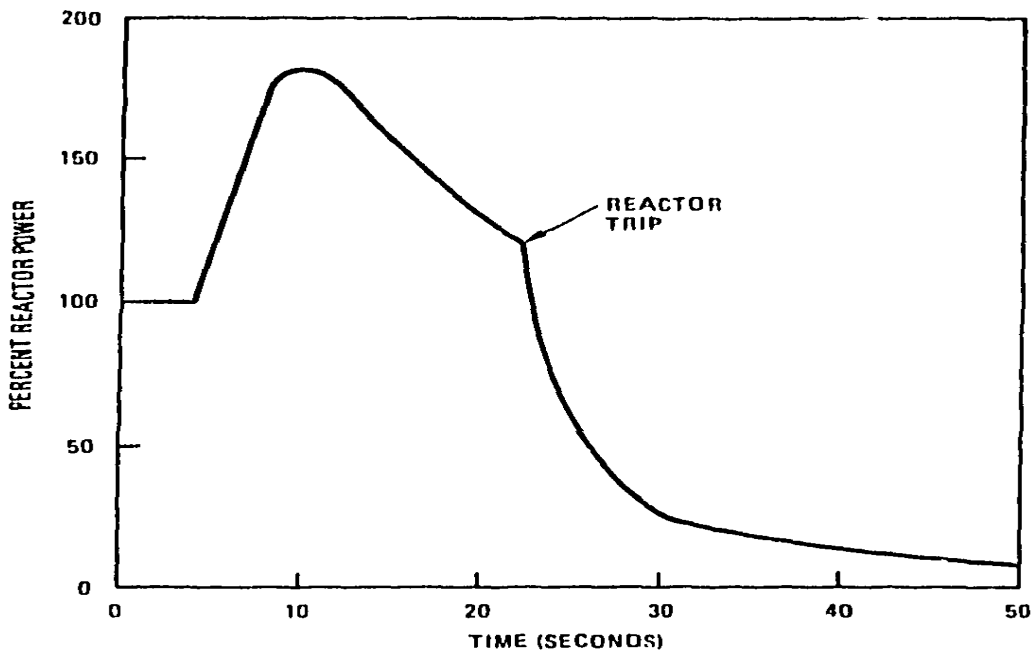

Figure 8. MHTGR water Ingress: PSID DBA reactor power (DOE 1992).

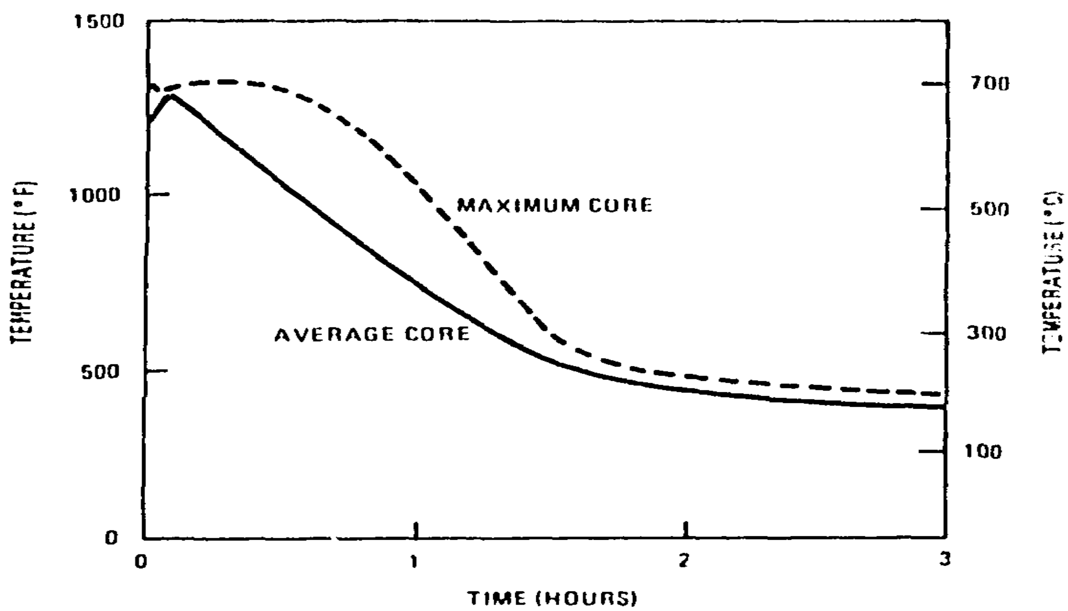

Figure 9. MHTGR water Ingress: PSID DBA fuel temperatures (DOE 1992).

6.4 Graphite (WEW)

Moisture ingress issues for graphite and carbon-based components falls into two main scenarios; acute moisture ingress events where large amounts of moisture are available over a short term, and chronic moisture exposure where low levels of moisture (ppm levels) are constantly available under normal operating conditions over the lifetime of the reactor. While oxidation thermodynamics for all potential reactions are all well understood, the kinetics remain difficult to predict because of the microstructure differences, amount of impurities within the graphite, and effects of partial pressure of oxidizing species. As a consequence, acute ingress events are better understood than the longer term chronic oxidation exposure to long lifetime components since the oxidation rate is controlled through relatively simple chemical kinetics. However, the mechanisms at work for either scenario depend upon the temperature and amount of oxidizing material available during reaction.

Generally, oxidation of nuclear grade graphite and carbon fiber components in 100% air environment starts to become significant at temperatures above 400°C. Different oxidizing environments or different graphite grades can change this temperature, but generally significant oxidation is considered to be possible at temperatures above 400–450°C. The rate of oxidation can be broken down into three different regimes which are a function of the oxidation temperature: (1) chemical kinetics controlled (450–600°C), (2) in-pore diffusion controlled (600–700°C), and (3) mass transport in boundary layer controlled regime (>700°C) as seen in Figure 10.

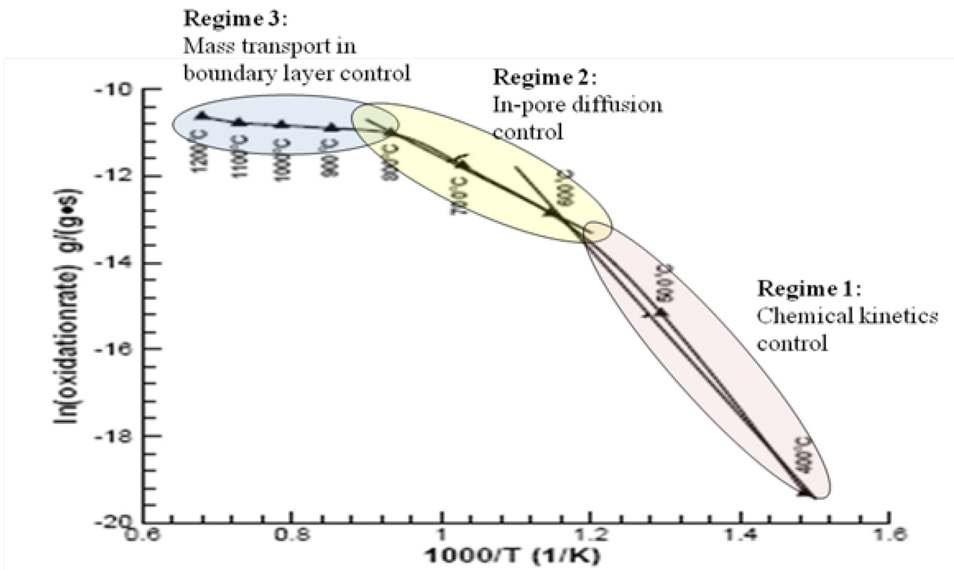

Figure 10. Graph showing rate of oxidation regimes.

However, it must be emphasized that the temperature where the different mechanisms control the rate of oxidation are highly dependent upon the type of graphite, the physical form/state of the carbon-based material (surface area to volume ratio or pore structure), the moisture content, and the oxidizing environment (Kim 2006). As a consequence, the transition from kinetic to diffusion to boundary layer controlled regimes must be determined for each specific graphite component.

6.4.1 Acute moisture ingress

For acute (short term) moisture ingress incidents the direct effect on graphite and carbon-based components will depend upon where the oxidation is likely to occur and the types of graphite employed. Specifically, if general oxidation over the entire core is assumed to occur after moisture ingress, the effects will be minimal. Either the amount of moisture introduced to the reactor core components will be minimal when compared to the entire mass of the reactor core, or the temperature will be reduced significantly over a relatively short period of time. Both assumptions will limit the overall impact of the moisture ingress events on the carbon-based core components.

However, if the oxidation is assumed to preferentially react on key components rather than a general, uniform degradation of the entire core, the effects of oxidation can be more significant. Specifically, preferential oxidation of core support structures, both for small and large moisture ingress events, can lead to a reduction in strength of the carbon materials and subsequent structural support issues. This is also true for carbon-based components such as the carbon-based insulation on the upper or lower plenum. Localized oxidation where the insulation is pinned or attacked to the plenum wall could weaken the joint and allow the insulation to become detached leading to coolant flow problems.

Clearly the assumptions made for each scenario are particularly important to the acute oxidation behavior of the graphite and other carbon-based components. The key assumptions made for both acute scenarios related to oxidation of core components are listed below (over-pressurization and structural integrity).

1. *System pressure increase*. The assumption is that all moisture introduced to the reactor core will react with the exposed graphite and carbon-based materials to form additional gases that will cause an increase the system pressure. Its importance on graphite component performance will be minimized because of the limited amount of moisture and the assumption that general oxidation occurs across the entire core. The knowledge base for this type of interaction is also high.
2. *Structural integrity*. Minimal impact on the structural integrity for general oxidation of the entire core. There will be a high to medium impact on the structural integrity of the core if key components, such as core support columns, are preferentially oxidized. For lower temperature oxidation in the Kinetic-controlled regime (< 600°C), the physics is fairly well understood and the impact can be calculated with a level of accuracy. For oxidation at higher temperatures in the Diffusion-controlled regime (> 600°C), the oxidation rates and physics are not as well understood. Additional R&D will be required to accurately calculate the impact of oxidation at these higher temperatures.

6.4.2 Chronic Moisture Ingress

It is anticipated that the coolant in an HTGR will have low levels of moisture present during normal operation. These are low levels of moisture (ppm) but the carbon-based materials will be exposed over a much longer time and at potentially higher temperatures. This chronic exposure of moisture potentially poses a significant impact on carbon-based components in the reactor core. All carbon-based components with long service lifetimes may be exposed to this slow oxidation phenomenon.

Chronic oxidation can impact in two ways: the slow degradation can lead directly to compromised components in the core and/or it could exacerbate some of the issues identified in the acute moisture scenarios by having previously compromised the core components.

At these higher temperatures and low concentration atmosphere, the oxidation mechanism is well within the diffusion-controlled regime (>600°C). Oxidation rates and the physics for diffusion-controlled oxidation are not well understood and models predicting the oxidation rate must be developed. These models must be especially accurate as they relate to the ability to track degradation of permanent or semipermanent graphite components and well within the design margin (i.e., PGX block oxidation in FSV was a significant material degradation issue). Additional R&D will be required to accurately calculate the oxidation rate, the effects on material performance and lifetime, and the mechanisms controlling the oxidation behavior at these higher temperatures and low moisture environments.

6.5 Fission Product Transport (JMK)

This discussion will only address fission product transport outside the fuel because fission product transport within the TRISO fuel particle was addressed in an earlier PIRT exercise (Morris, 2004). The following phenomena associated with moisture ingress may affect the transport of radionuclides within the reactor coolant system, and release from the reactor coolant system if the pressure boundary is not intact (e.g., a local break or opening of a pressure relief valve):

- *Mechanical removal of fission product deposits from reactor coolant system surfaces*. Liquid or steam may impinge on surfaces near the point of moisture ingress and mobilize fission products deposited or plated out during prior operation. If water droplets are suspended in the coolant during forced circulation, they may also impinge on surfaces remote from the ingress location.
- *Chemical reactions with fission products in the reactor coolant system*. Moisture ingress can significantly alter the chemical environment within the reactor coolant system, altering the molecular form of fission products to a more mobile state. This can affect both the release of fission products from the reactor coolant system and retention of released fission products within the reactor building.
- *Chemical reactions with structural graphite in the reactor coolant system*. Depending on local temperatures, moisture can react on structural graphite surfaces or diffuse into the graphite and react internally, releasing fission products sorbed on the surfaces or within the graphite.

6.6 Modeling and Experiments (YH, RRS)

The modeling needs and the specific experiments required to support validation of such modeling are rooted in the obligation of all model development and/or model validation efforts to ensure that the physics of the numeric models capture the dominant phenomena that are present in the various water ingress scenarios postulated for consideration. Thus the verification and validation (V&V) requirements for software intended for the design and analysis of HTGRs, whether prismatic or pebble-bed, are determined by the operational and accident envelopes of the reactor plant being considered. Specifically, as depicted in Figure 11, the V&V requirements can only be satisfied if the calculation envelope of the thermal-hydraulic software is demonstrated to either match or encompass the system operation and accident envelopes.

To ensure that analyses adequately cover the operational and accident domain of concern, the NRC has issued Regulatory Guide 1.203 to describe processes considered acceptable for development and assessment of evaluation models used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. In this context, the evaluation model includes all the numerical models used to calculate the nuclear system behavior, including system analysis and CFD software.

The range of applicability for a given software package is in part determined by the physics and the models contained within the software. Therefore, the V&V process must confirm the software physics models properly calculate the key phenomena over the entire range of conditions encompassed by the calculation envelope. Successful V&V can only be achieved if an adequate, high-fidelity data matrix and/or exact analytical solution set are available to benchmark the calculation results over the range of conditions that encompass the entire system envelope. Figure 12 shows the process used to ensure that the calculation domain and analysis tools are adequate for their intended purpose.

To define the scope of the required validation matrix, operational and accident scenarios that require analysis are first identified. The requirements of this step were satisfied, in a preliminary way in this effort, by making use of analyses and the experience of General Atomics.

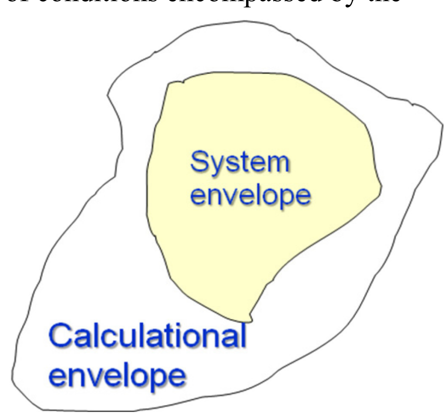

Figure 11. Venn diagram of system and calculation envelope.

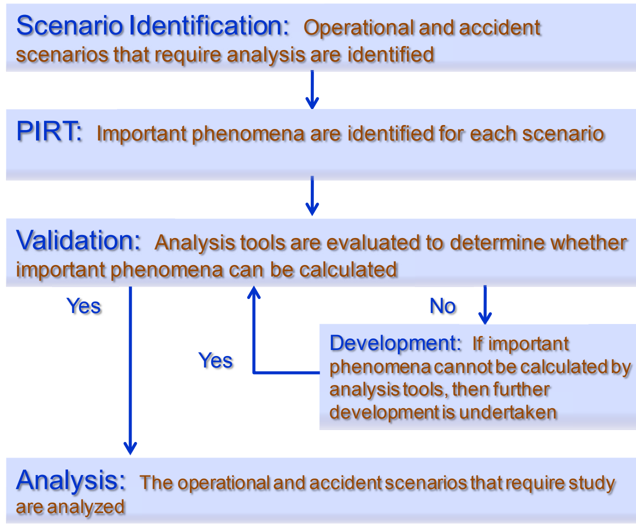

Figure 12. Process for evaluating calculation domain and analysis tools.

