

**Safety Reports Series**

**No. 54**

**Accident Analysis for  
Nuclear Power Plants with  
Modular High Temperature  
Gas Cooled Reactors**



**IAEA**

International Atomic Energy Agency

ACCIDENT ANALYSIS  
FOR NUCLEAR POWER PLANTS  
WITH MODULAR  
HIGH TEMPERATURE  
GAS COOLED REACTORS

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SAFETY REPORTS SERIES No. 54

ACCIDENT ANALYSIS  
FOR NUCLEAR POWER PLANTS  
WITH MODULAR  
HIGH TEMPERATURE  
GAS COOLED REACTORS

INTERNATIONAL ATOMIC ENERGY AGENCY  
VIENNA, 2008

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## FOREWORD

Deterministic safety analysis (frequently referred to as accident analysis) is an important tool for confirming the adequacy and efficiency of provisions within the defence in depth concept for the safety of nuclear power plants. Owing to the close interrelationship between accident analysis and safety, an analysis of a nuclear power plant that lacks consistency, completeness or quality is considered to be a safety issue. Developing IAEA guidance publications for accident analysis is thus an important step towards resolving this issue.

Requirements and guidance pertaining to the scope and content of accident analysis have been partially described in various IAEA publications. Several guidance publications relevant to water cooled water moderated power reactors (WWERs) and high power channel type reactors (a Russian design known as the RBMK) have been developed within the IAEA's Extrabudgetary Programme on the Safety of WWER and RBMK Nuclear Power Plants. To a certain extent, accident analysis is also covered in several publications in the IAEA Safety Standards Series, for example, in the Safety Requirements on Safety of Nuclear Power Plants: Design (NS-R-1) and in the Safety Guide on Safety Assessment and Verification for Nuclear Power Plants (NS-G-1.2). For consistency with these publications, the IAEA has developed a number of titles in its Safety Reports Series on accident analysis for nuclear power plants.

These safety reports aim to provide practical guidance on performing accident analysis. The guidance is based on current good practice worldwide. The reports cover all the steps required for accident analysis, that is, selection of initiating events and acceptance criteria, selection of computer codes and modelling assumptions, preparation of input data and presentation of the calculation results. The reports also discuss the various aspects that need to be considered to ensure that an accident analysis is of acceptable quality.

The first of these volumes is intended to be as generally applicable as possible to all reactor types. The specific features of individual reactor types are taken into account in the subsequent reports. The reactor types to be covered include PWRs, BWRs, PHWRs, RBMKs and modular high temperature gas cooled reactors (modular HTGRs). The present report is devoted to specific guidance for modular HTGRs.

This report is intended for use primarily by analysts coordinating, performing or reviewing accident analyses for nuclear power plants, on both the utility and the regulatory sides. The report will also be used as a background publication for relevant IAEA activities, such as training courses and workshops. The IAEA staff members responsible for this publication were Y. Makihara and N. Tricot of the Division of Nuclear Installation Safety.

### *EDITORIAL NOTE*

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# 1. INTRODUCTION

## 1.1. BACKGROUND

This report on modular high temperature gas cooled reactors (modular HTGRs) is intended to supplement the IAEA Safety Standards Series. The publications in this series related to the present report are the Safety Requirements on Safety of Nuclear Power Plants: Design [1] and the Safety Guide on Safety Assessment and Verification for Nuclear Power Plants [2]. Clarification of the accident analysis methodology at the next level is provided by the Safety Report on Accident Analysis for Nuclear Power Plants [3], which contains a comprehensive description of the general methodology for accident analysis. The objective is to establish a set of practical suggestions based on the best practices worldwide for performing accident analyses of nuclear power plants. These reports do not focus exclusively on a specific reactor type; however, although most recommendations are general and applicable to all types of nuclear reactor, specific recommendations and examples do apply mostly to water cooled reactors.

More specific guidance on performing accident analysis will depend on the characteristics of the nuclear power plant in question, and is usually developed for specific reactor designs or groups of reactor designs. Such design specific guidance is now being developed as separate safety reports. The reactor types covered include pressurized water reactors (PWRs) [4], boiling water reactors (BWRs), pressurized heavy water reactors (PHWRs) [5] and high power channel type reactors (a Russian design known as the RBMK) [6].

Furthermore, since several Member States are in various stages of design and deployment of modular HTGRs, and with several vendors developing HTGR systems and components, new guidance on accident analysis specific to modular HTGRs is useful.

This report complements the Safety Guides in the IAEA Safety Standards Series and supports the Member States in establishing their own analysis capabilities and safety standards.

## 1.2. OBJECTIVE

The objective of this report is to provide specific guidance on conducting accident analyses of nuclear power plants with modular HTGRs, taking into account specific design, operational and safety features. The term ‘modular

HTGR' as used here encompasses a family of reactor and plant designs, including those with prismatic and pebble bed cores, direct and indirect cycles, and gas and steam turbine electric balance of plants (BOPs). The definition also includes multi-unit reactor designs with shared support systems and a common control room. Although this report primarily addresses power plants, certain research and test reactors relating to the development and validation of HTGR technologies are also considered. The analysis of hazards associated with plant designs involving coupling with other applications (e.g. hydrogen gas generation, process heat) is not covered here. A summary of modular HTGR safety related design characteristics is given in Section 2 and in Ref. [7].

Since modular HTGR accident analysis is being conducted in a relatively new licensing environment, methodologies are not yet fully established or universally accepted. Thus this report is an attempt to assist with the transition. It deals with some current new approaches such as risk informed [8] and best estimate plus uncertainty (BEPU) licensing methods [9].

The report is intended primarily for reactor designers and code users or developers performing accident analyses. Regulatory bodies may use this report as a reference to assist in evaluating applicant submittals. While it is intended as a self-standing publication, reference is made to the first volume [3] of the safety reports on accident analysis for nuclear power plants and to the IAEA report on safety requirements for innovative reactors [7] for background information and more general guidance.

The present report contains details of initiating events and overviews of the safety aspects of events that could lead to fuel failure and potential activity release. Licensing type safety analyses aimed at demonstration of sufficient safety margins are also addressed. The analysis methods for the various events differ mainly in (a) the selection of events; (b) the event consequence acceptance criteria; (c) the structures, systems and components (SSCs) that may be credited<sup>1</sup> and modelled in accident mitigation analysis; and (d) the level of conservatism required in the analysis methods. Examples of consequence acceptance criteria are provided. Conservative assumptions (i.e. those leading to more severe consequences) for typical initial and boundary conditions are indicated for certain events. For other (i.e. rare) events, a best estimate analysis method may be acceptable, depending on the licensing basis approach allowed by the regulatory body. Suggestions for analysing various events related to the inherent safety features of modular HTGRs are also given.

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<sup>1</sup> The design analysis needed to demonstrate that there are margins to the quantitative design engineering acceptance criteria for credited SSCs is not included in this report.

This report is a result of a series of special IAEA technical meetings and is based on relevant IAEA [10] and other national publications.

### 1.3. SCOPE

#### 1.3.1. Events considered

As with the first safety report on accident analysis for nuclear power plants [3], this report deals only with ‘internal’ events originating in the reactor system or in its associated connected process systems. It does not cover originating events affecting broad areas of the plant (often called internal and external hazards), such as fires, floods (internal and external), earthquakes and aircraft crashes. However, analysis of the consequences of such events from a neutronic/thermohydraulic point of view (e.g. effect of core compaction associated with an earthquake) is partially covered by the present report. The emphasis here is on the transient behaviour of the reactor and its systems in postulated accidents, including the role of the confinement. The neutronics and radiological aspects are also covered to some extent. Except for proposed design engineering acceptance criteria for selected SSCs of importance to safety, very limited consideration is given to structural (thermomechanical) design aspects.

#### 1.3.2. Analysis types

This report deals with both best estimate and conservative accident analyses. While at present nuclear power plants with light water reactors (LWRs) are typically licensed by conservative accident analyses, Ref. [3] recommends carrying out best estimate analysis plus uncertainty evaluation so as to quantitatively identify safety margins. Since there are many research activities carried out on the BEPU method [9], it is envisioned that this method will be applied for certain accidents in licensing modular HTGRs.

A risk informed approach [8] applied to the licensing of nuclear power plants with modular HTGRs means in part that for frequent events (e.g. anticipated operational occurrences (AOOs)), only very low dose consequences would be permitted; for infrequent events (e.g. design basis accidents (DBAs)), somewhat higher dose consequences would be permitted; and for rare events (e.g. beyond design basis accidents (BDBAs)), even larger dose consequences would be permitted. Additionally, a risk informed approach means that the level of conservatism in the analysis methods may vary depending on the event classification. The use of best estimate analysis is

encouraged, but always with either a sensitivity analysis or preferably an uncertainty analysis, so that sufficient safety margins can be ensured for all demonstrations of plant safety.

This report does not intend to cover the task of assigning specific events or accidents to the various levels of event classification, with their corresponding probabilities and consequences. References are made, however, to levels of severity that serve as examples in the discussion of initiating events and accident sequences.

#### 1.4. STRUCTURE

This report is structured similarly to the first safety report on accident analysis for nuclear power plants [3], and accounts for the design specific features of modular HTGRs (Section 2). Appropriate references are made to the general guidance given in Ref. [3]. Section 3 deals with guidance on grouping and categorization of events. Event frequencies and challenges to fundamental safety functions are the bases for event classification and grouping. The frequencies of event initiators or accident sequences determine their categories. Section 4 presents some discussion of acceptance criteria for accident analysis that may be applicable to modular HTGRs. Since the acceptance criteria for various categories of events differ depending on specific designs, national codes and standards, and national regulatory bodies, they are meant to serve here only as examples. Section 5 is a guideline for the analysis methodology. Advanced methodologies (e.g. the BEPU method) are discussed. Section 6 attempts to provide a description of some accident analyses applicable to modular HTGRs, and for each category provides descriptions of the initiating events, the consequences and safety aspects, and the modelling and data needs. Analysis code attributes required for modular HTGRs are discussed in Section 7.