Each transient scenario is then evaluated by a group of experts using the PIRT process to identify and rank important phenomena associated with each scenario. The requirements of this step are being satisfied by the present document.

The analysis tools are then evaluated against experimental data to determine whether important phenomena can be calculated as shown in Figure 12. If not, then further development of the software must be undertaken. The process is concluded when it can be demonstrated that the software predicts the important phenomena over the entire calculation envelope that encompasses all scenarios in the operational and accident domains of interest.

The subsequent discussion in this section focuses on the types of models required and whether such models are available in existing software.

6.6.1 Summary of the Scenarios and the Relevant Phenomena that Must Be Modeled

In Section 6.1 the progression of the chosen water ingress scenarios are described and the specific events and progression are shown in Figure 13 and flow charts 1 to 4 in Figure 14. The flow charts of events given in Figure 14 illustrate that four dominant paths occur with the (a) timing of each relative to the others dependent on the timing of plant trip sequences, (b) quantity of water that enters the primary, (c) hardware configuration of the plant, (d) location of relevant sensors, and (e) geometry of the postulated tube break relative to the SG. Therefore, when evaluating the sequence of events in each of the four paths, it is important to remember that the occurrence of events such as 3d (see Figure 14 for the opening of the SG dump valves) may occur after 1e or 1j in Flow path 1.

Flow Path 3 describes the actions that may occur on the secondary side of the SG. The key action is the isolation of the secondary system such that feedwater and the steam line isolation valves are closed. The action of isolating the secondary defines the quantity of water that may or may not be available for transport to the primary system and thus may be available for reaction with the reactor structural and/or core graphite. Other important actions on the secondary side are centered on whether the secondary dump system is activated. Since the secondary dump system is not presently a safety-grade system, the presence of this system will probably not be accounted for during some scenarios required by the NRC. However, given that the secondary dump system is activated, choking will occur between the secondary system and the secondary dump storage tanks across the secondary dump valves.

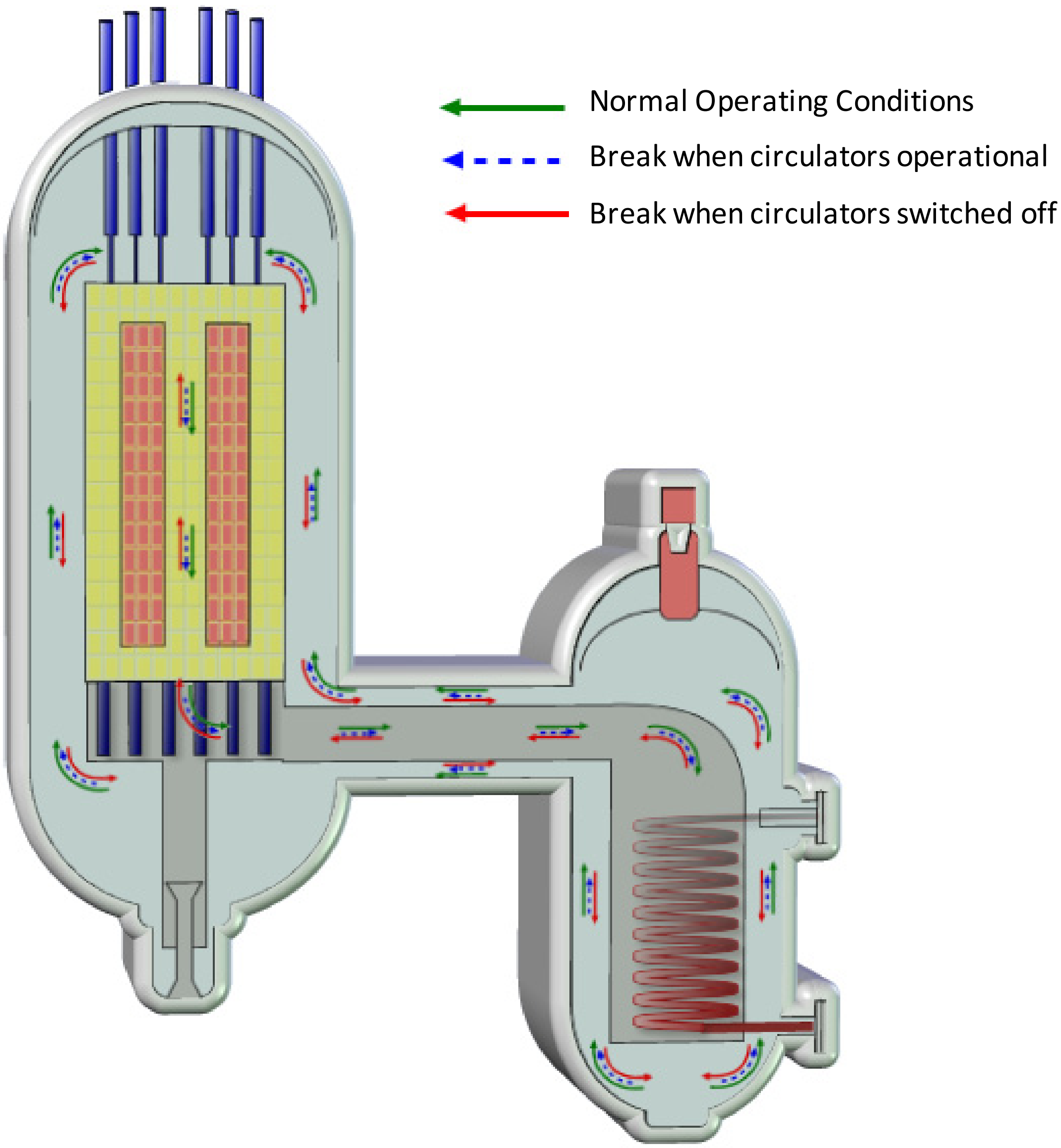

Figure 13. Comparison of normal operational flow directions with bulk water ingress flows following SG tube rupture while circulators are operational and switched-off.

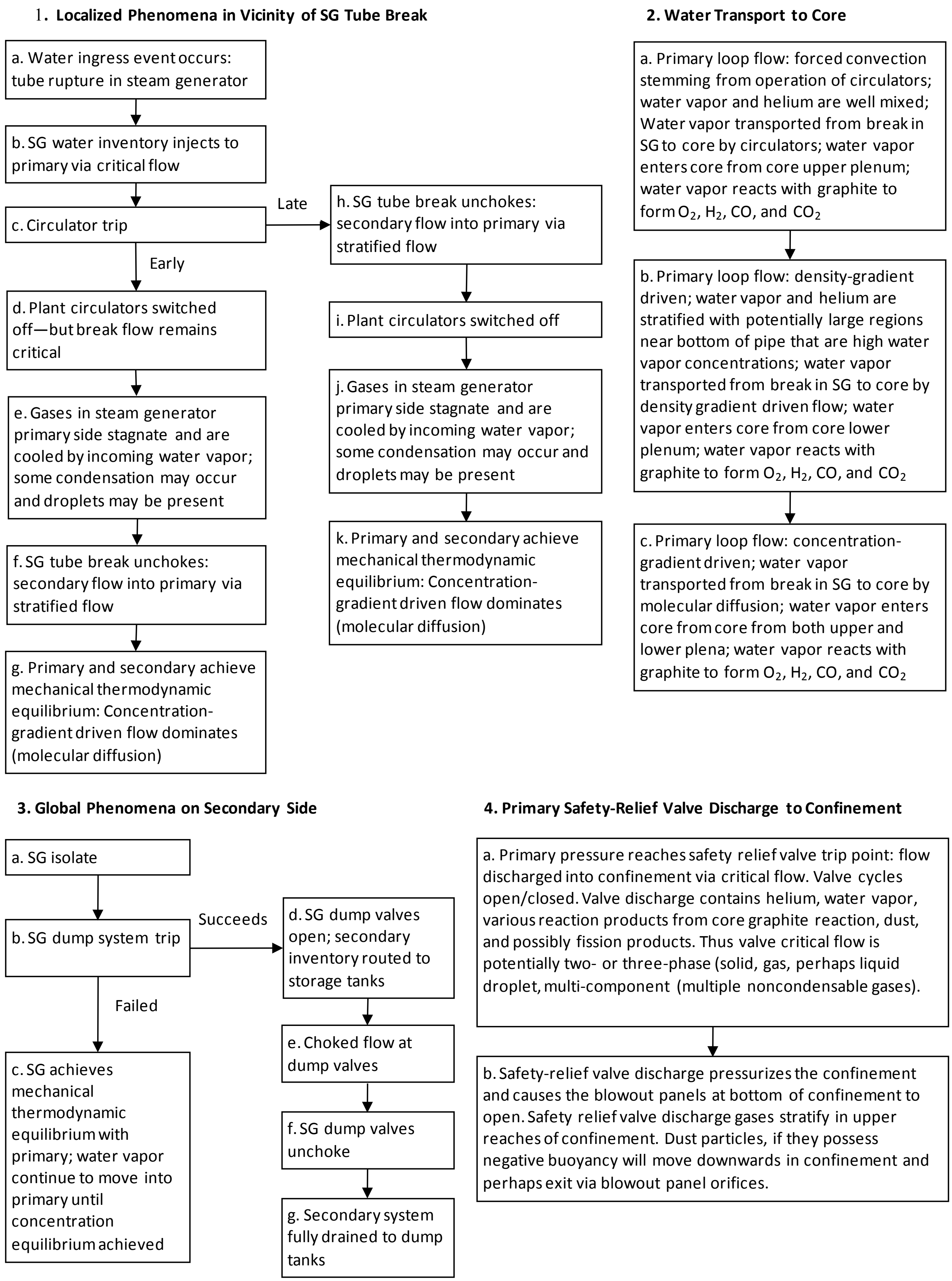

Figure 14. Flow chart: Qualitative Progression of Key Events in HTGR Water Ingress Scenario.

Flow Path 4 concerns whether or not the primary safety-relief valves are tripped open followed by the train events that will stem from pressurizing the reactor building (opening of blow-out panels, movement of various fission products, including dust, into the reactor building, etc).

In essence, each water ingress scenario is greatly influenced by the following boundary conditions and events. These events and boundary conditions are tied to the flow charts shown in Figure 14:

1. *Time period of the scenario where forced circulation of the primary system occurs via the action of the plant circulators (see 1c, 1i)*. This time period exists from the start of the scenario and ends when the circulators are switched off. Following this time period, the movement of the fluid within the primary is driven by natural circulation, i.e., density-gradient driven flow.
2. *Time period of choked water ingress from the secondary to the primary system (see 1b, 1d, 1e, 1f, 1h)*. This time period exists from the start of the scenario and ends when the secondary-to-primary flow unchokes. During this time period, the influence of the secondary system on the primary is defined by a choke plane between the two systems. Thus, the primary system exerts no influence on the secondary system and the choke plane represents the primary system boundary. Also, the choked flow enters the primary as a jet. If the tube rupture is high in the SG then the jet is first an incoming steam-liquid mixture that flashes into steam and only a short time later a steam jet. The jet itself may impinge on the wall of the primary pressure boundary, and part of the flow from the jet may condense or form droplets. Or the jet may be oriented such that it is aligned upward into the hot duct flow passage and is directed into the reactor vessel. The fluid mechanics may result in droplet deposits on solid surfaces followed by droplet resuspension as the scenario proceeds.
3. *Time period of unchoked water ingress from the secondary to the primary system (see 1f, 1g,1h, 1j, 1k)*. Termination of the choked water ingress phase, as described in Item 2, signals the beginning of the unchoked water ingress phase. Once the break unchokes, the interactive thermal-hydraulic system expands to include the primary and secondary systems influencing one another. That is, once the secondary pressure is low enough, the flow between the primary and secondary systems will be dominated, first by density-gradient driven flow at the break plane, then by concentration-gradient driven flow (molecular diffusion) at the break plane. A counter-current multiphase flow may thus be established as helium moves into the secondary while water (liquid/steam) continues to move into the primary via these two flow mechanisms. The thermal and momentum transport across the interface will play a role in the transport phenomena.
4. *Transport of the water to the core and interactions between the water vapor and the graphite supports and core (see 2a, 2b, and 2c)*. Transport of water from the break plane to the core will occur via forced convection while the circulators are running and via density-dominated and diffusion phenomena when the circulators are turned off. The core geometry (whether pebble-bed or prismatic), together with the water transport mechanism, is projected to have a great influence on the water distribution in both the graphite structural components and the core graphite. Forced flow, occurring when the circulators are active, will move water vapor quickly through the system. For a prismatic core, the water vapor will move through the core cooling passages and bypass regions and react with the graphite surfaces. For a pebble-bed core, the gas flow passages are considerably more complex creating a three-dimensional flow with circulation and swirl. Consequently, the potential to deposit the majority of water vapor near the inlet zone of the core is enhanced because of the large core surface area exposed to the flow and the centrifugal action of the flow as it moves in first one direction and then switches. Also during this phase, chemical reactions (oxidation) between the graphite and the water vapor occurs and leads to the production of a number of additional noncondensable gas products such as oxygen, carbon monoxide, carbon dioxide, and hydrogen.
5. *Isolation of the secondary system by closure of the feedwater and steam line isolation valves (see 3a through 3g)*. This event occurs when the trip signals are generated via the plant protection system and are received and acted upon by the plant control system. Following this event, the quantities of

secondary water mass, which may be transported into the primary, is established. Thereafter, this quantity of secondary water mass is available to be transported into the primary via some flow mechanism, unless a mitigating action is taken by the secondary dump system. That is, if the secondary dump system acts as designed, the majority of the secondary inventory isolated in the secondary system may be moved into plant storage tanks instead of being allowed to flow into the primary system. Conversely, if the secondary dump system, which is not a safety grade system, fails, then the entire secondary mass present when the SG was isolated is theoretically available for transport to the primary system.

6. *Initiation of the secondary dump system and its subsequent action to drain the secondary inventory to plant storage tanks (see 3b through 3g)*. Once the secondary dump system initiates, a choked flow condition will exist between the secondary and the secondary dump system as the secondary inventory is routed to the plant storage tanks. Thus, the conditions at the choking plane will define the movement of secondary inventory into the storage tanks. However, once the flow at the secondary to storage tank unchokes, a similar situation will exist as that described in Item 3. Thus, unless isolation valves are energized, the steam dump system storage tanks, secondary system, and primary system will all ultimately interact via density-gradient driven flow, once the pressure levels between the three systems are near equilibrium.

7. *Possible opening of safety-relief valves to discharge primary inventory into the reactor building (see 4a, 4b)*. As secondary inventory is transported to the primary system, if the primary pressure is increased sufficiently to open the primary safety-relief valves, then primary inventory is discharged into the reactor building periodically (or continuously if the safety-relief valves remain stuck-open). This event is of most concern (and thus importance) because with the opening of the safety-relief valves comes the potential discharge of fission products into the reactor building and the increased probability that fission products may be transported from the reactor building to the environment via the reactor building blow-out panels. Opening of the primary safety-relief valves introduces a third choke plane for evaluation into the scenario analysis that may entail critical flow evaluation considering dust, fission product gases, and even water vapor into the building structure. Consideration of these effects includes the resuspension of dust that was previously deposited throughout the system but resuspended as a large volume of gases exited the primary via the safety-relief valves.