## **2. GENERAL SAFETY FEATURES OF MODULAR HTGR DESIGNS**

The modular HTGR fundamental safety objectives, requirements and design guidelines are based on specific design characteristics and inherent safety features. These typically include the use of high quality, high

performance coated fuel particles (CFPs) capable of containing radioactive fission products for the full range of operating and postulated accident conditions with a very low CFP failure fraction; the use of an inert single phase coolant (helium); a core with the characteristics of low power density, large heat capacity, high thermal conductivity and large thermal margins; and negative fuel and moderator temperature coefficients of reactivity sufficient to shut down the reactor for most reactivity insertion events (for both startup and power operation), with the exception of large water ingress events. Design basis accident decay heat removal is typically achieved via a passive system utilizing natural convection driven processes (the reactor cavity cooling system, or RCCS).

Modular HTGR safety analyses also include accidents involving ingress of water (steam) or air. For indirect and direct cycle gas turbine designs, the probability and potential magnitude of water ingress are greatly reduced compared with steam cycle designs. Water ingress during operation could result in a large positive reactivity insertion as well as significant graphite corrosion and primary pressure surges. Air ingress accidents are analysed for a leak or break in the primary system resulting in significant inflow of air. Air ingress accidents may also lead to an increase of the CFP failure fraction as well as degradation of the graphite core support structures.

## 2.1. COATED FUEL PARTICLES

The ceramic CFP can withstand much higher temperatures than can metal clad fuels. Coated fuel particle designs have evolved over the past several decades, and the current TRISO (triple isotropic) coated particle design is typically a ~1 mm diameter fuel particle consisting of a fuel kernel surrounded by a porous buffer layer (to accommodate fuel kernel fission gas release), an inner pyrolytic carbon (IPyC) layer, a silicon carbide (SiC) layer and an outer pyrolytic carbon (OPyC) layer. Many variations of fuel kernel composition, buffer and coating thicknesses, and microstructure properties are seen in various designs. Efforts are ongoing in all major gas cooled reactor programmes to develop detailed fuel performance models (for normal operation and postulated accident conditions) that accommodate the details of the TRISO design and operating history. The models use either mechanistic or empirical methods to predict fission product release, from both intact and failed particles, combined with probabilistic analysis that considers variations in individual particle characteristics (such as coating thickness) and operating histories. Models are generally qualified for long term maximum operating temperatures (e.g. ~1250°C) and time dependent accident temperatures

(e.g.  $\sim 1600^{\circ}\text{C}$ ). Typical limits are somewhat arbitrary, and in fact depend on the type of fuel, irradiation history, manufacturing processes and many other factors. Currently, fuel particle failure and fuel fission product release are modelled empirically in accident analysis codes.

## 2.2. HELIUM PRIMARY COOLANT

Helium gas pressurized to several megapascals is employed as the primary system heat transfer medium. It is an inert noble gas with no heat transfer limits or phase change, and with very little effect of density on reactivity. Compressibility effects generally are not important for modelling accident behaviour. Unlike LWR loss of coolant accidents (LOCAs), helium leaked into the containment does not condense, thus resulting in higher sustained containment pressures (at least for 'sealed' containment designs). An elevated pressure could serve as a driving force for radionuclide release during the long time period of a core heat-up accident.

## 2.3. DECAY HEAT REMOVAL

Modular HTGRs typically rely on passive decay heat removal systems in cases where none of the active heat removal systems are available. Here, the core decay heat is transferred to and through a mainly uninsulated reactor pressure vessel (RPV) and ultimately to the RCCS in the reactor cavity, primarily by radiation heat transfer. In normal operation, the RCCS functions to cool the safety related cavity components such as the vessel, the vessel mechanical support structure and the surrounding concrete structure. If there is a need to switch the RCCS from normal to accident mode, typically no active component or operator actions are required, at least for very long periods (i.e. days).

## 2.4. CORE THERMODYNAMIC CHARACTERISTICS

Modular HTGRs utilize large reactor vessels with large graphite moderated cores having relatively low power densities. There is also a large temperature difference (margin) between maximum fuel operating temperatures and the fuel temperature at which fuel particle failures begin to occur. Cooling system failures result in very slow core heat-up (and cooldown)



transients. In loss of forced circulation (LOFC) accident scenarios, fuel temperatures typically reach peak values after a few days.

## 2.5. REACTOR SHUTDOWN

Negative temperature coefficients of reactivity for the fuel and moderator are typically attained over the entire fuel cycle and temperature range of interest. Combined with relatively low excess reactivities, these attributes significantly reduce the safety demands placed on the active reactivity control and shutdown systems. In general, modular HTGRs have two safe shutdown systems — the normal shutdown system and the reserve shutdown system — so as to meet the requirement that reactor shutdown systems provide diversity. Normally, cold shutdown can be attained without the use of the reserve shutdown system.

# 3. CLASSIFICATION OF EVENTS

## 3.1. GENERAL EVENT CLASSIFICATION GUIDELINES

A risk informed process for classification of events for modular HTGRs, or for any accident analysis, generally involves two steps. One step is to classify events by their frequency of occurrence; the other is to classify or group the events by type or event sequence. The classification by frequency requires insight into the event initiator probabilities and will affect the determination of the acceptance criteria and the type of analysis (i.e. best estimate or conservative) to be applied to the event sequence analysis. It involves sorting events into specific groups defined either by their characteristics (e.g. loss of pressure boundary, reactivity change, loss of normal heat removal) or by their challenges to safety functions.

Classification of plant events is implemented only after a thorough analysis and comprehensive review of the plant design for all postulated event initiators and event sequences. A plant probabilistic safety assessment (PSA) is to be used to develop a comprehensive analysis and evaluation of the possible event scenarios.

For the purposes of accident analysis, all events are grouped into frequency based categories. In this report, the classification of accidents

typically follows the guidance given in an IAEA report on the unique safety features of innovative reactors, primarily modular HTGRs [7].

The following is an elaboration of the three major event frequency based categories, with sample frequency ranges:

- *Anticipated operational occurrence (AOO)*: An operational process deviating from normal operation that is expected to occur once or several times during the operating lifetime of the power plant. Anticipated operational occurrences are typically associated with events having a mean frequency of occurrence of  $10^{-2}$  per plant year or higher.
- *Design basis accident (DBA)*: Accident conditions against which a nuclear power plant is designed according to established design criteria, and for which the release of radioactive material is kept within regulatory limits. Design basis accidents are typically associated with events having a mean frequency of between  $10^{-2}$  and  $10^{-4}$  per plant year.
- *Beyond design basis accident (BDBA)*: Accident conditions that can be more severe than those of DBAs; an accident falling outside the plant safety systems design basis envelope. Safety related and non-safety related SSCs that are credited and modelled in BDBA safety analyses must be shown by engineering design analysis to meet appropriate engineering design acceptance criteria. A BDBA may or may not involve core degradation. The release of radioactive materials must conform to national nuclear regulatory body regulations. Beyond design basis accidents are typically associated with events having a mean frequency of between  $10^{-4}$  and  $5 \times 10^{-7}$  per plant year. Typically, the lower limit is considered a ‘cut-off’ frequency below which it should be demonstrated by an analysis of the technical basis for the event frequency that an accident analysis is not required.

Event frequencies are specified on a per plant year basis, and as modular HTGR plants have a number (e.g. 4, 8, 10) of separate and largely independent reactor modules, the number of reactor modules must be considered in the calculation of event frequencies.

The risk informed approach to modular HTGR safety analyses may include an event frequency versus event dose consequence acceptance criteria framework. This proposed specific framework, which is to be approved by the national nuclear regulatory body, provides that: frequent events (i.e. AOOs) must have either very low or no dose consequences to on-site personnel or to off-site individuals or the surrounding population; infrequent events (i.e. DBAs) are permitted to have limited consequences; and rare events (BDBAs) are allowed much higher consequences.

The relationship between the event frequency and the event dose consequence is qualitatively illustrated in Fig. 1.

The following sections more fully describe each step of the two step process.

### 3.2. CLASSIFICATION OF EVENTS BY FREQUENCY

The classification of individual modular HTGR events into event frequency categories may be implemented in one of two ways, depending on the philosophy of the safety analyst and the national regulatory body. The first approach utilizes the event initiator frequency as the event frequency ('initiation frequency'), as in the traditional 'deterministic' approach used for LWR accident analysis. The second approach utilizes the event sequence frequency, also known as the scenario frequency, and takes into account the probability of failure or success of the protective and mitigation actions of the plant equipment and the operators. This is considered to be a 'probabilistic' approach in that the probability of occurrence of the event sequence is accounted for and the event tree analysis portion of the PSA is utilized. The uncertainties in this process are usually significant and need to be considered in any categorization.

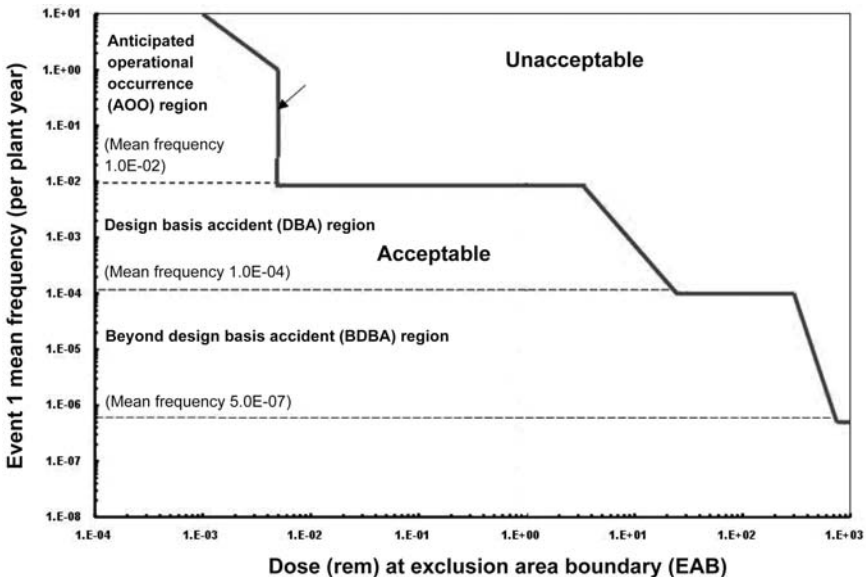


FIG. 1. Sample classification of events by frequency and consequence (dose).

Very rare events (those falling below the cut-off frequency for BDBAs) generally are not analysed. However, borderline events, which involve large frequency uncertainties, usually also suffer from large uncertainties in the data and models used to analyse the accident. If there is a potential for a large increase of the dose consequences (i.e. ‘cliff edge’ events), then the uncertainties are a factor in determining whether such events need to be included in the accident analysis. As stated in Section 3.1, it must be demonstrated by an analysis and evaluation of the technical basis for the event frequency that an accident analysis is not required.

The safety analyst and the regulatory body may utilize either a deterministic approach or a probabilistic approach for event classification. Additional information on event classification for modular HTGRs can be found in, for example, Refs [3, 11].

### 3.3. GROUPING OF EVENTS BY TYPE

For a certain design of modular HTGR, individual accident scenarios can be derived from combinations of plant operational states and initiating events. Computational analysis of all resultant scenarios is not practicable. Therefore, Ref. [3] suggests selecting a reasonable number of limiting cases from each event type in each event group. For accident analysis, these cases should be those that result in the highest consequences in the group relative to the relevant acceptance criteria.<sup>2</sup>

Typically, the grouping is based on a challenge to fundamental safety functions or on dominant phenomena occurring during the course of the event.

Grouping based on challenges to a fundamental safety function results in the following groups:

- Challenge to heat removal;
- Challenge to reactivity control;
- Challenge to confinement of radioactivity;
- Challenge to control of chemical attack.

Initiating events often challenge more than one safety function, as in the following examples:

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<sup>2</sup> Event sequences that result in the highest accident analysis consequences are not necessarily the most limiting from the standpoint of SCCs being shown to meet their associated engineering design acceptance criteria.

- *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 must be analysed for dose to the public and workers. They include all leakages greater than the normal operational leakage rate. Breaks with an accompanying loss of forced core cooling result in challenges to heat removal as well. Some pressure boundary breaks may also lead to air ingress, which challenges the control of chemical attack.
- *Primary system breaks in the interface with cooling water systems (i.e. heat exchanger tube breaks)*: These breaks may result in water ingress. Depending on the design configuration and pressure differences between the primary and water systems, there may be releases resulting in public or worker dose. Those water ingress events resulting in doses are included in this type of initiating event. If no doses result, the events might still be included in the reactivity transients or chemical attack sections.

Events may also be grouped according to dominant phenomena in the event sequence. Examples are:

- Primary system breaks;
- Loss of primary system heat sink;
- Air ingress events;
- Water ingress events;
- Reactivity transients;
- Depressurized loss of forced cooling (D-LOFC);
- Pressurized loss of forced cooling (P-LOFC);
- Turbine trip;
- Station blackout.

Whichever grouping scheme is selected, it is important that it be a consistent one, so as to avoid inadvertent omissions.

## 4. CONSEQUENCE ACCEPTANCE CRITERIA FOR ACCIDENTS

Consequence acceptance criteria for safety analysis of accident sequences are typically established by the national regulatory body so as to be consistent with the top level dose acceptance criteria. The safety analysis must show that the worker and public dose estimates are within the regulatory dose limits for the applicable event frequency category. Estimation of doses from a postulated accident involves consideration of the initial fuel failures and fission product releases, and transport and distribution within the primary coolant pressure boundary prior to the accident. It also requires a dynamic analysis of the plant response, coupled with evaluations and consideration of potential additional fuel failure due to high accident temperatures and/or chemical attack.

If significant additional fuel failures result, analysis of potential fission product release and transport (including retention, filtering and subsequent release) to the atmosphere, with on-site and off-site dose evaluations, must be included. Additionally, it must be shown that the overall risk to the public from the dose and frequency of all events combined does not exceed the specific risk guidelines established by the regulatory body. Radionuclide off-site (site boundary) dose limits will vary depending on the accident (frequency) category. Modular HTGRs are typically designed such that the usual 'leaktight' containment and off-site evacuation planning are not required to meet top level dose criteria (see Fig. 1).

Additional specific criteria are usually provided to ensure defence in depth, where the SSCs modelled in the accident analysis that provide accident mitigation are capable of performing their safety functions. These criteria involve both safety related SSCs for design basis accidents and non-safety related SSCs, which are potentially credited for mitigation in the analysis of AOOs and BDBAs. Limiting conditions may reference applicable national codes and standards, or may rely on specific qualification processes, with approval by the national regulatory body.

The specific criteria are determined by means of fuel performance models for the time at temperature (and chemical attack) histories of the core and fuel, and for the support structures for the core and RPV. The fractional release of fission products from CFPs depends not only on fuel temperature but also on the time at temperature [12]. Effects of irradiation must also be taken into account in specific criteria for fuel and for support structures. Specific engineering design acceptance criteria for pressure, temperature and displacement are also needed for the RPV and other primary pressure

boundaries, and for the confinement building plus any other accident mitigation SSCs that are credited in the accident analysis.

Additional design acceptance criteria are established to ensure plant investment protection. For example, acceptance criteria are generally established for the fuel, reactor vessel and vessel supports, to allow the reactor to be restarted following an AOO (or perhaps for some DBAs) without having to replace these SSCs.

The acceptance criteria for other reactor types [4, 5] may also be used as references to help determine acceptance criteria for modular HTGRs.

## **5. ANALYSIS METHODOLOGY**

Depending on the regulatory requirements and event category, a best estimate and/or conservative analysis may be required. The purpose of this section is to describe typical means of achieving a best estimate or conservative result. A conservative analysis is one that leads to pessimistic results relative to the specified acceptance criteria. However, a given accident sequence may involve multiple applicable acceptance criteria (e.g. dose, SSCs). In these cases, separate analyses are to be carried out to ensure that the conservative results and limitations for each problem area are well understood and achieved.

There are various means of performing best estimate and conservative analyses. Table 1 describes three such analysis methods: single point, root mean square (RMS) and Monte Carlo. To evaluate the 'results' for the conservative analyses, particularly for multiple case runs, one needs to establish an objective function (OF) that provides some measure of the severity of the accident. For a fuel failure acceptance criterion, for example, the OF may include measures of predicted peak fuel temperature, time and quantity at (excessive) temperatures, percentage fuel failure (from a fuel performance model) and fuel failure from oxidation, each with its own weighting factor.

To perform a Monte Carlo analysis, the important input parameters are assigned a predetermined probability distribution according to their expected variation or their uncertainty, typically a standard statistical bell curve with limits.

For each simulation run, a value for each important input is randomly sampled from its probability distribution. After many of these runs, the range of results for the OF can be plotted as a histogram or OF probability distribution. The best estimate output is the mean value of the OF, or the 50th percentile result. The conservative output from the Monte Carlo analysis is

TABLE 1. SELECTED ANALYSIS METHODS AND TYPES

Analysis method	Analysis type	
	Best estimate	Conservative
Single point	A single best estimate (nominal) analysis is performed using the most likely or expected values as input to the analysis.	A single conservative analysis is performed using conservative input values and assumptions (i.e. pessimistic OF with respect to an acceptance criterion).
Root mean square	A single best estimate (nominal) analysis is performed using the most likely or expected values as input to the analysis (same as single point analysis).	Multiple analyses are performed. In each case, a conservative value of one important parameter is used, while all other parameters are held at their nominal value to determine a sensitivity of the OF to each important input. The uncertainty is the root of the sum of the squares of the differences between the sensitivity and best estimate OF results.
Monte Carlo	The best estimate result is the mean or expected output of a Monte Carlo analysis, where the mean output is the arithmetic average. The mean is also the 50th percentile result. (See notes on Monte Carlo analysis in text.)	The conservative output from the Monte Carlo analysis is typically selected as the most pessimistic output (OF) with 95% probability, that is, the 95th percentile result.

typically selected as the 95th percentile result. The physical significance of the 95th percentile is that the probability of the outcome's being worse than this conservative result is 5%. The best estimate result obtained from a Monte Carlo analysis is judged to be more accurate than that from a single point or RMS analysis. Use of the Monte Carlo method allows for reducing the excessive conservatism in the design, since the results (OF) are more realistic than those of the RMS method. If the regulators allow both methods, which method to adopt is usually an economic decision.

The idea that the Monte Carlo approach gives better statistics assumes the use of accurate models of the data as inputs. In practice, and for the modular HTGRs in particular, one deals with passive systems (such as RCCS) that have not been built or tested. Also, many models used in the simulation are quite simplified, so extra benefit may be derived from the use of the more complex methods.