8. *Forced convection (during safety-relief valve discharge) and natural convection of fission products and gas components in the reactor building (see 4a, 4b)*. A number of mechanisms and phenomena occur in the various parts of the reactor building following discharge of the safety-relief valves including: stratification of the lightest and the highest temperature gases in the upper elevations of the confinement, natural circulation of the gases within the confinement promoted by heating from the hot vessel walls, radiation heat transfer, and coupled heat transfer to the environment from the confinement gas space to the surrounding soil and to the environment via the RCCS.

6.6.2 Discussion of Key Thermal-Hydraulic Phenomena that Require Modeling

The phenomena that must be modeled have been identified in Section 6.6.1 in the context of the progression of a water ingress scenario stemming from a tube break in an SG. The phenomena of importance are present in four separate paths as portrayed in Figure 14: (1) localized phenomena in vicinity of the SG tube, (2) water transport to the core, (3) phenomena of importance on the secondary side, and (4) important considerations that stem from the safety-relief valve discharge into the structure. Selective phenomena, considered in the four paths identified above are discussed in this section.

Critical Flow. Although many critical flow experiments have been performed over the years, specific effects that stem from the critical flow that will occur in plant scenarios of interest, still render the information availability “medium.” It should be noted that in previous PIRT studies, such as that performed for the AP600, critical flow phenomena were judged as having only “low” knowledge.

There are actually three locations where critical flow will occur during the HTGR water ingress scenarios: the break itself, the safety-relief valve, and the secondary dump valves. The first two have specific unknowns that are important in this scenario. For example, critical flow from the secondary to the primary will occur as high pressure saturated water is injected into the lower pressure primary system. Although for some fraction of the blowdown, the secondary inventory will arrive in the primary in the form of steam as flashing occurs, the low knowledge areas concern: (a) the influence of the jet on the primary fluid and the structures the jet may impinge upon and (b) the action of the jet on stagnant helium following the circulator trip. The jet momentum may play a role in the movement of the water toward the core, portions of the jet may undergo condensation following shutdown of the circulator, the resulting water droplets may be deposited and later resuspended, so additional work is needed to clarify the importance of these phenomena and their subsequent effect on the transient progression.

The critical flow discharge through the safety-relief valve will likely act to resuspend condensed droplets that reside on the structural walls and resuspend dust that has accumulated in low velocity regions of the primary. The resuspended droplets and dust will be in the discharge in addition to the gas mixture that follows from oxidation in the core stemming from water vapor reacting with graphite.

Moisture levels in the primary. Because the primary may have localized regions that are cooler stemming from stagnant gases, environmental heat losses, and structural members such as the control rod drives that are generally kept cooler, sites exist where condensation will occur. Therefore, not all of the water will be transported into the core in the form of water vapor. In addition, depending on the torturous path that the water-ladened helium may be forced to take, certain regions will likely serve as regions for greater water vapor deposition than others. An example is a pebble-bed core: gases moving through a pebble-bed core must move to-and-fro while going from the inlet plenum to the outlet plenum. For these reasons, there may be preferential water-graphite reactions in some regions of some designs.

Stratified Flow. Once the circulators are switched-off, stratified regions that have lighter gases (helium) on the top and heavier gases (water vapor) on the bottom of plena and pipes will likely develop. Stratified flow will also likely occur at the tube break, since the helical-oriented SG tubes are not strictly vertical but instead are oriented at an angle with respect to the vertical. Hence stratified flow will be an important player in the progression of the water ingress scenario.

Gas production and gas distribution in the core and HTGR structures. The moisture distribution, and the phenomena that affect that distribution, will have an important effect on the corresponding production of noncondensable gases such as oxygen, carbon monoxide, carbon dioxide, and hydrogen. The ultimate primary source for the production of these gases will following from the materials interactions described in earlier sections of this report. However, the movement of these gases from their source sites to other locations in the primary system is a key area of potential research.

Decay heat. Present licensing analyses of light water reactors require that a 20% uncertainty level be assigned to the decay heat power level used as a boundary condition for licensing analyses. A robust R&D effort should be undertaken to ensure that the decay heat power level uncertainties are known adequately to ensure such a large uncertainty contribution is not assigned to corresponding HTGR analyses.

Heat transfer. The phase transition of liquid water to water vapor on the hot surface (coolant channel surface, pebble surface) brings entirely different flow conditions for the surface wall regions impinged upon by water droplets. The heat flux and the heat transfer coefficient decrease by a factor of 100 when the surface temperature increases from 600 to 1000 K. The evaporation of water causes a steam layer between droplet and graphite, which limits the heat flux from the surface to the droplet. This temperature gradient induces thermal stresses. This heat transfer phenomena should be assessed.

Fission product transport. Graphite particles and fission products transported in gas flow with the presence of vapor (moisture) need to be assessed. Deposition and resuspension of fission products should be validated. The fission product transport has a significant impact on potential dose rate.

6.6.3 Discussion of Models Required to Perform Water Ingress Scenarios:

Numerical models sufficient to capture the physics for all the phenomena listed above are key. With respect to the transport of the water vapor and water droplets to the core and graphite structural members, the numerical models used in software should have the capability to model momentum-driven flow, density-gradient-driven flow, and concentration-driven flow. Within the flow fields, the numerical models should be able to capture the flow regime pattern and transition such as the formation of stratified regions where one gas concentration predominates (for example large helium concentrations in the tops of a pipe and large water vapor concentrations in the bottom of the pipe). Also, water droplets that may be deposited and then resuspended are important. Presently, the numeric tools that may be able to meet these requirements are computational fluid dynamics (CFD).

Systems analysis software (RELAP5, TRACE, GAMMA+) are generally used for one-dimensional analyses. However to compensate for the shortcomings of these tools, efforts are underway to couple them with CFD codes such that selected regions of the system that should be modeled—where stratified flow, for example, may be present—may be modeled adequately for the purposes of studying water ingress scenario.

6.6.4 Discussion of Key Experiments Required for V&V

The strategy underlying the design and construction of experiments to study specific phenomena is derived from first a sound scaling analysis that isolates the key phenomena in the governing equations that must be studied—usually in the form of dimensionless numbers. Establishment of the parameter ranges of interest for the prototypical system defines the specific regions of interest that must be studied in the desired experiments. Thus, the experiment designs begin with the variables and thermodynamics ranges of interest as defined by focused scaling analyses and then proceed by specifying experiments that are both integral (meaning that all phenomena are represented including the interactions between them) and separate-effects (meaning experiments designed to study specific phenomena undisturbed by other phenomena) experiments. The experiment design methodology calls for various experiments that focus on the key phenomena at different scales. The reasoning behind this approach is that when data are recorded at various scales, if the data correspond to one another via the scaling laws, the correspondence is a vehicle for reducing the uncertainties that stem from recording data in any scaled experiment.

In cases where more experiments are deemed necessary, a combination of integral and separate-effects experiments are called for to better understand the water ingress scenario.

7. EVALUATION TABLES

The panel selected two accident scenarios and one normal operation condition that were considered to be the most important to the evaluation of the effects of water ingress on the MHTGR. The results are summarized in the following sections. Each assessment area is discussed and summarized in a table. The tables were generated during the panel discussions. Additional review of the tables was performed by the panel members, and individual contributions are also shown in the tables. These individual contributions are identified with each panel member's initials (see Appendix A for the list of members).

7.1 Tables for Water Ingress Accidents

It was recognized that many of the phenomena involved with the accident evaluations were important, to varying degrees, in a variety of different postulated accident or event scenarios. A prevailing challenge was that many of the major design features of the NGNP system being evaluated had not yet been established. As noted previously, however, the panel chose to use the MHTGR (1986 design) as the reference design, with the assumption that it will likely be very close to that of the leading candidate.

One important consideration by the panel was an assumption of potential leakage paths for fission products and other gases escaping from the primary system. In cases of long-term (major) steam ingress, opening of the pressure relief valve(s) would cause releases to the reactor building and beyond. Regarding the potential for additional leakage because of primary system breaks and possible subsequent air ingress following a total depressurization, the consensus of the panel was that an SG tube rupture and a depressurization accident are both very unlikely events without credible common-cause factors that would link them. Hence, it was assumed for this assessment that relief valve openings were the only way material would get out of the primary system.

In considering postulated accidents, it is useful to refer to a typical protection systems logic design. In the case of the MHTGR, the protection system may be assumed to be similar to that shown in Figure 15, which was in a General Atomics report evaluating SG design options. The red lines refer to safety-grade operations, while the blue lines refer to nonsafety actions. In some cases (including examples used in the tables in this section), it is assumed that the SG dump is nonsafety grade, while SG isolation is a safety-grade function, not nonsafety as shown in Figure 15. Such design options would depend on sensitivity studies showing the differences in results with and without the isolation function working.

To evaluate the major phenomena related to steam/water ingress accidents, the panel decided to evaluate two postulated sequences that had been previously analyzed in detail for the MHTGR. These were judged by the panel to cover most of the significant features and phenomena related to steam/water ingress accidents in general. In the terminology used in the NRC draft Safety Evaluation Report (NUREG-1338), the MHTGR PSID (DOE 1992), and the MHTGR Plant Transient Analysis Report (DOE 1987), they are referred to as DBE-6 (an SG tube rupture, classified as a Design Basis Event), and SRDC-6 (an SG tube rupture with only safety system mitigation, classified as a Safety-Related Design Conditions Event). The conditions and assumptions for these two events are summarized in Table 6, which was part of the presentation by panel member Lew Lommers. The shaded regions in Table 6 represent the conditions or actions that are significant to the progress of the event. SG isolation is activated upon a loop trip.

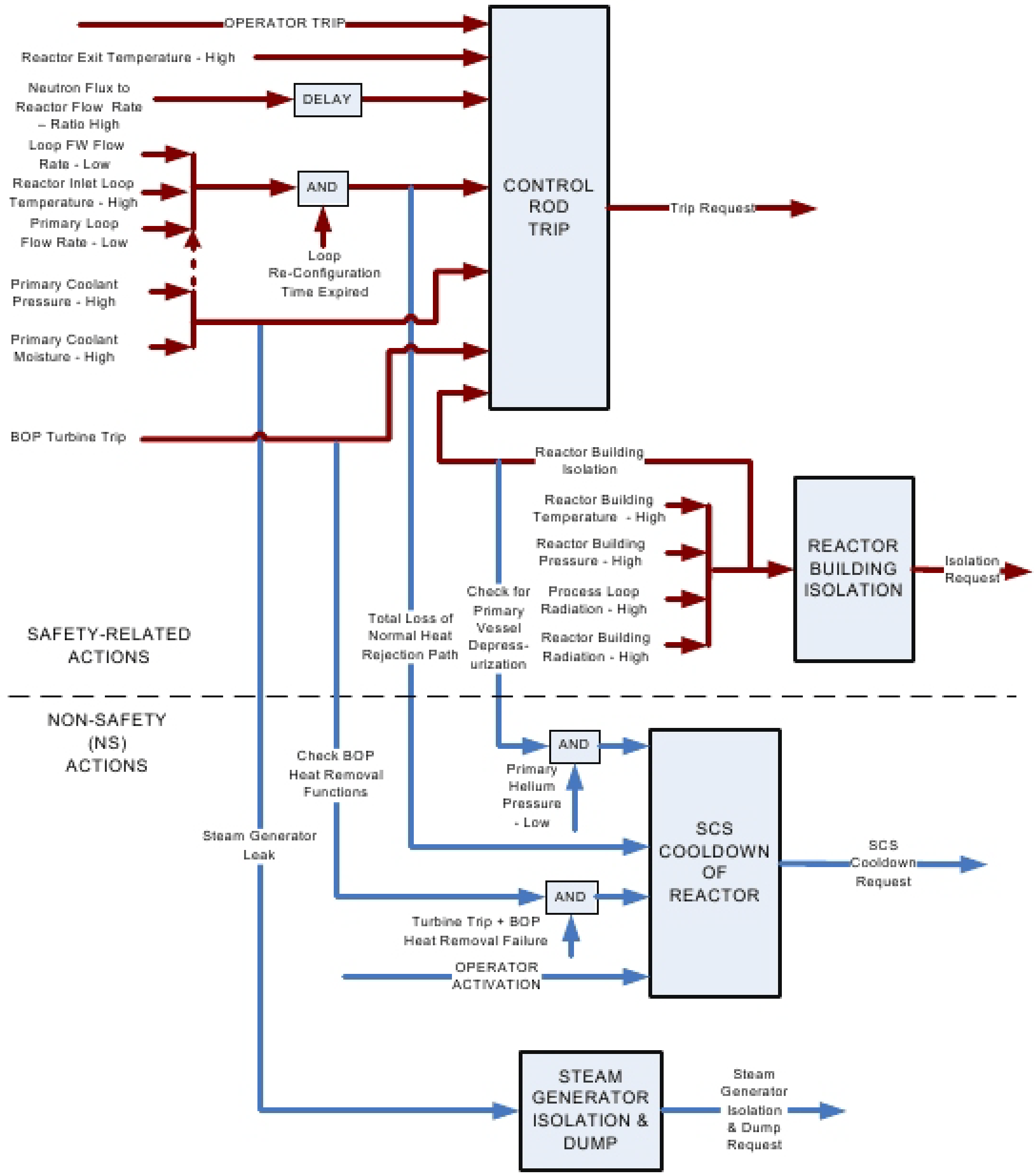

Figure 15. Example of protection logic for the case of the SG in the primary system (Labar 2008).

Table 6. Summary of MHTGR SG water ingress event analyses.

Event	Ingress Rate (kg/s)	Moisture Monitor	High Power Trip	High Press. Trip	Loop Trip Initiated	SG Dump	Cooling	Total Ingress (kg)	Total Oxidation Fraction	Relief valve open?	Dose at EAB
SG leak (nominal)	0.05	Detected	NA	NA	At 390 s	Yes	SCS	18	Minimal	None	None
Tube rupture (nominal)	5.7	Detected	Assume not reached	NA	At 32 s	Yes	SCS	272	Low	None	None
DBE-6 – tube rupture	5.7	Detected	No credit taken	NA	At 30 s	Yes	SCS	270	Low Bottom refl.: 2×10^{-4} avg 9×10^{-4} max	None	None
DBE-7 – tube rupture w/SCS failure (DCC)	5.7	Detected	No credit taken	Yes	Yes	Yes	RCCS		$5.2\text{x}10^{-4}$	1 cycle assumed at 10 hr*	$4.66\text{ x}10^{-4}$ Rem (whole body)
DBE-8 – SG leak w/moisture monitor failure	0.05	Failed	NA	Yes	4.8 hr	Yes	SCS	841	Acceptable Core: 1.3×10^{-3} Bottom refl: 1.6×10^{-3} avg 6.1×10^{-3} max	None	None
DBE-9 – SG leak w/dump failure	0.05	Detected	NA	NA	380 s	Yes – fails open	SCS	18	Low	None	None
SRDC-6 – tube rupture (only safety system mitigation)	5.7	Failed	Yes RT @ 8s	Yes RSC and Loop Trip	326 s	No	RCCS	4000?	Acceptable	3 cycles (fails open on third)	0.045 Rem (whole body)

* Calculated peak below set point.