## **6. ANALYSIS OF EVENTS RELEVANT TO MODULAR HTGRs**

The purpose of this section is to describe typical event sequences that should be analysed and characteristics of the tools (models and data) needed to perform the analyses. These event sequences are selected by the process described in Section 3, but particulars of the sequences may need to be modified based on what is carried out in the analyses. Reactor sequences are described in the general categories of loss of (active) heat removal, break or leak in the primary system, loss of reactivity control and unexpected pressure transients. Any given sequence may 'cross over' to one or more of the other categories, unless, for example, the frequency of such a cross-over event is lower than the cut-off frequency. The event descriptions are geared mainly to analysis needs.

Although not emphasized here, events allowed only very low dose consequences (AOOs) are typically simulated using whole plant models (see Section 7.3).

### **6.1. HEAT REMOVAL TRANSIENTS: LOSS OF FORCED CIRCULATION UNDER PRESSURIZED CONDITIONS**

#### **6.1.1. Initiating events**

Events involving a P-LOFC are assumed to occur during power operation, where the primary helium flow stops and the primary system remains pressurized. A P-LOFC may result from a variety of initiating events or event sequences. In fact, in typical HTGR designs, the primary helium flow is intentionally stopped by the reactor protection system upon shutdown of the reactor, and the shutdown cooling system (SCS) is then started up to remove the afterheat. Two groups of events or event sequences can lead to a P-LOFC: a reactor shutdown with a failure of the SCS (or other shutdown cooling mechanism) to provide forced cooling, and a station blackout that results in a loss of forced circulation. The first group includes events that directly lead to a P-LOFC, such as loss of power supply or loss of grid load, or failures of the primary circulating equipment (helium circulators or turbo machines). The second group includes all other events or accidents that actuate the reactor protection system and stop the primary helium flow. A primary piping leak or rupture can also lead to a P-LOFC if the leak is successfully isolated, for example, by closure of a local or remote manual valve.

### **6.1.2. Consequences and safety aspects**

There are two major safety related aspects of a P-LOFC. The first is the core heat-up transient and the potential for delayed radioactivity release from the fuel. The reactor core will heat up due to the decay heat, but because the primary system is still under pressure, natural circulation of helium within the core will help equalize the core temperatures, with the maximum temperatures appearing near the top of the core. The maximum fuel temperatures in P-LOFC events are highly dependent on the design, but typically are well below prescribed limits.

The second aspect is the heat-up, during P-LOFC conditions, of metallic structures and equipment, in particular the primary system pressure boundary and other important mechanical and structural components. Some important metallic structures — such as control rod sleeves, the core barrel and the RPV, and their mechanical support structures — will usually experience a temperature increase. It is important to determine the potential impact of such temperature increases on structural integrity. Such design analysis calculations are also needed to verify that there is a valid structural/mechanical basis for the SSC modelled and credited in the accident analysis. The output of thermo-hydraulic conditions from the accident analyses codes is an important input to such confirmatory structural/mechanical analyses.

In modular HTGR designs, provisions are usually made to limit natural circulation through the entire primary loop under accident conditions to prevent the overheating of metallic equipment such as connecting piping, helium circulator or turbo machines, and heat exchangers that may lose circulating coolant. Analysis of event sequences involving such natural circulation is essential.

Although the RCCS is considered a safety system, postulated RCCS failure and degradation are of interest, in particular owing to the potential for adverse effects on RPV temperatures during P-LOFC events.

### **6.1.3. Modelling and data needs**

Analysis of the P-LOFC accident requires two or three dimensional modelling of the core thermohydraulics to calculate heat transfer by conduction, natural convection and radiation, and to evaluate the temperatures of the fuel, graphite and metallic components in the core and vessel regions. Detailed modelling of the heat transfer from the fuel through core internals and the reactor vessel to the surrounding RCCS is essential. Models of a failed or degraded RCCS may be required. Heat transfer correlations for conductivity, convection and radiation (emissivity) coefficients, as well as heat

capacity, depend on a number of factors such as temperature, reactor operational history and special material properties, all of which must be taken into account in the modelling. An uncertainty evaluation is also necessary.

## 6.2. PRIMARY SYSTEM RUPTURE TRANSIENTS

Different break locations and sizes are analysed for their possible effect on:

- Prompt and delayed releases of fission products;
- Air or water ingress;
- Dust resuspension;
- Potential effects on control rod position (control rod ejection; see Section 6.3.1);
- Pressure differential forces on internal components;
- Pressure transients in the confinement building compartment.

### 6.2.1. Loss of forced circulation with depressurization

#### 6.2.1.1. *Initiating events*

In D-LOFC events, starting from power operation, the primary helium inventory is lost to the point that the primary system is depressurized to atmospheric pressure. Loss of forced circulation with depressurization conditions may come from primary pressure-containing equipment failures such as large leaks or piping ruptures that are not isolatable. The opening of primary system safety valves without reclosure may also lead to D-LOFC conditions. There may be some other event sequences that do not involve primary piping leak or rupture but also lead to D-LOFC conditions, depending on the plant design. In some cases, ‘leak before break’ assumptions may be allowed by regulatory bodies for large pipe or vessel rupture scenarios. The technical basis for crediting the effectiveness and timeliness of helium leak detection versus potential vessel crack propagation rates for applicable vessel materials and conditions needs to be established.

#### 6.2.1.2. *Consequences and safety aspects*

The D-LOFC event is known to be more limiting than the P-LOFC event, mainly owing to the lack of natural circulation mixing in the core. The primary consequences of D-LOFC accidents are core heat-up and potential

radioactivity release (immediate and delayed) into the confinement building and eventually into the environment (filtered and/or unfiltered). Compared with P-LOFC conditions, natural circulation of helium in the core is negligible. For a long term D-LOFC, maximum fuel temperatures typically reach peak values in a few days, near the middle of the core height, and then begin a long, slow decrease. In general, the reactor must be designed so that the maximum fuel temperature will not exceed a level at which a significant fuel failure fraction would occur. Typically, the reactor maximum core power level is selected based on this calculation of maximum fuel temperature in a D-LOFC accident. The heat-up of metallic structures and potential material damage also need to be evaluated.

During the initial depressurization process, much of the circulating activity is released from the pressure vessel system along with the outgoing helium. Depending on the size of the break, the exiting helium would also entrain and transport out some of the radioactive graphite dust that settled out in the primary system during normal power operation. Depending on the design of confinement, these large prompt helium releases would typically be unfiltered.

During the core heat-up process, radioactivity released from the fuel particles increases, although the core heat-up usually results in only limited additional fuel failures. These particle heat-up failures form the source term of delayed radioactivity release. Once the system is depressurized, the release is a slow process, and any discharges from the primary system are usually largely filtered out by confinement system equipment to avoid significant delayed releases to the environment.

Piping breaks and the depressurizations that follow cause pressure distribution transients within the reactor, which could be very significant for large breaks. Pressure redistributions need to be evaluated to ensure the structural integrity of the reactor internals. It may be necessary to analyse pipe whipping effects on surrounding structures or equipment and to check the potential for adverse impacts of the exiting helium on safety related equipment. Also, the pressure and temperature transients at different places inside the reactor confinement building need to be evaluated to assess the functionality of important structures and equipment.

Another possible event category involves very small breaks that result in depressurization times of the same order as the time to reach the maximum fuel temperature during a D-LOFC (i.e. days), and in a loss of forced cooling at the same time that the break occurs. This accident is a combination of a P-LOFC and a D-LOFC. The interesting aspect of this event sequence is that, with the lower (average) pressure compared with the P-LOFC, there is a greater chance of higher fuel temperatures (and fuel failure), and with the

higher (average) pressure compared with the D-LOFC, there is a greater driving mechanism for transport of fission products out of the confinement building. The tools needed for this analysis would be a subset of those used for P-LOFC and D-LOFC cases.

Depending on the nature of the depressurization process, the reactor core could experience cooling due to a rapid flow discharge of the helium, which might have a reactivity effect as well.

#### *6.2.1.3. Modelling and data needs*

For analysis of the D-LOFC depressurization phase, detailed thermo-hydraulic computer codes and modelling are necessary to evaluate the pressure and temperature transients at different places inside the RPV and reactor confinement building.

For evaluation of the source terms, specialized computer codes and mechanistic models are necessary to calculate fission product transport and release. For evaluation of the immediate release, the needs include an estimate of the circulating activity, along with a credible estimate or model of the release of graphite dust with adsorbed radionuclides. For analysis of delayed releases, modelling is necessary for the calculation of fuel failure from core heat-up plus the fission product retention capability of the primary system and reactor confinement building, including filtering.

### **6.2.2. Rupture with air ingress**

#### *6.2.2.1. Initiating events*

Air ingress into the primary system is a safety concern because of the potential for oxidation damage to graphite structures and components within the vessel, and to the fuel (TRISO particles). At the operating and accident temperatures that would be seen in the core following a D-LOFC, significant oxidation is a distinct possibility. The extent of the air ingress flow rates and the oxygen content of the available air are dependent on a wide variety of possible reactor and reactor cavity design features, initiating event factors and subsequent accident progression scenarios. Analyses of these eventualities must therefore cover a wide spectrum of cases, and some quantification of the uncertainties and consequences needs to be made. Air ingress accidents are typically categorized as very low probability events (BDBAs) [7]; however, they are considered to be very specific to modular HTGR designs and of much interest to designers and regulators, and need to be thoroughly analysed.

An air ingress event caused by a primary system leak or break starting from nominal operating conditions is usually assumed to follow complete depressurization in a D-LOFC accident. Depressurization of the primary system to atmospheric pressure is a prerequisite for (atmospheric) air to enter the primary system. Air ingress during a normal shutdown is not considered here, since the core graphite temperatures are below those at which any significant oxidation or corrosion would occur. The major variations of this scenario include the break size and location (which would affect the initiation time and the rate of air ingress flow from natural convection) and the oxygen content (versus time) of the ‘air’ at the intake point. A potentially significant accident sequence would be one where an SCS was inappropriately started by the operator following depressurization, in which case the forced circulation flow (possibly containing air) could be much greater than in the natural circulation flow case.