7.1.1 DBE-6—SG Tube Rupture with Credit Taken for Moisture Monitor Operation

The first event considered by the panel was DBE-6 from the MHTGR PSID (DOE 1992). This event is a moderate SG tube rupture. Following the offset tube rupture, plant systems are generally assumed to behave as expected. The two key plant responses that govern the progression of the accident are (1) isolation and dump of the SG, and (2) use of the SCS to gradually cool down the reactor following trip of the main heat transport system. The main analysis assumptions are:

- 5.7 kg/s steam entering primary circuit from offset SG tube rupture
- High power trip not activated (nonstandard assumption)
- Detect high moisture
- Trip reactor, SG isolation and dump
- SCS provides heat removal.

The DBE analysis assumed a bounding water ingress rate of 5.7 kg/s from the offset tube rupture. The analysis assumes that incoming water is in the form of steam, which has a more immediate effect on pressure and is more readily transported to the core.

Water vapor increases reactivity, resulting in a power excursion. However, the PSID analysis ignores the reactor trip on high power-to-flow ratio. This assumption delays the reactor trip until the moisture is detected. Even so, the brief power excursion only increases fuel temperatures slightly, and the effect of the reactor trip delay is not significant.

When high moisture is detected, the reactor is tripped using control rods. The high moisture signal also triggers isolation and dump of the SG (and main circulator trip). At the completion of SG isolation, the steam and water dump system dump valves are opened to drain most of the water inventory from the SG to the dump tank. This investment protection action minimizes the potential for subsequent core damage through oxidation and hydrolysis.

Following the main loop shutdown, the SCS is started after an assumed 5-inute delay. The SCS cools the reactor over the course of a few hours.

System pressure is the main FOM considered in the panel's evaluation, since it determines whether or not there will be a release (because of opening of the primary system relief valve). This has the strongest bearing on overall criteria of dose to public and workers.

The consequences of DBE-6 are relatively minor. The SG isolation and dump minimizes pressure increase because of sustained water ingress. Furthermore, operation of the SCS prevents any significant pressure increase because of system heat up. Therefore, the relief valve does not open, and there is no release to the reactor building or environs. The isolation and dump also minimizes the quantity of water available to oxidize reactor graphite. Predicted graphite oxidation as reported in the PSID is minor (0.02% average, 0.09% local maximum).

The phenomena generated by the panel for this event is presented in Table 7.

Table 7. Assessment of DBE-6: single SG tube hot end break with nonsafety system mitigation.

Item No.	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for Importance	Rationale for knowledge	Recommended Action
1	Flow through break	system pressure [Pressure is a less important FOM for DBE-6 scenario, since moisture detection provides early action. In general pressure is of interest because of timing of HP trip (backup) and relief valve opening (if ingress not limited and/or SCS not operational): LJL]	H	H [M YH, RRS]	Determines rate of water influx into primary system	Critical flow of water is well understood initially, hydraulics prior to dump are understood. [Extensive studies performed using experimental data in the various international standard problem validation tests resulted in the a range of discharge coefficients between 0.6 to 1.8 for typical critical flow models used in systems analysis numerical models (Ransom-Trapp and Henry Fauske). Such a large range of answers indicates a fundamental lack of understanding that still remains, even after all the R&D that has been performed in this phenomena area. The problem postulated for the scenario addressed in this report is more complicated in that it flows through a tube with changing frictional characteristics that stem from the flow in the secondary flashing while moving through the tube and thus increasing the influence of the two-phase friction as a function of location and time (considering the change in heat input to the flow as a function of time. Thus the specific knowledge required to clearly model this phenomena is either low or medium at best. YH, RRS]	No additional action required [It is noted that the action of the jet and the break flow orientation may have an unknown influence on the progression of the scenario because of jet forces and the direction of the jet may result in additional tubes failures. Additional R&D specific to quantifying the uncertainty of this phenomenon should be performed. YH, RRS]

Item No.	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for Importance	Rationale for knowledge	Recommended Action
2	Moisture level in primary	System pressure	H	H [M YH, RRS]	Trips the reactor, dumps the SG	Part of flow stream [Although it is true that the moisture entering the primary is part of flow stream, the nonuniform concentration distribution and the fraction of the moisture which condenses in some parts of the primary, e.g., the control drive mechanism, influences (a) timing of the moisture detector system and (b) the quantity of the vapor transported to the core. YH, RRS] [Knowledge is high if the rate and amount of ingress is known. SJB]	No additional action required
3	Transport phenomena to moisture monitor	System pressure	H	H	Trips the reactor, dumps the SG	This phenomenon addresses the transport of the moisture through the sampling tube to the instrument. The physics of this transport are well understood	No additional action required
4	Moist monitor instrument response	System pressure	H	H	Trips the reactor, dumps the SG	Technology is understood or can be demonstrated as part of the procurement of the instrument	No additional action required
5	Gas production because of moisture reaction with graphite	System pressure	L	H	In context of gas generation because of reaction w/ graphite, relatively small amount of water, [For DBE-6 gas generation has negligible contribution to pressure, since ingress is limited and SCS cools core. LJL] [Minimal impact on graphite components because of small amount of moisture ingress compared to overall core mass. WEW]	Known physics	No additional action required

Item No.	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for Importance	Rationale for knowledge	Recommended Action
6	Transport of moisture to core	System pressure	L	H	Given assumption that the SG is dumped, not that much water enters the primary system. [Also, circulator is tripped so transport to core is significantly reduced as circulator coasts down (significant only for 10's of seconds following detection). SCS may transport limited moisture to Rx outlet plenum as it cools system down. LJL]	Straightforward thermal/hydraulic (T/H) [There is not much moisture to transport because the SG dump system discharges most of the secondary inventory. YH, RRS]	No additional action required
7	Dump system hydraulics	System pressure	M	H	Amount of water remaining in SG which may eventually get into system	Exiting codes, such as RELAP, are adequate to analyze the system	No additional action required [The entire secondary dump system, including the holding tank and the system drains, should be modeled to characterize the SG dump system behavior and to ensure that potential condensation-induced water hammer doesn't occur as well as damaging condensation-induced oscillations. A systems analysis code should be able to model the secondary hydraulics. This action should be performed. To quantify the importance and behavior of the system. YH, RRS]
8	Reactivity insertion	System pressure	L	H	The short duration and magnitude of the reactivity insertion results in small fuel temperature rises that do not affect the helium pressure significantly.	Reactivity insertion (via hydrogen moderation) and counter-mechanisms (Doppler temperature feedback) are well understood, and even simplified point kinetics models should not result in significant differences in system gas pressure results.	No additional action required

Item No.	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for Importance	Rationale for knowledge	Recommended Action
9	Heat removal by SCS	System pressure	H	H	Cools primary system	Basic system T/H	No additional action required
10	Decay heat [YH, RRS]	System pressure	L	L	Although this is a primary boundary condition, it is known that the fraction of the energy provided to the system by decay heat is inconsequential compared to the other sources.	For LWRs the decay heat curve used for licensing calculations is ANS +20%. The uncertainty of the decay heat must be reduced to a reasonable value.	No action required. [Decay heat is not a crucial parameter for this accident sequence since it is terminated early; however, it is for SRDC-6 (long term) events-SJB]

7.1.2 SRDC-6 – SG Tube Rupture with Only Safety System Mitigation (No Credit Taken for Moisture Monitor Operation)

The other event considered by the panel was SRDC-6 from the MHTGR PSID (DOE 1992). This event also begins with a moderate SG tube rupture. However, the subsequent analysis is more conservative, since it credits only safety-related systems. This affects both detection of the water ingress and mitigation of the event.

The main assumptions of the PSID SRDC-6 analysis are:

- 5.7 kg/s steam entering primary circuit from offset SG tube rupture
- Reactor trip because of high power-to-flow ratio
- High moisture NOT detected
- High pressure signal initiates SG isolation
- No SG dump
- No SCS decay heat removal
- Relief valve fails open (after ~3 cycles).

The SRDC analysis assumed the same bounding water ingress rate of 5.7 kg/s from the offset tube rupture. The analysis assumes that incoming water is in the form of steam, which has a more direct effect on pressure and is more readily transported to the core.

Water vapor increases reactivity, resulting in a power excursion. The reactor is tripped on high power-to-flow ratio. The brief power excursion only increases fuel temperatures slightly. Initially, the normal MHTGR reactor trip sequence is followed. Cooling continues on the main loop. Controlled cooldown of the core begins. Water ingress continues, since the SG is still being used for cooling.

Eventually, the high pressure trip set point is reached because of the ongoing water ingress. This initiates circulator trip and SG isolation. The high pressure signal also activates a backup reactor trip using the reserve shutdown system (although the reactor is already subcritical). However, the SG is not drained by the steam and water dump system, since that is not a safety system.

The SG isolation delay results in much more water initially entering the primary circuit (~1,860 kg per the PSID analysis). Moreover, since the SG is not dumped, additional water continues to enter from the isolated SG (~2,200 kg additional steam per the PSID analysis). This larger ingress leads to higher system pressure and increased oxidation.

Following the main loop shutdown, the SCS is assumed to not operate, since the SCS is not a safety-related system. Therefore, a loss of forced circulation occurs (initially pressurized).

The primary coolant pressure increase is significantly higher for this event, due primarily to the large volume of water entering the system. Soon after the high pressure reactor trips, the primary relief valve cycles, temporarily reducing the system pressure. However, the system pressure will increase again gradually, because of both graphite oxidation product buildup and the heat up resulting from the loss of active cooling. This causes the relief valve to cycle a few more times. For the PSID analysis, it is assumed that the valve fails open on the last cycle.

The resulting depressurization leads to discharge of the radionuclides in the primary coolant. It also minimizes natural circulation within the reactor, leading to higher peak fuel temperatures. No subsequent air ingress is assumed after the system pressure is reduced to atmospheric.

The consequences of SRDC-6 are more significant than for DBE-6. Significantly more water enters the primary system so more oxidation occurs. The relief valve cycles and is assumed to eventually fail open, resulting in a release of radionuclides. However, the oxidation is acceptable, and the total environs dose is well within limits, based on the PSID analysis.

The phenomena table generated by the panel for this event is shown in Table 8.

Table 8. Assessment of SRDC-6: single SG tube hot end break with only safety system mitigation.

Item No.	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for importance	Rationale for knowledge	Recommended Action
1	Flow through break	System pressure [System pressure is important because it determines when high pressure trip set point is reached (which leads to SG isolation) and it determines if relief valve will open (creating a possible release path).- LJL]	H	H [M - YH, RSS]	Determines rate of water influx into primary system	Initial critical flow of water is well understood, hydraulics prior to dump are understood [See rationale in DBE-6. - YH, RSS]	No additional action required
2	Moisture level in primary	System pressure	H	H [L - YH, RRS]	Determines when SG is isolated	Part of flow stream [knowledge is high if the rate and amount of ingress is known – SJB] [Although it is true that the moisture entering the primary is part of flow stream, the nonuniform concentration distribution and the fraction of the moisture which changes phase (condensation, evaporation, etc) and other phenomena such as deposition and resuspension in some parts of the primary, e.g., the control drive mechanism, influences (a) timing of the trip and (b) the quantity of the vapor transported to the core. YH, RRS]	No additional action required

Item No.	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for importance	Rationale for knowledge	Recommended Action
3	Gas production because of moisture reaction with graphite	System pressure	M	H	Steam [and reaction products are – JMK] significant contributors to system pressure [reaction products are a lesser secondary contributor to system pressure, particularly during initial phases of event which lead to high pressure trip – LJL] [Minimal impact on graphite components because of small amount of moisture ingress compared to overall core mass – WEW]	Known physics	No additional action required
4	Transport of moisture to core	System pressure	M [H – YH, RRS]	H [L – YH, RRS]	Affects high power trip time and increase of gas generation	Straightforward T/H [during forced circulation prior to circulator trip - JMK] [The transport mechanism is complicated by the recognition that a certain, unknown fraction, of the effluent from the secondary will be in liquid form—perhaps droplets. The behavior of the liquid droplets introduces a complexity that is difficult to deal with, especially considering that some droplets may become attached to nearby surfaces but later may become resuspended as the scenario progresses. - YH, RRS]	No additional action required [Additional R&D to examine the facets and possibilities of these phenomena influencing the scenario trajectory should be performed. – YH, RRS]

Item No.	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for importance	Rationale for knowledge	Recommended Action
5	Reactivity insertion	System pressure	M	H	The short duration and small magnitude of the reactivity insertion results in small fuel temperature rises that does not affect the helium pressure significantly. [The reason importance is M is not concern over the energy addition and resulting temperature rise. Rather, it is the question of whether or not the power excursion will be large enough to activate the reactor trip on high power. If the power pulse does not pass the set point, the reactor will not be tripped until some later limit is violated. – LJL]	Reactivity insertion (via hydrogen moderation) and counter-mechanisms (Doppler temperature feedback) are well understood, and even simplified point kinetics models should not result in significant differences in system gas pressure results.	No additional action required
6	[Decay heat production – GS] and long term heat transfer within primary circuit	System pressure	H	M [L-YH, RRS]	Although the short term reactivity pulse has an insignificant effect on the system pressure, the longer term (up to 100 hours) heat up of the core through decay increases the system pressure ~20% (from 6 MPa to around 7 MPa).	1. The presence of a helium-steam gas mixture complicates pressure and heat transfer calculations. 2. Long term decay heat models have a significant associated uncertainty.	Obtain experimental data for code V&V; assess capabilities of existing system codes
7	Reactor power (decay heat) [GS recommends delete this line]	System pressure	H	M [L-YH, RRS]	High decay heat drives thermal transient [Delete “high” – LJL]	Long term decay heat may not be well understood [Pressure transient primarily important early in transient, before start cycling relief valve, so not clear why long-term decay heat is the issue. – LJL]	[Adequate understanding of decay heat is required for ALL passive decay heat removal scenarios. This will require review of the current neutronic/ burnup methods to determine applicable uncertainty margins. – LJL]

Item No.	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for importance	Rationale for knowledge	Recommended Action
8	Heat removal by RCCS	System pressure [Structural integrity – SJB]	M or L	M	1. Impact of RCCS is negligible in first 10 hrs - Importance is L. [RCCS has modest contribution to overall system temperature. Overall system temperature determines system pressure. – LJL] [The disagreement stems from what the influence of the system would be if RCCS were not present. The RCCS seems to make an important contribution that, if not present, would result in a different system behavior. YH, RRS] 2. RCCS is [more – LJL] important to [~~short~~ long – LJL] term heat rejection - M [Relief valve opens long before 10hr, so closer to L than M – LJL] [RCCS operation mainly impacts long term reactor vessel temperatures – SJB]	RCCS performance uncertainties are well documented	Need more design detail before recommendation can be made
9	Temperature distributions in the core and support structures	Structural integrity	H	M	Affects the oxidation rate and fission product release rates through diffusion	Local temperature distributions	Improvement of analytical tools for calculating local temperatures [The dominant contributors to calculating long-term local temperatures are graphite thermal conductivity properties (prismatic cores, depressurized) [SJB] and decay heat. Graphite property variation is inherent so it must be covered by margin. Friction factor data for extremely low flow could be improved (Re~20). – LJL]