#### 6.2.2.2. *Consequences and safety aspects*

In larger D-LOFC events, vessel or other primary system breaches would result in a relatively rapid blowdown of the helium into the reactor (and/or) power conversion unit (PCU) cavity, displacing the air in the cavity. The resulting atmosphere for the duration of the potential ingress event would depend greatly on whether the confinement system was designed to release the initial discharge of the gas to the atmosphere (where the primary system circulating activity is likely to be low) and then ‘re-seal the confinement building’, or if the discharge is confined in a ‘leaktight’ conventional confinement building. The former case is considered the preferred design for modular HTGRs for safety reasons.

There are two major variations of the event affecting natural circulation air ingress flow: a single and a double break in the primary system.

- For the single break case, there is typically a long period (usually a number of days) during which the helium remaining in the reactor vessel mixes (via molecular diffusion) with the outside air before natural circulation of air through the core is initiated owing to the higher density gases (e.g. nitrogen, oxygen) entering the core. The time at which this transition from diffusion to natural circulation occurs needs to be calculated. The calculation depends on the details of conditions within the vessel and on the break location and size, and is sensitive to minor external disturbances and any actions that may be taken by the operators in the interim.

- While the double break case is of a considerably lower probability, analysis may be required unless the frequency of the event can be shown to be below the cut-off frequency. In this case, assuming that there is direct outside air access to the top and bottom of the core, forming a ‘chimney’, the ingress flow would begin soon after depressurization, and the ingress flow rates typically would be much higher than those of the single break location.

Natural convection ingress flow rates are usually limited to relatively low values owing to the high core flow resistances and resistances in other parts of the flow paths.

Typical sensitivity scoping calculations of air ingress accidents have indicated that, although the oxidation rates for reactor grade graphite are quite low, the oxygen in the entering air is typically completely consumed well before it exits the core. In fact, at least for the first several days of significant ingress, analyses show that most of the graphite oxidation occurs in the graphite core support blocks below the core and the graphite lower reflector, with relatively little oxygen reaching the active core. Also, any heat released from oxidation in the active core typically affects only the lower regions and does not add to the peak fuel temperatures that would be reached under non-air-ingress accident conditions.

Scoping calculations have shown that, if the oxygen in the confinement building volume is limited to approximately that initially present in the confinement building space (or less, considering displacement by the helium), the total damage to the core would probably not be significant, and would take place mostly in the core support and lower reflector regions. Variations of that scenario have also shown that if the onset of significant oxidation occurs late in the D-LOFC accident sequence (~3–5 d or later), the lower regions may have cooled down to temperatures below the range of significant oxidation rates, in which case oxidation would be more likely to occur further up the core, into the fuel regions. Studies in Japan have shown that TRISO fuel retains fission products even when the OPyC layer is oxidized in the temperature ranges predicted for the lower core regions [13].

#### 6.2.2.3. *Modelling and data needs*

Since the oxidation rates for graphite are very sensitive to temperature, rather fine structure nodalization of the core lower support structure and reflector regions, in addition to the fuel areas, is recommended for determining any potentially significant loss of mass and strength in critical areas. At least a two dimensional, and preferably a three dimensional, core thermohydraulic

model with oxidation modelling would be advisable. The model should account for the differences in rate equations for the various types of graphite used in the structure and the fuel regions.

The graphitized shell used in pebble fuel has higher oxidation rates (at a given temperature) than those in prismatic fuel blocks, since during manufacture the pebbles cannot be heated to high annealing temperatures because the fuel is encased in the shell.

Graphite oxidation rate models need to account for the differences in the lower temperature ‘chemical’ range and the mass transfer limited rates in the higher temperature ranges [14]. The differences in prism block designs also need to be accounted for, since there are variations in the protective graphite shells around the fuel compacts. (A thinner shell reduces the temperature rise between coolant and fuel.)

Model experiments have shown that oxidation of fuel elements is typically very non-uniform, so it is not safe to assume that a pebble’s protective shell, for example, oxidizes uniformly. Gas flow and oxidation rate estimates typically do not account for core geometry changes, and are progressively less realistic as the percentage of the total graphite oxidized in a given region increases. Additional information and data on and models of the effects of oxidation on fuel particle failure rates for irradiated fuel forms are provided in Ref. [12].

Modelling and data needs for the D-LOFC and P-LOFC would also apply here, since the air ingress accident is an add-on to these cases. For scenarios involving operation of an SCS, models and design data for that system would be required as well, including details of potential access to air at the intake.

The availability of oxygen in the ingress air is probably most crucial to the final outcome of core and other structural damage if the oxidation is not stopped by other means. It has been found to be difficult to limit air leakage into a large confinement building volume for situations like those following a D-LOFC in which the initial primary system inventory release has been vented [15]. Data from representative experiments may be needed to check models of these effects to enable determination of a validated range of expected or required limits on leakage rates, and thus the oxygen potentially available to the core.

Accident scenario predictions that indicate a possibility of serious fuel or core support structural damage for ‘credible’ (though very low probability) accidents may indicate the need for a very careful and detailed analysis with sensitivity studies, as well as, perhaps, some design modifications.

The scenarios described noting ‘scoping calculation results’ are not necessarily applicable to specific designs and situations that may be encountered, but are noted to indicate possible courses the postulated accident



progressions might take. For specific designs, benchmarked computational fluid dynamics codes have been used to model diffusion of air into the core to calculate the time to the onset of natural circulation.

### **6.2.3. Rupture with water ingress**

#### *6.2.3.1. Initiating events*

Water/steam ingress into a modular HTGR core can result from steam generator heat transfer tube leaks or breaks in steam cycle designs, where the pressure of the secondary water/steam is usually much higher than that of the primary helium. Design provisions are usually made to limit the effect and amount of water ingress, such as isolation of the water source and evacuation of the high pressure water inventory. There are also lower pressure water sources in both steam cycle and Brayton cycle designs, such as water in the secondary side of the heat exchanger of the SCS or the helium circulator cooling system, or in precoolers and intercoolers. Event sequences involving tube failures in such coolers are unlikely to result in water ingress, since water pressures typically are kept below helium pressures; nonetheless, there may be some sequences that could lead to some water ingress. However, experiments conducted in Japan indicate that, because water seeks its own level, some liquid water can flow from the water side to the helium side of a failed heat exchanger tube for transport as water vapour into the graphite core [16].

#### *6.2.3.2. Consequences and safety aspects*

Water/steam ingress into a hot reactor core causes three major safety concerns, namely, a positive reactivity insertion, chemical attack and a breach in the radioactivity confinement.

Modular HTGR cores are usually undermoderated by design, so a moisture ingress event can cause a positive reactivity insertion. An undesirable side effect could be the reduction of control rod worth. These effects depend on the degree of undermoderation and the total mass of moisture penetrating the core. These positive reactivity insertions could cause large transient increases in reactor power.

Chemical attack by moisture causes oxidation and corrosion of the graphite material in the core and, if exposed, the CFPs as well. It could also challenge the structural integrity of graphite reactor internals and fuel elements. The reaction of moisture with graphite causes an increase of primary pressure and produces gases, including carbon monoxide and hydrogen, which would present additional safety concerns. Unlike the chemical reactions

between graphite and air, graphite–water reactions at high temperatures are endothermic.

In steam cycle designs, steam generator tube ruptures can cause pressure surges in the primary system and lead to the wash-off of radioactivity plated out on tube surfaces and possibly to its release out of the confinement building. This could significantly add to the source terms.

#### *6.2.3.3. Modelling and data needs*

Water ingress events typically involve complex interactions of neutronics, thermohydraulics, chemical reactions and radioactivity releases. Detailed computer codes and models would be needed to calculate the rate and amount of water/steam ingress, the reactivity effects and resulting power transients, pressure and temperature transients, production of oxidization gases, and the added radioactivity source terms. Uncertainty analyses are likely to be necessary to facilitate understanding of the possible range of accident parameters.

### 6.3. ACCIDENTS ASSOCIATED WITH REACTIVITY CONTROL

#### **6.3.1. Reactivity events**

##### *6.3.1.1. Initiating events*

Reactivity events are those in which positive (or negative) reactivity is accidentally or inadvertently introduced for various reasons. Typical initiators causing reactivity events are:

- (a) Control rod or rod bank withdrawal;
- (b) Control rod ejection;
- (c) Inadvertent control rod movement;
- (d) Accidental reactor shutdown;
- (e) Increase or decrease of the primary heat removal rate;
- (f) Compaction of a pebble bed core caused by earthquake;
- (g) Break of heat transfer tubes in a steam generator or other water cooled heat exchanger that could result in water or steam ingress into the core;
- (h) Control rod drop;
- (i) Fuel loading error.

Note that control rod ejection events (item (b) above) may also be associated with a primary system rupture (control rod housing failure) (see Section 6.2). The event probability may be reduced and potentially eliminated from analysis consideration by design. There is a wide variety of initiators capable of causing reactivity disturbances. For most of them, the relatively large negative temperature–reactivity feedback coefficient typical of modular HTGRs tends to make these events inconsequential from a safety standpoint. Some of the more interesting initiators are rod movement due to loss of control of the control rod drive system or operator error, water or steam ingress into the core region, earthquakes and loss of control of the fuel element (pebble) handling system.

### *6.3.1.2. Consequences and safety aspects*

A continuous control rod withdrawal causes a rapid increase of reactor power and typically results in an automatic reactor shutdown. The amount of reactivity inserted depends on many factors, including the initial state of the reactor and the initial position of the control rods. With a rod control system failure or operator error, the primary heat removal rate can be increased or decreased, leading to a decrease or increase of helium and fuel temperature, and consequently a positive or negative reactivity change in response to the initial event. In another type of failure of a control rod drive system, a group of control rods may move up or down inadvertently, leading to an abnormal power distribution. It is worth noting here that, owing to its ceramic features, the core heat-up response is slow, and such an event does not necessarily imply imminent loss of fuel integrity as in other reactor designs.

In an earthquake, positive reactivity can be inserted (in a pebble bed core) by two mechanisms. One is the reduction of core porosity (pebble compaction from shaking) and a consequent reduction of neutron leakage. The other is an upward shift in the positions of the control rods relative to the top surface of the active core.

The effects of leaks or breaks in the heat transfer tubes of a steam generator or pressurized water heat exchangers may be classified as a reactivity event as well as a water ingress event (see Section 6.2.3).

The negative temperature–reactivity coefficients play an important role in the reactivity events. Reactivity changes due to variations in reflector (centre and side) temperature changes in LOFC accidents are very slow but can have an effect on recriticality in an anticipated transient without scram (ATWS) and on shutdown margins.

### 6.3.1.3. *Modelling and data needs*

For reactivity accident analyses, a neutron kinetics simulation model, either a point or a one dimensional approximation, is typically coupled to a thermohydraulics code. The choice of a point or spatial kinetic representation of the neutronics will depend on the nature of the transient being analysed. For events such as a control rod ejection, where a large spatial flux peak is expected, a two or three dimensional core model would be needed to provide accurate results. In addition, detailed spatial nodalization may be needed to track the rapidly changing spatial core temperature distribution. The reactivity insertion, reactor power and maximum fuel temperature are key safety parameters. For determining recriticality, a two or three dimensional core thermohydraulics model would be needed to track the average temperature of the core and reflectors.

In reactivity modelling analysis, because of the difficulties of obtaining accurate heat balances in HTGR systems, it is advisable that a reactor power measurement deviation error be added to the nominal reactor power. Any assumed positive reactivity insertion rate should also be conservative, based on uncertainties in rod movement speeds and rod worth. Typically, the delay times assumed for shutdown action and the rod insertion times should be conservative, although for most credible HTGR accidents these factors are inconsequential because of the very slow responses. The control or shutdown rod with maximum worth is typically assumed conservatively to be blocked (insertion prevented). Because the core negative temperature reactivity coefficient plays an important role for all reactivity accidents, it is typically given a conservative value.

Control rod withdrawal accidents from the cold startup condition need to be analysed owing to the potential available reactivity in the core, temperature compensation, xenon poison and power adjusting reactivity.

For analysis of a control rod withdrawal accident from at-power conditions, various initial power levels should be analysed in order to determine the most severe case. Normally, the neutron flux (or reactor period) is the first parameter to reach a trip point. Conservatively, the first shutdown signal may be ignored and later ones used to initiate a scram.

For analysis of pebble bed compaction events, a conservative initial porosity should be input to cover all possible reactivity insertions.

To simulate reactivity events caused by water/steam ingress, the ingress rate and total ingress might be simulated by a thermohydraulic code before the reactivity insertion is calculated by a reactor physics code. The accident could then be approximated by a transient analysis code in which the water induced reactivity transient is introduced as an external input.

From the viewpoint of fission product transport and release, water ingress into the core can increase fission product release from the fuel kernels of failed CFPs. Water ingress can also introduce additional fission product transport mechanisms within the reactor system. These transport mechanisms must be considered when evaluating fission product release for breaks in the helium pressure boundary.

### **6.3.2. Anticipated transient without scram**

#### *6.3.2.1. Initiating events*

Normally the initiating event for an ATWS event sequence is an AOO followed by a failure to shut down the reactor. If the frequency of event for an ATWS is lower than the cut-off frequency, analysis may not be required; however, if the event is considered as one of the ‘cliff edge’ events (see Section 3.2), analysis may be required.

Typical ATWS initiating events are:

- (a) Loss of off-site power;
- (b) Loss of heat sink;
- (c) Loss of forced cooling under pressurized or depressurized conditions;
- (d) Withdrawal of one group of control rods;
- (e) Compaction of a pebble bed core caused by a safe shutdown earthquake (SSE);
- (f) Ingress of water or steam into the core (causing a positive reactivity insertion).

#### *6.3.2.2. Consequences and safety aspects*

In a modular HTGR, there are normally two active reactivity shutdown systems in addition to the normal reactivity control system. One is the normal shutdown system. When a scram signal is received, it actuates automatically, inserting sufficient negative reactivity (typically, control rod absorber material) into the core. The second is the reserve shutdown system, typically employing small neutron absorbing balls. If the first shutdown system fails, the reserve shutdown system can be actuated manually by the control room operator.

In an ATWS, the normal shutdown system is assumed to fail. Analyses of extreme cases may be required to assess the consequences of longer term ATWS accidents assuming failure to actuate the reserve shutdown system.

In an anticipated transient, when a reactor protection parameter reaches its trip set point, a scram signal is sent to the protection system. A series of

protective actions is then taken, such as dropping the control/safety rods, stopping an electric powered primary helium circulation blower (or BOP turbine driven compressors in a direct cycle), closing a blower flap and isolating the secondary system. In an ATWS sequence, typically only the scram action is assumed to fail, and other protective actions are considered to be successful. For this ATWS sequence, the transient is a combination of two factors: loss of normal cooling and no scram.

With a loss of forced helium flow, the average fuel temperature increases and is ultimately controlled by the negative temperature reactivity coefficient, which quickly brings the reactor to subcriticality, reducing the fission power to zero. This loss of fission power causes the maximum fuel temperature to decrease even as the average core temperature increases.

Owing to the buildup of xenon poison, the reactor typically remains subcritical, at least until sometime later, when the xenon decays (1–2 d). Once the xenon concentration drops sufficiently, the reactor will become critical again, and following recriticality, the increasing reactor power may oscillate and gradually achieve a stable low power level and core temperature, which depend strongly on the total residual reactivity and the net core heat removal rate.

The combination of low core power density, large reactor core heat capacity, high core thermal conductivity and large thermal margin to fuel failure allows for long delay times before manual operator actions, such as insertion of the absorbing balls into the side reflector, must be taken to avoid fuel failure. Even with only limited reactivity shutdown capabilities, it may be possible to attain hot shutdown but not cold shutdown conditions. The RCCS may also play an important role in this sequence, preventing the temperature of the RPV from exceeding its limits.

### 6.3.2.3. *Modelling and data needs*

In analyses of ATWS events, it is assumed that all control and safety rod positions are fixed and that no rods drop in response to scram signals. Other protective actions, such as core heat removal via the RCCS, are assumed to be successful. However, there may be situations where other assumptions result in adverse consequences; for example, the termination of active cooling is a protective action in accident conditions, where failure of such action in an ATWS can represent a serious hazard, and these eventualities also need to be considered. The reactor power, primary pressure and maximum fuel temperature must be carefully evaluated for the short term responses. The temperature histories of key components such as the core barrel also need to be assessed against acceptance criteria. The transient xenon negative reactivity

effect also represents an important contribution to the dynamic response of the core. A detailed three dimensional analysis and a coupled thermal–neutronic dynamic core treatment may be needed for accurate analysis of ATWS events.

In a conservative analysis, uncertainties in measurements and modelling are to be taken into account, and either conservative value or uncertainty analyses need to be performed.

## 6.4. ACCIDENTS ASSOCIATED WITH PRESSURE TRANSIENTS

### 6.4.1. Initiating events

Significant pressure transient effects for all initiating events are considered in this section. The effect of pressure transients both within and external to the helium pressure boundary (i.e. within the confinement building) needs to be considered.

The following events are considered for various break assumptions, both within and at pressure boundaries:

- Breaks resulting in differential pressures across cavity walls;
- Turbine trip;
- Loss of load;
- Gas cycle valve failures (closed or open);
- Internal pipe failures;
- Water ingress;
- Any transient that increases the overall pressure.

### 6.4.2. Consequences and safety aspects

Within the helium pressure boundary, pressure transients may cause increased differential pressures across components, especially the core structures. The effect on core geometry needs to be carefully considered. Should a pressure increase reach the point where a pressure relief valve opens, release of radioactivity needs to be considered. In addition, it must be ensured that the pressure relief valve can vent any feasible increase of pressure sufficiently rapidly to prevent overpressure of the helium pressure boundary vessels, connected piping and pressure retaining components. Both short term shock waves and longer term pressure rises need to be considered. The jet forces at the relief valve location also need to be considered (both the reaction force on the pressure boundary and the impact force on confinement building walls).

### **6.4.3. Modelling and data needs**

In general, a two dimensional compressible gas dynamics model is required. In select locations, a three dimensional compressible gas dynamics model is necessary.

## **6.5. OTHER SOURCES OF RADIOACTIVITY: LOSS OF PEBBLE BED REACTOR SPENT FUEL STORAGE FORCED COOLING**

Modular HTGRs have sources of radioactivity outside the main helium pressure boundary circuit. Examples include the:

- Spent fuel storage;
- Waste handling system;
- Helium inventory control system;
- Fuel handling system.

Typically, these radionuclide source terms are much smaller than that associated with the reactor core, and there are only a few release mechanisms. Nonetheless, the dose consequences associated with the failure of these systems need to be analysed. Spent fuel storage for a pebble bed reactor is used here as an example.

For a pebble bed reactor, spent fuel is typically stored in large tanks, where decay heat removal is achieved by forced convection on the outside surface of the tanks. The tanks are filled with helium at a pressure above atmospheric pressure such that the spent fuel remains in an inert environment. The temperature of the spent fuel is also kept low enough that corrosion does not occur with air ingress. The tanks are also designed such that the fuel remains subcritical even if the pebble bed is compacted owing to a seismic event, or if there is additional moderation due to water ingress.