Item No.	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for importance	Rationale for knowledge	Recommended Action
10	Graphite oxidation of core support at low temperature	Structural integrity	H [M because of low temperatures – WEW]	M	Assumes attack of core support column [Scoping analysis and comparison to design margins would allow better estimate of importance. – LJL] [“Low” temperatures implies Chemical Kinetic controlled oxidation regime which has lower overall oxidation rate. – WEW]	Do not have a large data base for moisture [Oxidation rate assumed to be other than strictly low temperature Chemical Kinetic controlled oxidation. Limited data in this regime. – WEW]	Obtain more data
11	Graphite oxidation of core support at high temperature	Structural integrity	H	L	Assumes attack of core support column [Scoping analysis and comparison to design margins would allow better estimate of importance. – LJL] [Specific attack on support columns implies diffusion controlled oxidation rate. – WEW]	High for low temperatures. Low for high temperatures [Diffusion controlled oxidation not well understood. Must be studied for each type of graphite microstructure. – WEW]	Obtain more data [Diffusion and mass-transport mechanisms as a function of graphite microstructure need to be determined. Langmuir-Hinschelwood equation parameters – WEW]
12	Moisture distribution in primary system	Structural integrity	H	M	Assumes attack of core support column [Scoping analysis and comparison to design margins would allow better estimate of importance. – LJL] [Specific attack on key components with oxygen partial pressure implies diffusion controlled oxidation rate. – WEW]	Mass transport analyses are complex [Diffusion controlled oxidation not well understood. Must be studied for each type of graphite microstructure. – WEW]	Improvement of analytical tools for calculating moisture distribution [Detailed evaluation of scenario phases to determine potential moisture transport vectors would be extremely valuable to determine which analytical methods are needed. – LJL] [Diffusion and mass-transport mechanisms as a function of graphite microstructure need to be determined. Langmuir-Hinschelwood equation parameters – WEW]

Item No.	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for importance	Rationale for knowledge	Recommended Action
13	Power transient because of reactivity insertion fails fuel particles	Fission product release	L	H	The magnitude and duration of the reactivity insertion is too small to result in significant fuel temperature rises or fission product gas buildup trapped in the kernels.	HFIR irradiations were conducted at 10 times the max expected power level	No additional action required
14	Attack on failed particles	Fission product mobilization			Covered by NUREG/CR 6844		See NUREG/CR 6844
15	[Release of fission products sorbed in fuel element graphite or fuel compact matrix – line added by JMK]	Fission product mobilization			Covered by NUREG/CR 6844		See NUREG/CR 6844 [Mechanisms for releasing fission products and diffusing out of graphite microstructure remain difficult to calculate/predict. – WEW]
16	Wash-off in SG	Fission product mobilization	M	L	Not certain what fraction of the total source term is made up of material that was [deposited or – JMK] plated out in the SG. Assume this is not a large fraction ["Source term" has gotten to be a nebulous term. It is viewed variously as total radionuclide inventory, as total release from the fuel, as free radionuclides in primary circuit ready for discharge, or as total release from Rx building to environment. – LJL]	Limited data available to address wash-off of radionuclides under these conditions	Obtain data

Item No.	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for importance	Rationale for knowledge	Recommended Action
17	Chemical reaction with [prior deposition and – JMK] plate-out [in the reactor coolant system – JMK] releases radionuclides to gas stream	Fission product mobilization	H	L	High for Cs. Maybe Low for I because I is treated as a noble gas [in current analyses – LJL], anyway. This rating is very uncertain. More knowledge is needed to assess whether or not the plate-out inventories are significant.	LJL-To amplify, to (our?) knowledge, detailed evaluation has not been performed previously. Overall thermo-chemical environment will be complex. This may be within scope of modern thermochemical analysis tools, but this has not been evaluated.	Evaluate relative amounts of [deposited and – JMK] plated out material vs. total source term if quantities are significant. Evaluate characteristics of re-mobilization of Cs (and which form) and re-plateout on carbon in a new form. [Evaluate applicability of modern chemical simulation tools. If applicable, determine required validation matrix. – LJL]
18	Rx building decontamination factors	Fission product mobilization	H	L	High, assuming that this decontamination factor is necessary to meets limits [The importance really depends on how much the analysis relies on building decontamination factor (DF). Since some HTRs do not take any credit for DF, it could actually be L or NA. Others take limited credit for it. If it is essential to meet limits, then it could be H. But on average, M might be best if a single value must be indicated. – LJL]	complex processes (stratification, separation). Very design dependent (e.g., may have HEPAs) [Again, the state of knowledge really depends on how much credit is taken for DF. If only modest reduction factor is taken, then a relatively primitive model might be adequate. But if large DF credit is taken as major means to meet limits, then sophisticated model required and current knowledge would be L. – LJL]	[Given uncertainty in importance, following steps proposed: - LJL] 1. Assess the importance of the DF [Then, depending on importance and credit taken in analysis, remaining steps are suggested as appropriate:-LJL] 2. Perform more detailed analysis on building DFs including stratification and separation 3. Obtain data to quantify separation and stratificationSee NUREG/CR-6944 Vol. 3
19	[Combustible gas generation –SJB]	[Explosive gas mixtures - SJB]	[M - SJB]	[M - SJB]	[Previous safety assessments indicated that the injection of enough water to generate significant quantities of combustible gases would be a BDBA - SJB]	[Conservative assumptions regarding the amount of gas generated can be made for each event considered. – SJB]	Evaluate the gas generation as part of the recommended system code analyses for each event considered. – SJB]
	Aside: It would be a good idea to have a code	Look at integrating with specialty codes	[See write-up in Section 6.1- SJB]				

7.2 Normal Operation

Normal Operation refers to steady-state, routine load changes, startup and shutdown, and other conditions and transients not involving failures that challenge safety-grade systems or components. Some event sequences nominally classified as AOOs could fall into this category. Also, included within this category is the scenario where a small leak from the secondary to the primary may develop that delivers moisture to the primary at levels that maintain a water concentration just below the plant technical specifications limit. Event classification was not meant to be one of the panel's tasks; the objective here was to try not to exclude any significant phenomena, processes, or events related to long-term modest moisture levels in the primary system. Rankings are shown in Table 9.

Chronic oxidation and performance degradation of key core components is the primary concern. Similar to the assumptions stated in the acute phenomena section, if there is uniform oxidation throughout the core, the performance will be minimally affected. These phenomena assume that key components such as the core support columns, insulation attachment points, and contamination sources are preferentially oxidized over a long period of time.

It is assumed that the primary coolant will normally have trace amounts of oxidizing species (ppm of H_2O) that will allow a slow, steady oxidation of carbon-based material for very long life components. This slow oxidation mechanism—because of low levels of contaminants—will occur at normal operating temperatures and for long periods of time. As such, oxidation will primarily occur in the diffusion and boundary layer of controlled regimes, which are much more complicated than simple chemical kinetically controlled oxidation. Additional R&D will be required to ascertain the rate of oxidation within these regimes.

Chronic oxidation was anticipated to impact the performance of carbon-based materials in two ways: the slow degradation can lead directly to compromised components in the core, or it could exacerbate the issues in the acute moisture phenomena identified previously since the components will not be in as-fabricated condition.

Table 9. Long-term exposure in the primary system to low-level moisture (ppm oxygen).

Item No	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for importance	Rationale for knowledge	Recommended Action
1	Changes to mechanical properties of key structural components (e.g. permanent reflector blocks, core support structure, insulating composite structures)	Structural integrity	H	L	Compromises structural integrity and configuration of the carbon based materials [No doubt that oxidation is important in assessing structural margins. Having said that, its immediate importance is only relevant in the context of the existing structural margin relative to the allocated margins. If excess margins are minimal, then detailed phenomenological understanding is critical. – LJL] [Assumed Importance is based upon performance degradation of key components. Assumption is that support columns are compromised – WEW]	Little understanding of long-term effects [On the one hand, it is hard to argue with the desire for additional data, particularly for current grades of graphite. BUT, on the other hand, it seems that there is significant experience with graphite reactors operating in high moisture or oxidizing environments without adverse structural consequences. More data would certainly allow better optimization of margins, but I would not say that it is essential to path forward. – LJL] [Agree that there is ancillary experience (qualitative data) to indicate that these reactors can run for extended periods of time but there is no direct data to support calculations or predict oxidation rates. Since oxidation will occur at high temperatures with a low oxygen atmosphere the rate is definitely within the diffusion controlled region which is the least understood oxidation region. – WEW]	Get more data and improve analytical model [Importance H and knowledge L would suggest that current situation is unacceptable. I do not think this is the case. It is certainly not consistent with FSV and AGR experience. The situation is more balanced than this H/L suggests. We do desire more knowledge, but design margins and ISI can help manage less than 100% knowledge. – LJL] [The only debate is one of importance since the state of knowledge is definitely Low. If the importance is rated M or L then we need to address the graphite structural integrity importance listed in the acute phenomena since that is less likely to occur because of small time scale. – WEW]

Item No	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for importance	Rationale for knowledge	Recommended Action
2	Changes to thermal properties of key structural components (e.g. permanent reflector blocks, insulating composite structures) [Should distinguish between different thermal properties (conductivity, heat capacity, emissivity), since the impact of oxidation would probably be quite different for each one. – LJL]	Thermal performance and structural integrity	H	L	Change in thermal properties affects temperature and structural properties [given that we already have to design for fairly wide range of thermal conductivity because of irradiation, I am not sure the variation because of low level oxidation is really an H. – LJL] [I agree but the argument is "We can always design larger RPVs to hold more pressure, or more detectors to determine moisture faster - yet the acute phenomena are still high." Where do we draw the line? – WEW]	Little understanding of long-term effects [Agree that there is ancillary experience (qualitative data) to indicate that these reactors can run for extended periods of time with potential thermal degradation but there is no direct data to support calculations or predict oxidation rates. If we state that we can design around this problem then I assert that we can do that for all phenomena acute and chronic. – WEW]	Get more data and improve analytical model to calculate long term oxidation rate of carbon-based components - WEW
3	Effect of this long term degradation on system response during accident conditions	Structural integrity	L	L	Will determine design margins INCLUDING property degradation	Little understanding of long-term effects	Should be covered by design based on info from 1 and 2 above

Item No	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for importance	Rationale for knowledge	Recommended Action
4	Long term oxidation changes bypass flow areas	Fuel or structure temperature	L [H-YH, RRS]	L	Committee agreed that long term oxidation is less of a factor than radiation induced dimensional changes [No data were available to the panel at the time of the meeting to confirm that the opinion, held by some panel members, that radiation induced dimensional changes would overwhelm the dimensional changes associated with the long-term presence of moisture. Therefore, we rate the importance high - YS, RRS]	Little understanding of long-term effects. [Changes to the bypass flow distribution will cause changes to the core temperature distribution. – YH, RRS]	No additional action required [Obtain data on the dimensional changes in graphite because of corrosion in the presence of low concentrations of water vapor at representative temperatures. Also obtain data on the dimensional changes in the graphite because of long-term exposure to radiation. Compare the magnitudes of these changes to assess their relative importance - YS, RRS]
5	Long term corrosion or other impacts to performance	Reactivity control	H M [LJL]	L	For example, reserve shutdown spheres gummed up [Scope of impact was limited and within capability of system to respond. Impact of a stuck RSS hopper or slow rod bank on overall reactivity control and reactor shutdown capability was minor. – LJL]	Historically, unexpected conditions exceeded design expectations [Primary issue was not inadequate phenomenological knowledge. It was simple failure to consider the potential situation. Once situation is actually considered, then need for any additional phenomenological information can be assessed. – LJL]	Apply lessons learned [specifically, apply lessons learned to a) consider full range of local chemical environments, b) design to preclude adverse environment or effect, and c) specify reasonable ISI to ensure no adverse effect. – LJL]

Item No	Phenomenon	Figure of Merit	Importance (H/M/L)	Knowledge (H/M/L)	Rationale for importance	Rationale for knowledge	Recommended Action
6	[Moisture distribution in the core and transport to the core – item added by YH, RRS]	Structural integrity	H [L – LJL]	L	The mechanisms that transport the moisture to the core will in part decide the moisture distribution within the core, which may be preferential and thus may result in local damage and hot spots. [The condition we are talking about is a very low equilibrium concentration of water (and other oxidants) circulating in the coolant during normal operation. Tech specs will limit this to a few ppm (e.g., < 10 ppm). This is typically residual moisture left in the system from startup. At these low concentrations, the moisture concentration through the system is essentially uniform, because of the low rate of oxidation under this condition. (If the rate of oxidation was significant enough to noticeably deplete the concentration passing through the core, then the moisture would have relatively quickly been depleted from the system.) Hence the concern for this regime is very low rate of oxidation sustained over very long periods (e.g., years) after repeated startup and shutdown cycles at the tech spec limit of oxidant concentration. Even if a very small undetected leak were present, the limiting condition would still be the tech spec concentration limit. The components are designed for this environment, and if the concentration exceeds the limit, then the operating temperature is reduced or the reactor is shut down. – LJL]	Little information is available concerning these phenomena.	[See recommended action for item 4 of this table – YH, RRS]

Additional comments on this scenario were made by some panel members and are as follows:

Yassin Hassan, Richard Schultz: Some considerations for pebble beds include: (a) because of the complex flow structure between the pebbles, it will have a significant effect on the local distribution of the moisture which is important for the oxidation phenomena and (b) because of the presence of the dust, the phase change phenomena will likely be affected and might create an embryo (site) for condensation.

8. OTHER CONSIDERATIONS FOR PBRs

The panel discussed the differences between prismatic and pebble bed cores with respect to moisture ingress, and generated Table 10 to capture the major points and conclusions. As with the prismatic cores, PBRs are under-moderated, where the effect of steam/water ingress on reactivity is greater the more the under-moderation. This can be controlled by design.

Table 10. Differences between prismatic and pebble bed cores pertinent to moisture ingress.

Item No	Area	Comments
1	Rx physics	No major differences The PBR has a lower operational excess reactivity than the prismatic design, especially at BOL. This will result in a smaller reactivity insertion during water ingress, coupled with lower fuel temperature rises. The magnitude of the reactivity feedback coefficients in the PBR UO_2 fuelled core will also differ from the UCO fuelled prismatic core, but this is expected to play a second-order effect during the short-term dynamic phase of the transient.
		AREVA's PBR is based on the HTR-Module design, which has a cylindrical core as opposed to General Atomics' prismatic MHTGR, which has an annular core. This is not expected to make a significant difference on the water ingress results, since the control rods in both designs are located outside the core in the side reflectors. SJB- Control and shutdown rods in the outer reflector would be more effective controlling the neutronics in the prismatic (annular) core than they would in the pebble bed (cylindrical) core. Since they are closer to the center of mass of the neutronics in the annular core, their effectiveness would be less affected by the moderation from the water ingress.
2	Thermal hydraulic calculations	PBR has a more complex geometry, making calculations more complex. Whereas the prismatic designs core flow paths can be well approximated by 1-D channel flows, the PBR core flow distribution requires proper 2-D modeling, and for accurate core-reflector interface flows even 3-D models. SJB – PBR coolant flow modeling is especially difficult at very low (accident condition) flow rates.
3	Water effects on friction (lubricates)	Possible a small compaction.
		Need to distinguish between effect of water vapor and liquid water on friction. Liquid water is only credible during shutdown, in which case effect of compaction would not be very relevant.
4	Fuel element oxidation	The coatings of the pebbles are matrix material, not nuclear graphite. While the fabrication process and raw materials are similar to nuclear graphite, there are specific differences because of the different material parameters for pebbles. These differences (different binder resins and a final firing temperature of only 1850°C instead of 3000°C) will create a different microstructure, and thus oxidation rate, than nuclear grade graphite. However, since pebbles are relatively short-lived components and can be replaced after each cycle through the core, the importance level is only M or L. - WEW More quantitative data and improve analytical model to calculate long term oxidation rate of carbon-based components
5	Moving core	Does the motion increase the oxidation potential? Possible for long term, probably not for short term accident.

Item No	Area	Comments
6	Dust?	If dust is strongly adhered to surfaces, not sure it will make any difference [Dust in PBRs may affect fission product transport and chemical environment in releases – SJB]
7	Fuel Handling Machine	Probably no impact on BUMS
		Maybe impact on mechanical performance? [This is a design issue. Not relevant for safety. – LJL]
8	Safety shutdowns are different	Accident sequences would be different.

9. SUMMARY AND CONCLUSIONS

An assessment of modular HTGR moisture ingress events, making use of a phenomena identification and ranking process, was conducted by a panel of experts in the related areas for the NGNP design. The NGNP Project had decided to develop an assessment of the impacts of postulated water/steam ingress events on the HTGR to better understand the needs for additional R&D, analytical tools, and experiments to validate the codes. Consideration was given mainly to the prismatic core gas-cooled reactor configurations incorporating an SG within the primary circuit. The focus of the study was on the 1980s version of the MHTGR. Some aspects of ingress events and consequences peculiar to pebble-bed cores were also noted.

In safety studies, the dynamic effects of water ingress into operating and shutdown core are considered of high importance. The 1986 MHTGR PSID included several DBA and BDBA versions of steam leaks and tube breaks, with and without safety system intervention, including SCRAM, turbine trip, and isolation and dump valve closures. The DBA scenarios resulted in modest power and maximum fuel temperature increases well within material limits, and with no fuel failure expected to occur. The consequences of a BDBA event (only safety systems responding) were more significant, with more water entering the primary system leading to more graphite oxidation. The primary pressure relief valve cycles resulted in releases of radionuclides. However, the PSID analysis concluded that the oxidation and the total dose to the environs would be well within limits. There are variations in the BDBA sequence (from the PSID) that could lead to more oxidation and dose; consequently, sensitivity studies, with variations in both sequence assumptions and models used, are recommended. This will require a systems accident code capable of simulating phenomena associated with moisture ingress, used to acquire a better understanding of the potential consequences of moisture ingress events and to optimize the design of mitigation systems in the process.

Fission product releases would result mainly from removal of fission product deposits from primary system surfaces, and from chemical reactions with the fission products and graphite. Moisture ingress would have no significant effect on in-tact fuel particles, only on defective particles in which the kernels are exposed. Releases to the environment would occur only upon relief valve opening(s). The panel identified the need for more data and improved modeling for decontamination factors for the reactor building.

The prismatic and pebble bed HTR designs are both under-moderated, which implies that any water ingress (additional neutron moderation) will increase the system’s reactivity. An additional important factor is the decrease of control-rod worth for HTR thermal systems, where the control rods are usually at the thermal neutron peak locations in the side reflectors. With a very hot core, however, any moisture ingress would be in the form of low-density steam, which would have only small effects on reactivity.

For the case of long term, low levels of moisture present during normal operating conditions are a significant concern, as oxidation rates and the physics for diffusion-controlled oxidation are not well understood. Additional R&D would be required to accurately calculate the oxidation rates, the effects on material performance, and the mechanisms controlling the oxidation behavior at high temperatures and low moisture environments. Long term structural damage would be a consideration as it may affect initial conditions in the evaluation of significant moisture ingress accidents.

Considering resource limits and the lack of more detailed NGNP design information available for this assessment, many of the possible sequence options and design variations were not covered here. As the design progresses, the assumptions should be revisited in any subsequent PIRT-like activities.

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Appendix A

NGNP Moisture Ingress Assessment Committee Membership

Appendix A—NGNP Moisture Ingress Assessment Committee Membership

Name	Function	Expertise	Organization
S. J. Ball	Committee Chair	Accident sequences	ORNL
G. Strydom	Member	Reactor Physics	INL
J. M. Kendall	Member	Fission product transport	Global Virtual LLC
L. J. Lommers	Member	Reactor Design and Safety Analysis	AREVA
Y. Hassan	Member	Modeling and Experiments	Texas A&M
R. R. Schultz	Member	Modeling and Experiments	INL
W. E. Windes	Member	Graphite Properties	INL
S. Basu	Observer		NRC/RES
D. Carlson	Observer		NRC/NRO
S. Rubin	Observer		NRC/RES
M. Holbrook	Observer		INL
P. Jordan	Observer		INL
W. Landman	Facilitator		INL
P. Mills	Observer		INL

Appendix B

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Appendix B—Bibliography

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Appendix C

Presentation Viewgraphs

Appendix C—Presentation Viewgraphs

Presentation 1 Modular HTGR Ingress Accidents

Presentation 2 MHTGR Design for Water Ingress Events

Presentation 3 Moisture Ingress Effects on Fission Product Transport

Presentation 4 Overview of the MHTGR Core Physics during Moisture Ingress Events

Modular HTGR Ingress Accidents

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NGNP Moisture Ingress Assessment

Salt Lake City February 16-17, 2011

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1

Water in gas-cooled reactors???

- New graphite @ startup: “tons of water”
- AVR (Germany): “tons during a shutdown”
- Fort St. Vrain: persistent inleakages through circulator bearing seals
- Windscale accident dousing
- English & French ingresses
- MHTGR: steam generator tube leak *worry*

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2

Modular HTGR moisture ingress – major phenomena/events might include:

Reactivity effects (increase w/under-moderated core)
Reduction of control/shutdown rod worth
Moisture detection
SCRAM action (response; interference/corrosion?)
Pressure increase in primary system
RPV pressure relief valve action
Steam generator isolation & dump action
Graphite oxidation/corrosion
Fission product releases from fuel and graphite
Explosive gas mixtures – RPV & confinement

Others?? Thus the committee discussions...

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3

Characteristics of event sequences impacting the consequences

- Core: pressurized or depressurized (& between)
- SCRAM
- Operating or startup or shutdown conditions
- Rate and quantity of water/steam ingress to the core before water source termination
- Core temperatures & flow distributions vs. time
- RPV pressure relief valve operation
- Confinement/containment release characteristics
- Fuel & graphite fission product "inventories" →dose
- Plant configuration and protection system response
- Other…..? NOTE: sensitivity studies are needed!

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4

Figures of merit (FOM) – evaluation criteria

- Examples (from T/H accident PIRT):

- **top level**: dose at the site boundary or radioactive release from the confinement structure
- **second level:** worker dose
- **third level:** fuel failure fraction during accidents
- **lower level criteria**:
  - Fraction of fuel above a critical fuel temperature for a critical time period (per fuel performance model) [precursor to level 3 - fuel failure]
  - RPV and vessel supports, core barrel, and other crucial in-vessel metal components service conditions (time-at-temperature, pressure, etc.)
  - Reactor cavity concrete time-at-temperature
  - Primary coolant circulating activity (including dust)

PIRT Interpretations

- "Sufficient" knowledge
  - very well known, or
  - known well enough for the application

- Importance determination
  - PRAs might come in handy here
  - Accident classification: AOO, DBA, BDBA

PBMR project changed its PIRT process

– Revised ranking bins
  - Difficult to decide what to do with "medium bins"
  - Too many combinations of uncertainties were available to adequately recommend action resolution

– Assessment can focus on reviewer's opinion of what action is required:
  - Analysis tools upgrade, &/or
  - More validation (V&V data) needed
  - ….. or not

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PBMR PIRT Status Decision Chart

Status	Rank (High / Low)	Confidence in Rank (Sure / Unsure)	Confidence in Value (Sure / Unsure)	Symptom	Action required
8	High	Unsure	Unsure	Phenomenon is perceived as significant but is not well known.	High priority requirement for analysis and validation.
7	High	Sure	Unsure	Phenomenon is significant and confidence in value is low.	High priority requirement for validation.
6	High	Unsure	Sure	Phenomenon is significant and the confidence in rank is low.	High priority requirement for analysis.
5	High	Sure	Sure	Phenomenon is significant and well known.	Should be well represented in the model. Should be readily validated.
4	Low	Unsure	Unsure	Phenomenon is not significant but not well known.	Requires analysis and validation to determine rank and value.
3	Low	Sure	Unsure	Phenomenon is not significant and the confidence in value is low.	Low priority requirement for validation.
2	Low	Unsure	Sure	Phenomenon is not significant and the confidence in rank is low.	Low priority requirement for analysis.
1	Low	Sure	Sure	Phenomenon is well known and is not significant.	May be modelled without validation.

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Assessment process - suggestions

1. Pick an interesting accident scenario
2. Follow potential paths to resolution
3. For each path, evaluate impacted phenomena & FOMs, with varying assumptions of severity [sensitivity analysis, bounding sequences?]
4. Rankings – needs for more analysis and R&D
5. Tally up, & next scenario……??

PBR “differences” re: water ingress

- More/less under-moderated?
- Excess reactivity
- Cylindrical vs. annular core effects
- Moisture access to TRISO particles
- FP inventory in core
- Others ???

MHTGR Design for Water Ingress Events

Lew Lommers

Salt Lake City
February 16-17, 2011

Presentation Topics

- MHTGR designer's perspective on water ingress
- Water ingress mitigation in MHTGR
- Analysis of water ingress events
- MHTGR response to water ingress
- Conclusions

Topics NOT covered:

- Likelihood of water ingress (not covered)
- HTGR steam generator robustness (not covered)
- Overall risk spectrum of water ingress events (not covered)

Why Does MHTGR Designer Care About Water Ingress?

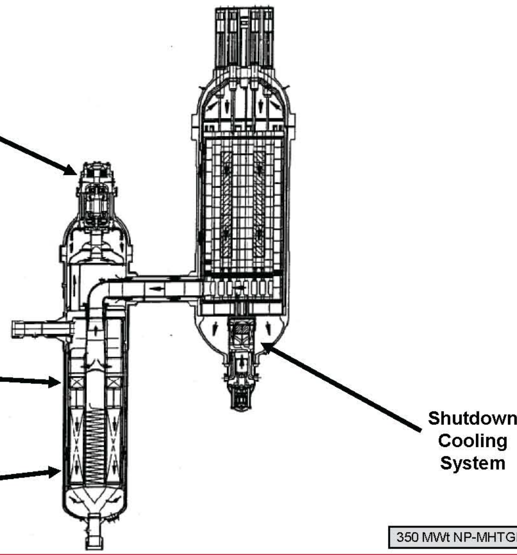
Safety

- Pressure increase
- Reactivity
  - Moderation and absorption
  - Rod worth
- Graphite oxidation
  - Structural margin
  - Combustible gas generation
- Fission product mobilization
- Fuel hydrolysis

Investment Risk

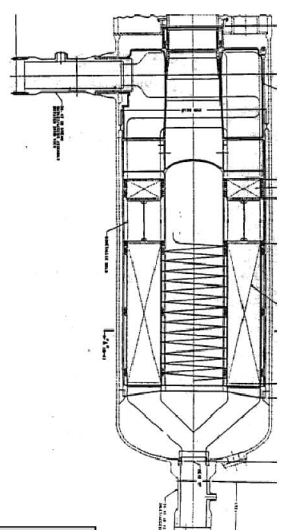
- Component degradation
  - Graphite oxidation
  - Corrosion
- Helium purification system loading
- Availability

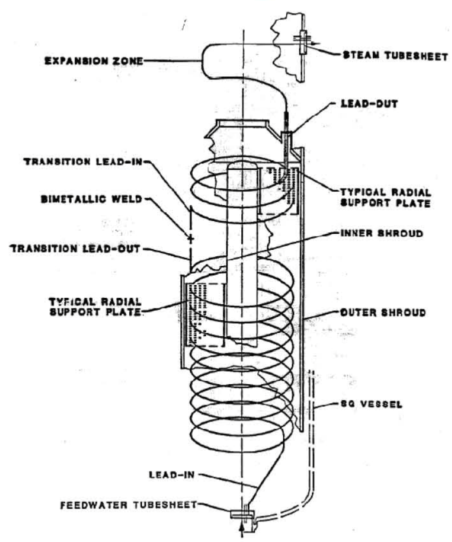
Main Sources of Water in MHTGR



350 MWt NP-MHTGR

MHTGR Once-Through Steam Generator Configuration





450 MWt MHTGR

Safety Features Available to Prevent, Detect and Respond to Water Ingress

- Detection Systems
  - Moisture Monitors
  - Pressure
  - Reactor Power
- Protective Actions
  - Reactor trip (rods or RSS)
  - Steam Generator Isolation
  - Primary Coolant Circulator Trip
  - Automatic Steam Generator Dump System
- Recovery Systems
  - Steam Generator and Reactor Vessel Drains
  - Helium Purification System
- Reactor and SG Design
  - TRISO Fuel – withstand very high temperatures
  - Graphite – very high heat capacity
  - Helium – inert
  - Negative Temperature Coefficient of Reactivity
  - SG – robust design

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Categories of Steam Generator Leaks

- Small steam generator leaks
  - Pinhole leaks
  - ~0.05 kg/s
- Moderate steam generator leak
  - Single tube rupture
  - 4.1 kg/s (offset rupture at superheater end)
  - 3.4 kg/s (offset rupture at economizer end)
  - 5.7 kg/s used in analyses
- Large steam generator leak
  - Multiple tube rupture
  - Very unlikely
    - Single tube is ~0.01/yr
    - Simultaneous rupture probability is infinitesimal
  - Propagation precluded
    - Robust tube design
    - Tube restraint
    - Analysis

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Detection and Mitigation Depends on Range of Water Ingress Scenarios

- Location (affects state of water and resulting transportability)
  - SG cold end
  - SG hot end
  - Circulator cooling water
  - SCS heat exchanger
- Magnitude (affects whether and how quickly detection thresholds reached)
  - Small leak
  - Large leak (less than 1 tube)
  - Tube rupture
  - Multiple tube rupture (BDBE)
  - Tube sheet failure (not credible)
- Plant state (affects pressure differential and state of water)
  - Hot operating
  - Cold pressurized
  - Cold depressurized

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Mitigation of Water Ingress

- Trip reactor and cool down
  - Terminate oxidation
  - Terminate reactivity excursion
- Trip main circulator
  - Reduce water/steam transport
- Isolate steam generator
  - Reduce water inventory entering primary circuit
- Dump steam generator inventory
  - Minimize water entering primary circuit
- Drain liquid water
  - Must provide low point drains

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Detection of Water Ingress Depends on Scenario (Ingress Rate)

		Reactor Trip	Loop Trip (Circ trip and SG Isolation)	Steam Generator Dump
Operator	Non-safety Sensitive 30 min. delay	?	?	?
RPS- High Power	For very rapid ingress Fast detection	X		
RPS- High Pressure	For very large ingress Slow detection	X (RSS)	X	
IPS- High Moisture	Non-safety Sensitive for small ingress	X	X	X
SCWS high pressure		Isolate SCHE and shut down reactor		

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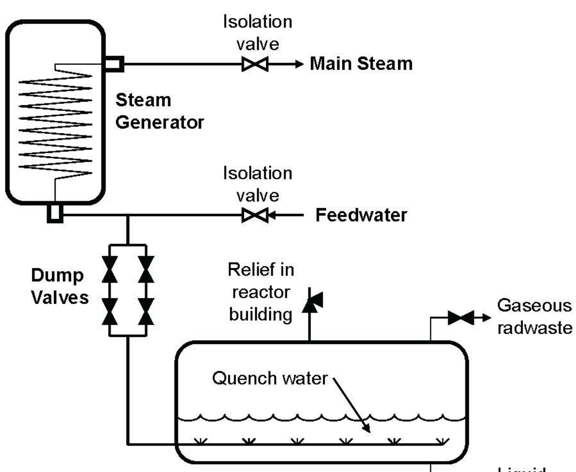
Steam Water Dump System