### **6.5.1. Initiating events**

A possible initiating event is a station blackout, which would stop the forced convection cooling on the outside of the storage tanks. Decay heat removal by passive means would be employed. Another initiating event for pebble bed reactors is a failure of the pebble transport lines that load spent fuel into the tanks, which may result in air ingress. Depending on the temperature of the spent fuel, corrosion may take place. Depending on the event category and the safety classification of the forced convection cooling system, a



postulated failure of the pebble transport line with a concurrent loss of forced convection may need to be considered.

### **6.5.2. Consequences and safety aspects**

In the example discussed here, the spent fuel and tank temperatures increase, and it must be determined if the passive heat removal system is sufficient to keep the fuel and tank within temperature limits, typically  $\sim 400^{\circ}\text{C}$ . It will also need to be determined if there is air ingress. Should that occur with fuel temperatures above  $400^{\circ}\text{C}$ , corrosion and potential release of radioactivity would need to be considered.

### **6.5.3. Modelling and data needs**

The cases described might have model requirements similar to those of a D-LOFC with air ingress, but with simpler geometry needs and data applicable to lower temperature oxidation. The time at which the accident occurs during the life of the power plant will determine how full the tanks are, as well as the activity (afterheat) in the tank pebble population.

## **7. RECOMMENDED FEATURES OF ACCIDENT ANALYSIS CODES**

The computer codes currently used for modular HTGR accident analysis have a wide variety of features and capabilities. The range of capabilities and the verification and validation of these codes are important in determining how effectively they can be used in assessments of postulated accidents and safety cases. This is especially true at the present time because of the lack of completed reactor designs to evaluate and deficiencies in the supporting databases. More work is needed on the development of a consensus design, a comprehensive suite of safety analysis and design analysis codes, accident cases for regulatory acceptance, and a robust supporting experimental database. Until these are available, significant uncertainty will remain for the types of accident to be reviewed and the analytic approaches to be used. Some general guidance for the development, verification and validation of accident analysis codes is provided in section 6 of Ref. [3].

The following sections provide summaries of the recommended features and capabilities of codes used in particular areas of accident analysis.

## 7.1. NEUTRONICS AND THERMOHYDRAULIC CODES

Accident codes typically model the core neutronics and thermohydraulic behaviour in both steady state and transient conditions. At present, these codes normally use two or three dimensional thermohydraulic models of the reactor core. The use of a multidimensional model is recommended to simulate the temperature profiles and flow distributions in the tall cylindrical (annular) cores, particularly when local or asymmetric effects are important. In P-LOFC accidents, for example, there are significant recirculation (bidirectional) flows within the core. In transient analyses, point kinetics approximations are often used for the neutronics because the three dimensional flux shape transient effects in the core during postulated accidents (other than rod ejections) are relatively small, especially when compared with those in light water reactors. Multidimensional neutronics codes are necessary for providing steady state power peaking distributions for accident codes using point or one dimensional neutronics models. For ATWS events, the dynamics of xenon and samarium poisoning are considered. These codes are sometimes also used in combination with the overall system thermohydraulic codes.

The following is an annotated checklist covering some of the recommended features for codes used for neutronic thermohydraulic accident simulations, noting features needed when coupling the neutronics with a multi-dimensional thermohydraulic core.

### *Thermohydraulic features:*

- Dose and temperature dependent graphite thermal properties, noting the relatively high core effective thermal conductivity and considering annealing effects, particularly in prismatic cores, during long term heat-up (LOFC) accidents.
- Reactor pressure vessel heat removal by the RCCS in LOFC events, where most (typically ~70–90%) of the heat from the RPV is transferred to the RCCS by thermal radiation, and the balance is transferred by natural convection in the reactor cavity air.
- Maximum fuel temperature plus time at temperature (the critical limiting factors) for all fuel regions, in order to provide inputs to fuel failure models to determine a source term.

- Core pressure drop correlations (pebble bed): Standardized and well documented correlations are available. These show the pressure drop to be very sensitive to assumed packing fractions of the pebbles.
- Core heat transfer algorithms: Correlations are available for the prediction of heat transfer in HTGR cores. These should include conduction, radiation and convection.
- Changes of flow direction within the core, including recirculation within the core in P-LOFC accidents.
- Chemical reactions in air or water ingress accidents.
- Irregular coolant flow paths for pebble beds: Detailed pebble bed analysis may require a fuel management capability.
- The modular HTGR core is relatively large, so from the fuel zone to the RPV there are large temperature gradients. Likewise, there are many coolant flow paths within the RPV, with significant flows typically bypassing the fuel regions. The amount of bypass flow is difficult to estimate owing to variations in gap sizes (due to thermal gradients and irradiation induced deformations). Bypass flow has a significant effect on maximum fuel temperatures during operation.
- For fast transients, especially, detailed temperature profiles of the fuel and graphite are to be taken into account for thermal stress calculations.
- There are several non-flow cavities within the RPV. The code must be able to deal with these cavities reasonably.
- For the pebble bed type HTGR, the pebble bed is a type of porous medium. The code is required to be able to simulate the flow and heat transfer in a porous medium.
- In some accident conditions, such as rupture of a steam generator heat transfer tube and accidents involving an open safety valve, critical flow occurs.

*Neutronics features:*

- The decay heat versus time after shutdown is a critical parameter in LOFCs.
- Negative temperature–reactivity feedback coefficients for the fuel and moderator are temperature and burnup dependent, with potentially positive coefficients for the central reflector.
- The neutronics codes need to account for core heterogeneity.
- To obtain correct resonance self-shielding of the cross-sections, it is necessary to model the detailed heterogeneity of the fuel. For example, in pebble beds it is advisable that the fuel particles not be smeared into the

graphite matrix, and that there be a capability for handling cross-sections dependent on temperature.

- The code must account for the time dependence of xenon and samarium effects on reactivity.
- Control rod movement: The analysis must account for the effect of asymmetrical control rod movements.
- While significant reactivity initiated accident events are not considered to be likely in current concepts, they cannot be ruled out entirely until the designs are finalized. The code must be able to model these types of event to a reasonable extent.
- If water can enter the core (e.g. in a steam cycle design), the effects of water ingress on reactivity and control rod worth need to be included.

## 7.2. FUEL PERFORMANCE CODES

Both mechanistic and empirical fuel performance models are used to predict fuel particle failure and fuel fission product release, and may be coupled with core wide fission product transport models to calculate release into the primary system and beyond. Releases will come from both intact and failed particles in the fuel during accidents, primarily depending on CFP time at temperature exposure. Releases predicted using empirical correlations of fuel performance and fission product release are to be based on test data developed for the specific fuel design, manufacture, operational service conditions and accident conditions. Mechanistic performance codes recently have been improved significantly, but are still at a point where fuel material property data and performance data would be needed to qualify any particular fuel, backed up by data from irradiation and accident condition testing. The complexity of coupling a mechanistic fuel performance model to an accident analysis code to develop three dimensional core wide particle failure rates and fission product releases generally makes such a modelling approach less practical than an empirical fuel performance modelling approach for modular HTGR accident analysis.

The modelling and material properties of the CFP are still under active investigation. Historically, pressure-vessel-like failures were the most common models, but work in the past few years has looked at additional, more complex and subtle mechanisms.

The temperature increase of the CFP during a heat-up accident increases the diffusion coefficient of the fission gas and metallic products, and eventually allows fission product release as the limiting temperature is exceeded. The release mechanisms also vary depending on the fission product inventories, as

some fission products can corrode the SiC or ZrC layer. In addition, a chemically active atmosphere due to air or steam ingress can damage particles or increase the release from already damaged fuel. Thermal decomposition of the SiC or ZrC pressure vessel is also of concern.

Typically, consideration must be given to:

- Particle design and operating conditions;
- Supporting databases;
- Fuel and fission product inventory at the beginning of an accident;
- Release/failure mechanisms during accident conditions;
- Initial particle failure fraction;
- Operating history prior to the event (fuel burn-up, fluences, etc.).

In summary, fuel behaviour codes can typically simulate (or at least bound) fuel fission product release mechanisms, provided that sufficient data are available to qualify the particular fuel type within the actual operating and accident condition envelopes. Until mechanistic fuel performance models are fully qualified and effectively coupled to accident analysis codes, it is expected that empirical fuel performance models will continue to be the most practical approach to modelling of the fuel fission product source term for modular HTGR accident analysis.

### 7.3. PLANT SYSTEM ANALYSIS CODES

There are several important basic design features related to the whole plant transient thermohydraulics of modular HTGRs that should be modelled, for example:

- Single phase inert coolant (helium).
- Various heat transfer modes: conduction, convection and radiation.
- Whole plant models analyse the coupled dynamic helium flow and heat transfer in the core, the turbine(s) and compressor(s), the recuperator and the helium/water heat exchangers (or steam generator system BOP). A model of the SCS must also be included.
- In order to model the primary circuit with the BOP, the code should be capable of simulating a one dimensional flow network.
- Because the helium may change its flow direction in some accidents, the code needs to be able to deal with such situations.

- To simulate reactivity events, temperature–reactivity feedback effects and effects of xenon poisoning are to be coupled to a neutron kinetics model.
- The RCCS plays an important safety role and ideally will be modelled and coupled to the RPV model.
- For accident studies involving a complete loss of flow through the BOP, the BOP dynamic response can usually be ignored, and the simulation will require a more detailed (multidimensional) core model to simulate the low or no flow conditions.