Dump Sequence

- Close FW isolation
- Close MS isolation
- Open dump valves
- Reclose dump valves based on secondary pressure



AREVA Schematic Functionally same as MHTGR

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Reactivity Effects Due to Water Ingress

- Modular HTGR core is typically undermoderated
- Core relies on reflector neutrons to stay critical
- Control rods located at interface between active core and side reflectors
- Thermal neutron flux from reflectors to core amplifies rod worth

- Water ingress increases in-core moderation
- Core becomes more reactive
- Less reliance on reflector neutrons
- Effective rod worth is reduced
- Results in double reactivity effect of water ingress

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Pressure Response Considerations

- Location of leak (quality of incoming water)
  - Hot end of SG – 100% steam
  - Evaporator – mixture of steam and water, significant flashing
  - Cold end of SG – liquid water, limited flashing
- Location of leak (quantity of incoming water)
  - Flow resistance from FW side of break (with orifice)
  - Flow resistance from MS side of break (no orifice)
  - Relative resistance of liquid, two phase, and vapor flow
- Phase change of water in system
  - Evaporation on hot surfaces
  - Condensation on cold surfaces
  - Strongly depends on operating state of reactor
- Transport of liquid to hotter locations
- Syngas production in core
- Relief valve staging and setpoints

Most of these factors ignored in routine analyses

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Typical Water Ingress Analysis

- Identify credible leak size (and location, etc.)
- Determine ingress rate
  - Initial
  - Sustained
- Evaluate water/steam transport
  - Typically assume steam
  - Leads to step increase in water concentration at core every ~10 seconds
- Evaluate initial transient
  - Pressure response
  - Reactivity transient
- Determine protection system response
- Incorporate impact of protection actions
  - Ingress rate
  - Water transport
- Detailed evaluations
  - Oxidation
  - Combustible gases
  - FP transport

Normally combined in integrated system code

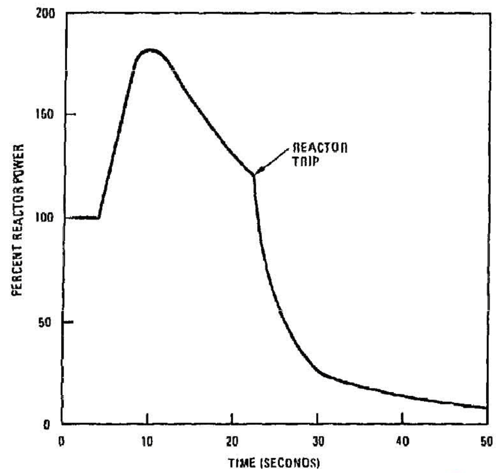
Ideally included in system code for pressure response

Conservatisms in Past MHTGR Water Ingress Analyses

- Larger ingress rate assumed (5.7 kg/s vs actual of 4.1 kg/s)
- Assume leak at top (to get steam ingress)
- Assume all water drains into primary circuit
  - Not consistent with leak at top
- No credit for HPS water removal
- No credit for high power trip (in case amount of water actually reaching core is too small to cause high power excursion)
- Oxidation does assume water reaches core

PSID DBE-6
Representative Water Ingress Event

- SG tube rupture (moderate ingress)
  - 5.7 kg/s
- No credit for high power trip
- Moisture monitor detects leak
  - Reactor trip with rods
  - SG isolation and dump
- Shutdown cooling system provides cooldown



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Summary of MHTGR Steam Generator Water Ingress Event Analyses

Event	Ingress Rate (kg/s)	Moisture Monitor	High Power Trip	High Press. Trip	Loop Trip Initiate	SG Dump	Cooling	Total Ingress	Total Oxidation Fraction	Relief valve open?	Dose at EAB
SG leak (nominal)	0.05	detected	NA	NA	at 390 s	yes	SCS	18 kg	minimal	none	none
Tube rupture (nominal)	5.7	detected	Assume not reached	NA	at 32 s	yes	SCS	272 kg	low	none	none
DBE-6 – tube rupture	5.7	detected	no credit taken	NA	at 30 s	yes	SCS	270	low bottom refl.: $2x10^{-4}$ avg $9x10^{-4}$ max	none	none
DBE-7 – tube rupture w/ SCS failure (DCC)	5.7	detected	no credit taken	yes	yes	yes	RCCS		$5.2x10^{-4}$	1 cycle assumed at 10 hr *	$4.66 x10^{-4}$ Rem (whole body)
DBE-8 – SG leak w/ moisture monitor failure	0.05	failed	NA	yes	4.8 hr	yes	SCS	841	Acceptable Core: $1.3x10^{-3}$ Bottom refl: $1.6x10^{-3}$ avg $6.1x10^{-3}$ max	none	none
DBE-9 – SG leak w/ dump failure	0.05	detected	NA	NA	380 s	yes – fails open	SCS	18 kg	low	none	none
SRDC-6 – tube rupture (only safety system mitigation)	5.7	failed	Yes RT @ 8s	yes RSC & Loop Trip	326 s	no	RCCS	4000? kg	acceptable	3 cycles (fails open on third)	0.045 Rem (whole body)

Source: MHTGR Plant Transient Anal Rpt (DOE-HTGR-86-121, Rev 1) and MHTGR PSID (HTGR-86-024 Amend 13)

* Calculated peak below setpoint

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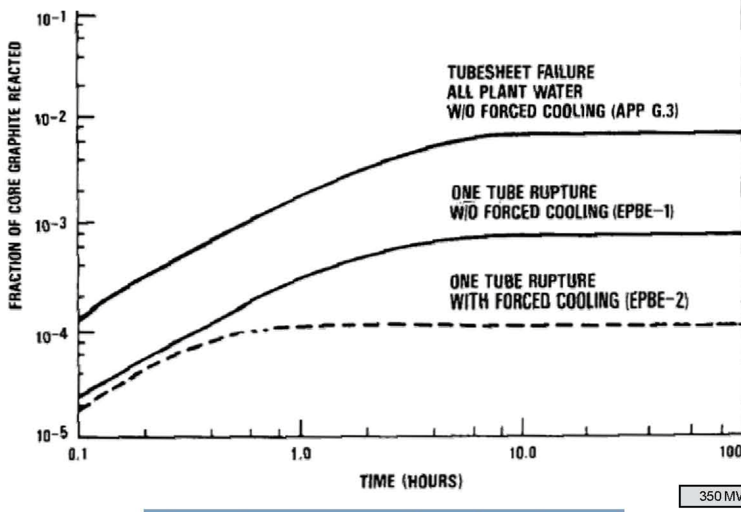
MHTGR Scoping Water Ingress Oxidation Results



350 MWt MHTGR

Large benefit for rapid cooling of core

Considerations for Other HTGR Designs

- **Protection system logic may be different**
- **Multi-loop concepts**
  - **Introduces the question of which loop to trip (already considered for large HTGRs)**
  - **Can revert back to MHTGR situation – trip all loops**
- **Pebble bed concepts**
  - **Oxidation of pebbles not identical to graphite blocks**
  - **Protection system logic significantly different for current concepts**
  - **Reactivity balance is more delicate**
    - Moderation ratio
    - Rod worth variation
    - Do not have large inherent reactivity swings which prismatic designs accommodate

Basic issues, fundamental physics, and general event evolution are similar for all HTGR concepts

Conclusion

- Water Ingress is an important HTGR event
  - Investment risk
  - Safety
- Multiple sources of water ingress
  - Steam generator is main consideration during normal operation
- Multiple detection
  - Power
  - Pressure
  - Moisture
- Multiple responses
  - Trip reactor
  - Loop trip (circulator trip and SG isolation)
  - SG dump
- Pressure transient is a key safety consideration
  - Does relief valve open?
- Safety requirements satisfied for all cases

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Moisture Ingress Effects on Fission Product Transport

J. M. Kendall
Global Virtual LLC

NGNP Moisture Ingress Assessment Committee Meeting

**Salt Lake City, UT
February 16, 2011**

1

•Outline

- **Fission Product Transport Considerations by type**
  - **Dose**
  - **Locations in reactor coolant system**
  - **Release and retention mechanisms**
- **Characteristics of release from the fuel from moisture ingress**
  - **Noble gases and Halogens**
  - **Metallics**
- **Other Considerations**

Representative Fission Product Types by Dose Considerations

	Fission Product Types			
Exposure	Noble Gas Kr, Xe	Halogen I	Metallic Cs, Sr	Noble Metal Ag
Plume – Whole Body	X	x	x	
Plume – Thyroid		X		
Ingestion		X	X	
Maintenance			x	X

Representative Fission Product Types by Location

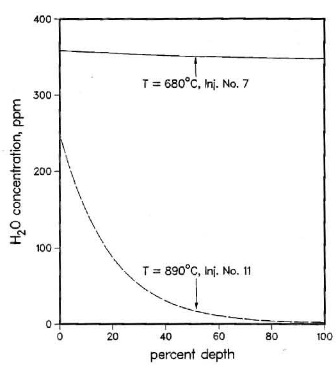
	Fission Product Types			
Location	Noble Gas Kr, Xe	Halogen I	Metallic Cs, Sr	Noble Metal Ag
Exposed Kernels	X	X	x	x
Graphite/ Matrix		x	X	
Coolant	X	X		
Coolant Pressure Boundary		x	X	X

Representative Fission Product Types by Moisture Ingress Release & Retention Mechanisms

Mechanism	Fission Product Types: Noble Gas Kr, Xe	Halogen I	Metallic Cs, Sr	Noble Metal Ag
Kernel Hydrolysis	X	X	x	x
Matrix / Graphite Reactions		x	X	
Liftoff/ Washoff		x	X	
Plateout/ Deposition		x	X	x

Moisture Ingress Graphite and Matrix Penetration

- Permeable to moisture in the coolant
- Matrix more reactive than Graphite
- Increased reaction rate at elevated temperature (> 800°C) reduces fuel particle exposure

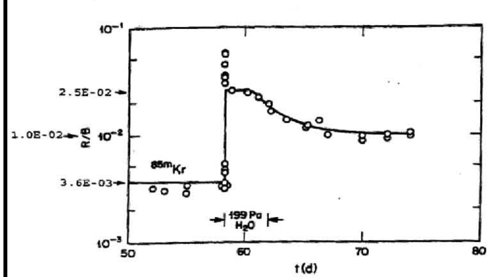


REACT calculations for water-vapor concentration as a function of depth from the graphite-gas interface in experiment HFR-B1

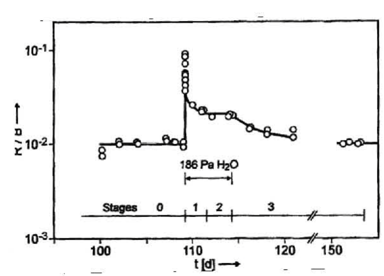
Moisture Ingress - Fuel Particle Hydrolysis

- No significant effect on intact particles
- Hydrolysis of exposed kernels
  - Release of stored fission gases
  - Enhanced release/birth ratio (R/B)
  - R/B decreases when moisture is depleted
- UCO more affected than UO_2
  - Theoretical Densities: UC – 13.63, UC_2 – 11.28, UO_2 – 10.96 Mg/m^3
  - ($UC_xO_{2-x'} \rightarrow UC_{x-y}O_{2-x'+y} \rightarrow UO_2 \rightarrow UO_{2+z}$)

HRB-17 Exposed Kernel Kr^{85m} Release Response



Kernel Containing UCO



Kernel w/ UCO Converted to UO_2

Additional Considerations

- Local fission product wash-off and lift-off from coolant system surfaces
- Release of fission products sorbed in graphite and matrix materials
- Changes in chemical form of fission products affect:
  - Release from fuel
  - Plateout and deposition on coolant system surfaces
  - Plateout and deposition in reactor building

Summary

- No significant effect on intact particles
- Effect of moisture ingress on fission product transport is dependent upon fission product type
- Local effects in core are strongly temperature dependent
- Needs for understanding of phenomena depend on safety analysis assumptions regarding:
  - Release from fuel
  - Deposition and release from coolant system surfaces
  - Plateout and deposition in reactor building

Overview of the MHTGR Core Physics during Moisture Ingress Events

MHTGR Moisture Ingress Assessment Technical Review Meeting

February 16-17, 2011, Salt Lake City, Utah

Gerhard Strydom

VHTR Methods

Idaho National Laboratory

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Presentation Overview

- HTR Water Ingress Core Physics Overview
- MHTGR Reactivity Control Design Philosophy
- MHTGR Reactivity Requirements and Control Rod Worths
- Dynamic Reactivity Effects of Water Ingress: MHTGR & HTR-PM
- Reference List

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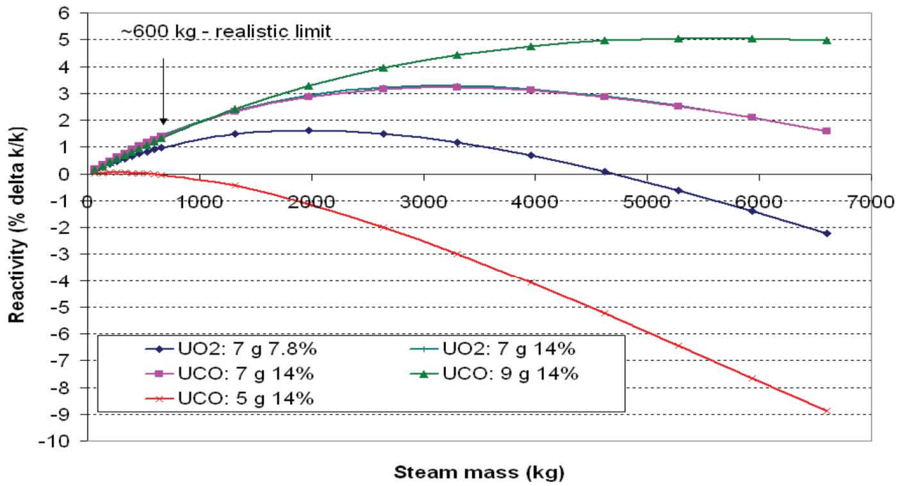
HTR Water Ingress Core Physics- Overview (1)

- The MHTGR core, like most HTR designs, is *under-moderated*. This implies that any additional neutron moderation will increase the system's reactivity.
- The positive reactivity change that occurs with water/moisture ingress is the combined effect of three phenomena:
  - ↓ – Less thermal neutrons available for U-235 fission due to neutron absorption by Hydrogen
  - ↑ – Neutron energy spectrum softens (less high energy neutrons), which increases the fission XS and decrease resonance capture in U-238
  - ↑ – Reduced neutron leakage out of the core region decreases ex-core control rods effectiveness
- This effect can be decreased by lowering the fuel:moderator ratio (e.g lower HM loadings and U-235 enrichment), but fuel designers usually require higher HM loadings to lower fuel costs.
  - PBMR design – 9 g HM/sphere, 9.6% enriched (~15000 UO_2 particles per sphere)
  - HTR Module – 7 g HM/sphere, 7.8% enriched (~13700 UO_2 particles per sphere)
  - MHTGR - 15.5 % enriched (~6000 UCO particles per compact)

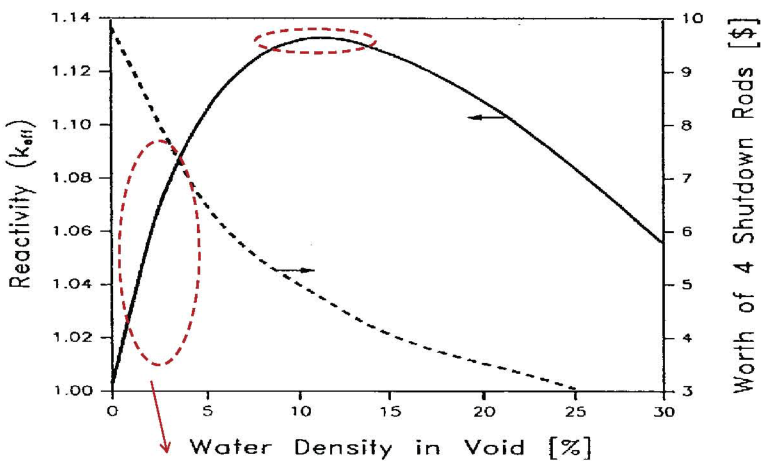
HTR Water Ingress Core Physics- Overview (2)

- The system reactivity behavior during water ingress events depends on:
  - System size, geometry and moderator material used
  - Fuel (and to lesser degree moderator) temperature (e.g. cold standby and hot operating conditions should be considered)
  - Fuel type (HEU, LEU, MOX, etc) and fissile/fertile mix
  - Burn-up status (e.g. MHTGR BOC core with fresh fuel and deep control rods vs. EOC core with more Pu and shallow control rods).
- These factors "combine" into the measurable temperature and density reactivity coefficients. Uncertainties in these coefficients are the main drivers for variations in the predicted power/temperature behavior during water ingress events.
- An important factor is the loss of control rod worth for HTR thermal systems. The decrease in shutdown margin needs to be taken into account, as well as ingress events at cold shutdown core states.