#### 7.4. CONFINEMENT ANALYSIS CODES

The confinement analysis code calculates pressure, temperature and gas composition transients in the confinement space during and following depressurization accidents. If subdivided, compartments and paths between compartments are modelled by nodes with given volumes and junction characteristics. Gas compositions calculated in the compartment(s) account for air, helium, products of oxidation and steam for water ingress accidents. Gas temperatures and compositions in the break area are used in the determination of ingress flows into the reactor vessel. Considerations are made for air in-leakage to the confinement space(s) and for any special injections of inert gas (by the operators) during a long term accident. For cases assuming damage caused by a rapid depressurization, changes in compartment or junction characteristics, or in relief valve sealing capability, would need to be modelled.

There are significant differences between LOCAs in a largely steam atmosphere (LWRs) and in a largely inert gas atmosphere (HTGRs):

- The lack of a phase change with helium as compared to steam may increase the importance of gas cooling due to contact with the structure.
- Unlike steam, helium cannot be condensed at the temperatures of interest.
- Aerosol and dust generation and transport may be different if the system is dry rather than wet.
- Reactor confinement building filters may have to operate (and survive) at higher temperatures.
- Organic forms of iodine can challenge filter designers confronted with varying conditions.
- Removal of fission products by dissolution in water may not be a viable option.

- A large pressure pulse could damage the RCCS, reducing cooling and/or opening up another release path.

## 7.5. CODES DEALING WITH CHEMICAL REACTIONS

Special code features are required to simulate chemical reactions due to air and water/steam ingress into the core. These models need to work in conjunction with thermohydraulic codes and (for water ingress) with codes that simulate the neutronics due to the steam/water effects on reactivity. Moreover, these chemical reactions can result in additional releases from damaged fuel and fuel failure.

Air ingress accidents typically assume that a D-LOFC is followed by ingress of ambient air or mixed gases in the confinement space in the vicinity of the break, affecting leakage into the primary system. This may occur either just after the depressurization is complete (to ambient pressure) or at some later time. The oxidation of core graphite, which follows, generates heat, in addition to the afterheat, and the air (or gas) flows subsequently provide for convective cooling (or heating) in the core. Oxidation of the core graphite can cause structural damage and expose fuel to chemical attack. This oxidation also releases the fission products contained in the graphite. The change in the reactor atmosphere from reducing to oxidizing may affect the plated out fission products on the metal components. These fission products could then be entrained in the convective flow and may exit the RPV.

Graphite oxidation models rely on data for specific types of graphite and their irradiation histories. For the lower temperature ranges, the oxidation rates are determined by the chemically controlled rate, while at higher temperatures, the process is mass transfer limited. These important distinctions must be included in the modelling. Less is known regarding how to handle the fission products.

Key factors are the net air flow rate into the reactor vessel and core, and ultimately the ‘availability’ of fresh air over the course of the accident. The net air flow through the core is strongly dependent on the buoyancy forces due to differential temperatures, and on the flow resistances in the core and at the break(s). Other scenarios that involve forced convection (blower operation) have the potential to cause significantly more core and core support structure damage, or at least to increase oxidation rates until the available oxygen is depleted.

For a single ‘break’ or opening in the primary system, calculations and experiments have shown that it may take a long time (i.e. days) before a significant natural circulation net air inflow is established. This process involves

the diffusion of air into the helium filled top region of the reactor vessel. For a much less likely case of a double break in the vessel, allowing access to both the top and bottom of the core, a chimney-like configuration could promote a higher net air flow that would be established more quickly.

Water and steam ingress modelling has many similarities to air ingress modelling, as both types of event have corrosive effects on the graphite and can cause structural and fuel damage. While water/steam reactions at high temperatures are endothermic, rather than exothermic, water/steam mixtures in the core are likely to increase reactivity, which could cause significant power increases if the accident also assumes an ATWS, so fuel damage could occur if the power increase is large. Owing to the extremely large range of uncertainties in terms of possible ingress rates and quantities, parametric studies would be required. Since the water cooled heat exchangers in gas turbine designs typically keep water side pressures lower than the primary helium pressure (except at shutdown), water/steam ingress accidents are usually only of concern for steam cycle plants, where steam pressures are much higher.

The analyst must not assume that water ingress is impossible just because the cooling water pressure is lower than the reactor operating pressure. Water could still enter the high pressure core if the break allows the water source to pressurize in a vertical configuration (water supported by gas), in which case instabilities could drive the water into the core.

Thus, consideration should be given to:

- Chemical reactions between graphite and air/steam;
- Coupling between thermohydraulic codes and chemical kinetics models;
- The formation of gases and aerosols containing fission products;
- Details of the flow paths;
- Graphite fission product inventories prior to the accident;
- Reactions between the released fission products and the oxidizing nature of the gas flow;
- Local fuel temperatures and an aggressive chemical environment that may increase fuel failures and failed fuel releases;
- Data available on TRISO fuel failure in oxidizing atmospheres;
- Damaging effects due to a reactivity insertion.

## 7.6. CODES DEALING WITH FISSION PRODUCT RELEASE, TRANSPORT AND DOSE

Internationally, many codes have been developed to simulate fission product transport. The codes usually start out with the releases that occur



during normal operation owing to damaged fuel and any matrix impurities. Over the operational history of the reactor, fission products move from the fuel and matrix into the graphite and then into the coolant, and the metallic fission products can plate out on the cooler downstream metal components. During an accident, higher temperatures, different core atmospheres and sudden increases in local flow can transport the fission products.

Transport of fission products encompasses two major aspects: the transport of fission products during normal operation and that during accident conditions. During normal conditions, the primary concern is the dose that maintenance personnel receive during component repair and replacement. As long as the fission products remain in the reactor system, this does not represent a public safety issue, but large fission product inventories outside the fuel can greatly complicate maintenance.

For accident conditions, the prompt release of fission products within the reactor system followed by the failure of additional fuel particles and transport of fission products must be modelled. If the core remains pressurized and the temperatures remain below the fuel failure limit, transport of the fission products through the reactor system may be enhanced, moving from the hotter to the cooler regions, but as long as no releases occur, the fission product inventory is limited to the primary system.

Releases from the reactor vessel can occur if a helium pressure boundary failure (e.g. opening of a relief valve) occurs. For a relatively small leak, it is likely that a reactor building filter train can effectively scrub all but the inert gases from the effluent (depending on the particular configuration). If a relief valve discharges into the reactor building or does not reset and overloads the filter system, a release path from the core can be established. Releases from medium or large diameter piping or cross-duct breaks are the most serious, since they will not be filtered. In this case, the release depends on the transport through the core, and on any settling in the reactor building and any building filtering. Once outside the reactor building, the standard dispersion codes can be used unless the reactor concept produces some unusual fission product form.

Transport modelling has proved to be formidable because multiple mechanisms are involved, including diffusion, sorption and desorption, gas phase transport, vapour phases, dust, and chemical reactions [17]. The physical parameters needed for the models are difficult to obtain and sensitive to the particular materials and surface conditions involved, the nature of which can change during the accident.

Generally, the fission products are broken down into two groups: the fission gases (including iodine) and the condensable radionuclides. Often the major radionuclides of interest are Kr, Xe, I, Cs, Sr, Ag and Te. After release

from the fuel, the gases make their way into the coolant stream, where the iodine chemisorbs on graphite, metallic components and circulating dust. Caesium is chemisorbed on graphite, the oxide layer on metallic components, and oxide and graphite dust particles. Strontium is believed to form a refractory oxide, although, if the coolant is reducing enough, it may exist in a carbide or elemental form. The oxide would exist on dust; the elemental form could exist in a vapour phase. Silver could exist in an elemental form, either in a condensed phase or as a vapour. It could move around as a dust, by sorption or, if hot enough, as a vapour. Tellurium is believed to behave somewhat like iodine.

To perform the detailed system calculations, the following data are needed:

- Diffusion coefficients in graphite, matrix, pyrocarbon, SiC and component metals as functions of temperature and radiation damage;
- Chemical form of the radioisotopes;
- Oxidizing properties of the coolant;
- Sorptivity properties of the materials at low surface concentrations as a function of temperature;
- Chemical properties of the surface layer likely to form on the metal components;
- Local temperatures of the core, ducts and power conversion system;
- Amount of dust present in the reactor system;
- Dust entrainment and transport during normal flow and accident flow conditions;
- Inventories of fission products (both outside and within the fuel) in both replaceable and permanent components.

Historically, this information has been difficult to obtain because of the high sensitivity to actual surface conditions and the complex and expensive test apparatus required. The issues identified in the confinement building outside the core will need to be addressed along with the dispersion data and plume models once the material leaves the reactor confinement building.

The transport of radionuclides influences dose design issues in two categories. The first is the dose received by maintenance personnel. Silver ( $^{110m}\text{Ag}$ ) and caesium ( $^{134}\text{Cs}$ ) are the two major isotopes for consideration. Contamination of the reactor ductwork and power conversion equipment increases personnel dose and can greatly increase costs and the difficulty of maintenance and repair. While remote maintenance techniques can be employed to circumvent this problem, the costs and system complexity increase. Both these radionuclides have a half-life long enough to make waiting for decay impractical.

The second issue is dose to the public from accidents (normal operation doses are expected to be very low). This dose is driven by releases from the reactor confinement building and can be mitigated to some extent by better filtering of any releases from the building if other means, such as fuel performance, fall short. The dose calculations depend on the rather complex transport to the reactor site boundary from the fuel kernel, internal reactor sorption and plate out. This is an area that needs more work and data to provide a convincing case for the safety of a design.



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**This report provides guidance on conducting accident analysis for nuclear power plants with modular high temperature gas cooled reactors (modular HTGRs). It contains details of initiating events and overviews of the safety aspects of events that could lead to fuel failure and potential releases of radioactive material. Licensing type safety analyses, aimed at demonstration of sufficient safety margins, are also addressed, and suggestions for analysing various events related to the inherent safety features of modular HTGRs are given.**

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