Example 1: Reactivity change (%) vs. primary circuit steam inventory for 5 HM and enrichment loadings in the HTR-MODUL [1]



Example 2: Keff and rod worth changes vs. water density for the HTR-PROTEUS [2]



For safety studies, both the gradient and peak value of the reactivity increase is important.

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MHTGR Reactivity Control Design Philosophy

- MHTGR core reactivity is controlled by a combination of fixed lumped burnable poisons, movable poisons, and a negative temperature reactivity coefficient.
- The *number* (30) and *location* of control rods and independent Reserve Shutdown Control (RSC) were designed to ensure that control reactor power for normal and off-normal conditions.
- The radial thickness of the active core annulus was designed to ensure the reflector control rod worths would meet all shutdown and operating control worth requirements.
- Evaluation of the control rod and RSC reactivity worths under hot and cold conditions showed a large margin [3] between the maximum reactivity requirements and the calculated rod worths.
- Typical MHTGR active-core power coefficients are strongly negative: -7 pcm/°C at BOC conditions and ~ -4 pcm/°C at EOC (for a typical moderator operating temperature of 700°C).
  - BUT: MHTGR safety studies assumed large 2σ uncertainties: ±20% on control rod worth, ±15% on power distributions, ±1.5% on keff and ±25% on water ingress!

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MGHTR Reactivity Requirements and Worths [3]

Control Requirement (% Δρ)

	BOC	EOC
Core operating Excess Reactivity	1.0	0.5
Temperature effect	4.8	1.2
Xenon + other FP decay	3.7	5.0
Shutdown + Uncertainty	2.0	2.0
Total	**11.5**	**8.7**

Control Worths (% Δρ)

	BOC	EOC
24 outer control rods	8.1	11.0
24 outer plus 6 inner control rods	10.1	11.3
Reserve shutdown control (RSC)	10.1	11.3
24 outer plus 6 inner control rods plus RSC	30.4	35.7
Margin available @ cold shutdown	**18.9**	**27.0**

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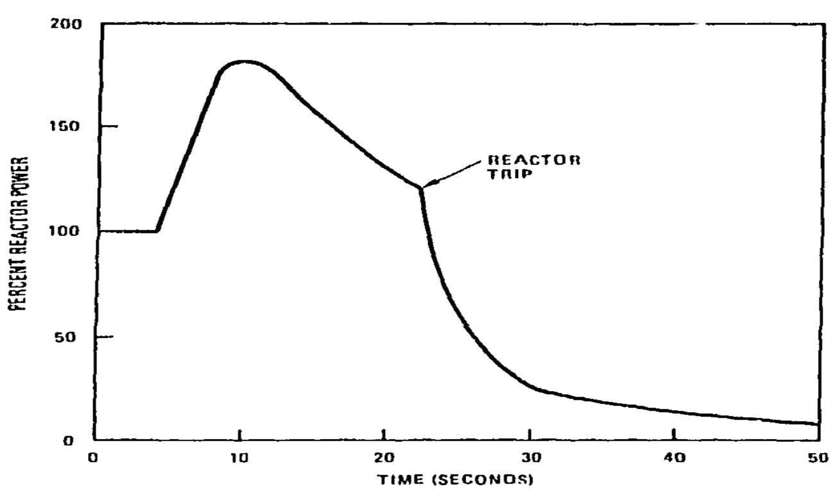
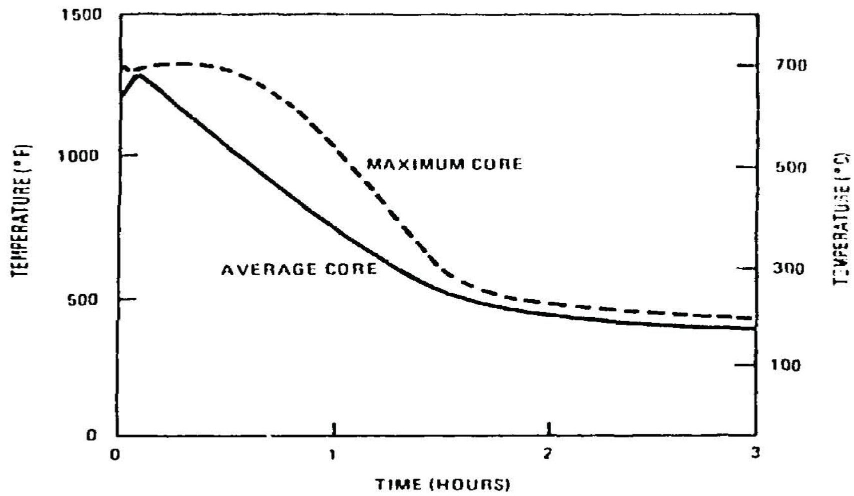
Dynamic reactivity effects of water ingress: MHTGR [4,5]

- The 1986 MHTGR PSID [4] included several DBA and BDBA versions of steam leaks and breaks, with and without safety equipment intervention (SCRAM, turbine trip, valve closures) .
- DBA scenario (Section 15.7) : Single SG tube rupture with reactor trip on 1200 ppm moisture detection. A bounding leak rate of 5.7 kg/s was assumed (starts with 27.2 kg/s), which lead to a peak reactivity increase of 0.196%, a power peak of 180% within 10 s, and maximum fuel temperatures rise of 48°C. A total of 270 kg steam entered the primary system, but only 28% of this mass reacted with the core.
- An ORNL 1992 study [5] used longer ingress times based on better estimates of the secondary system dynamics, resulting in lower power and temperature peaks. It was noted that:
  - most of the steam generator tube rupture scenarios required a few minutes to reach their peak steam ingress rates (e.g. 17 min for a single tube rupture),
  - uncertainties in the temperature and H_2 density reactivity feedback coefficients were responsible for most of the variation in the power behavior.

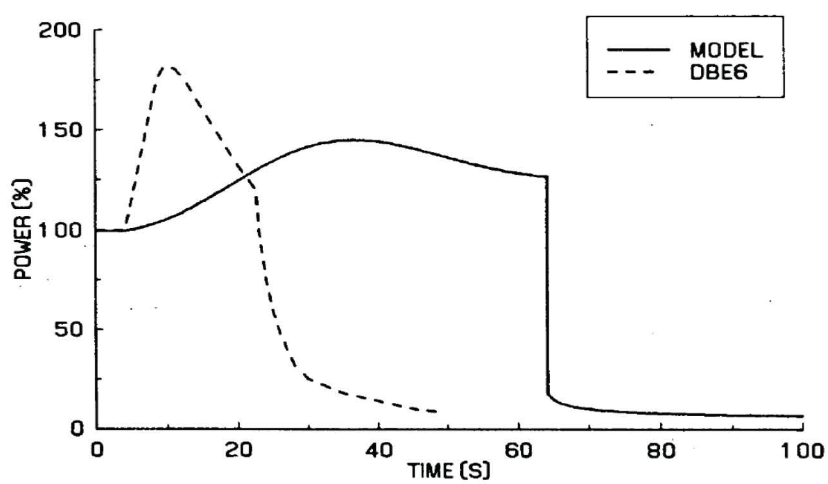
MHTGR Water Ingress: PSID DBA Reactor Power [4]



MHTGR Water Ingress: PSID DBA Fuel temperatures [4]



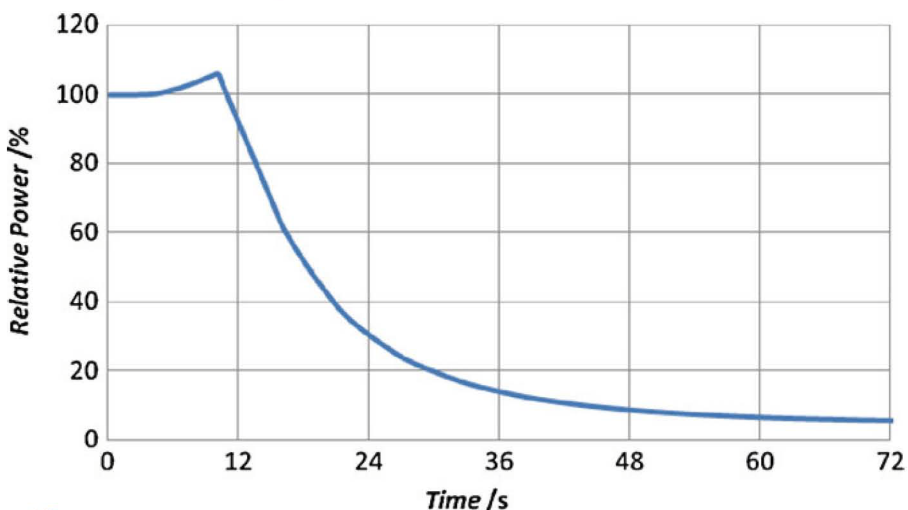
MHTGR Water Ingress: ORNL DBA Power comparison [5]



Dynamic reactivity effects of water ingress: 200 MW HTR-PM [6]

- Cylindrical pebble bed Chinese 200 MW HTR-PM design.
- Only *recent* example of coupled dynamic water ingress analysis (neutronics, T-H and chemistry), but performed with legacy tools (VSOP99 + TINTE).
- DBA: Ingress rate 5 kg/s over 120 s, capped at 600 kg total.
  - Power excursion limited to 278 MW, starting from a conservative steady state. Peak fuel temperatures rise from ~900°C to 1026°C in first 4 hours. SCRAM signal tripped by moisture detection and reactor period.
- DBA without SCRAM: Same scenario without SCRAM. Power stayed below 120%, and peak fuel temperatures < 1100°C.
- BDBA: Worst case possible used 200 kg/s for 5 s, then 100 kg/s for 15 s. Total ingress 2500 kg, with no SCRAM.
  - Power peaks at 1800% nominal within 4 s, but back to 200% in 20 s. Maximum fuel temperatures reaches 1250°C within 16 s.

HTR-PM Water Ingress: DBA Reactor Power [6]



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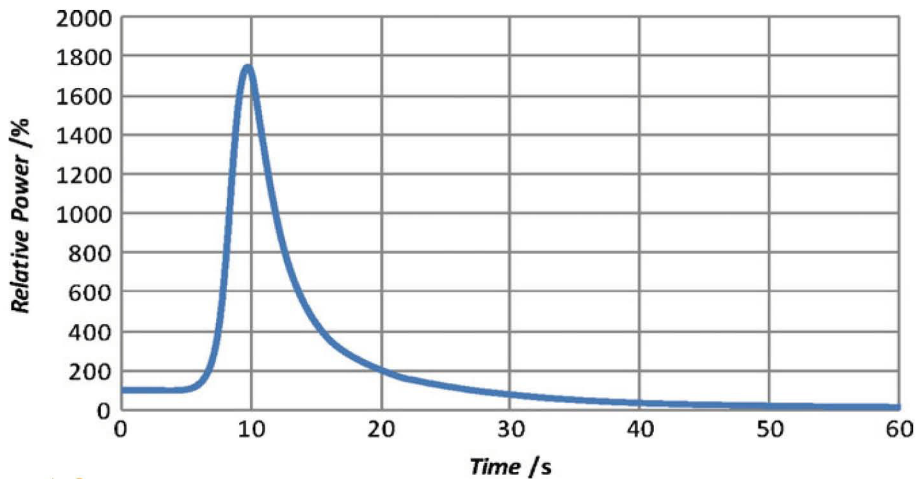
HTR-PM Water Ingress: BDBA Reactor Power without SCRAM [6]



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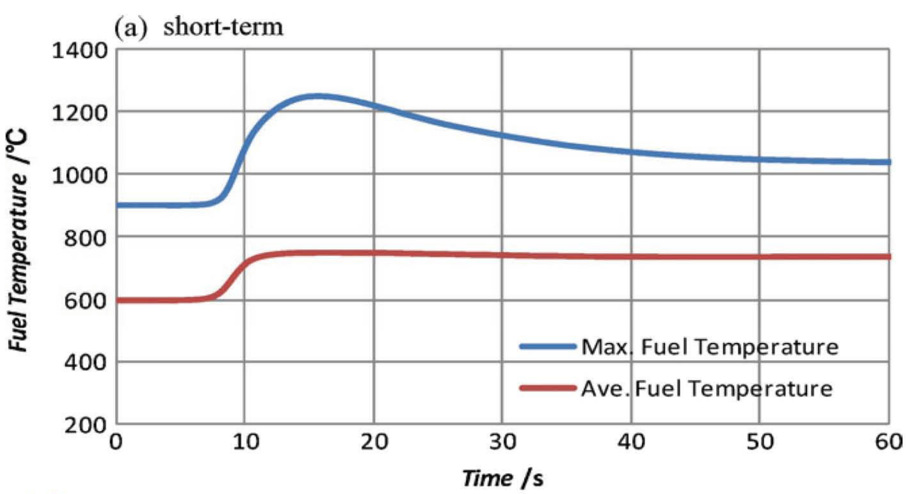
HTR-PM Water Ingress: BDBA Reactor Power without SCRAM [6]



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2. IAEA-TECDOC-784, "Response of Fuel, Fuel Elements and Gas Cooled Rea ctor Cores under Accidental Air or Water Ingress Conditions", 1993.
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6. Y. Zheng, et al, "Water-ingress Analysis for the 200MWe Pebble-Bed Modular High Temperature Gas-cooled Reactor", *Nuclear Engineering and Design, Vol. 240*, pp. 3095-3107, 2010.

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