Safety approach and accident analysis of high temperature gas cooled reactors

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Los reactores de alta temperatura refrigerados por gas (HTGRs) tienen un enorme potencial para participar en un sistema energético sostenible. Sus características intrínsecas de seguridad suponen un cambio radical respecto a los actuales reactores de agua ligera (LWRs). Como consecuencia, la implementación del concepto de defensa en profundidad ofrece peculiaridades, entre las que destaca la mayor importancia adquirida por la prevención frente a la mitigación. La proximidad de construcción del primer reactor PBMR en Sudáfrica, ha puesto de manifiesto la necesidad de desarrollar un nuevo marco de licenciamiento en el que los criterios de aceptación se definen en función de la dosis a público y a personal profesionalmente expuesto.

Este artículo revisa las características de seguridad más distintivas de los reactores HTGR y analiza su efecto en la aproximación de defensa en profundidad. A continuación, presenta el marco general del análisis de accidentes que ha de demostrar el cumplimiento de los límites de dosis y riesgo definidos por la regulación. Finalmente, el análisis concreto del reactor PBMR y de su estado de licenciamiento permite ilustrar la realidad práctica de algunos de los conceptos introducidos anteriormente.

High Temperatura Gas-cooled Reactors (HTGRs) do have a huge potential to contribute in a sustainable energy mix. Its inherent safety features entail a substantial innovation with respect to Light Water Reactors (LWRs). As a consequence, implementation of the defense-in-depth philosophy is not straightforward and requires some adaptation. One generic but main change is the relevance shift towards prevention over mitigation. Construction of the first PBMR in South Africa in the near future has highlighted the need to develop a new licensing framework where acceptance criteria are based on public and worker dose.

This paper reviews the main inherent safety features of HTGRs and discusses how they affect the implementation of the defense-in-depth approach. Additionally, the overall framework of accident analysis is presented and some specifics concerning the advances in PBMR licensing are highlighted.

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INTRODUCTION

This century power engineering is facing one of the greatest challenges ever faced by humankind: the achievement of a sustainable system of energy supply. Most scientists and engineers agree that such a system should consider all the present and future energy sources with a proven sustainable performance. The nuclear sector, supported by its excellent performance in its more than 12600 reactor-years of experience, is getting ready to respond to this requirement with innovative, more economically competitive, and even safer power plant designs. As a whole, these new designs have been termed Generation IV (US-DOE, 2002).

Presently, nuclear fission is dedicated almost exclusively to electricity generation, which, however accounts for only 16% of the energy consumed worldwide. However, nearly 80% of the remaining energy was obtained by burning fossil fuels (IEA, 2006). Therefore, nuclear energy contribution to overcome issues like depletion and supply shortages of fossil fuels and global warming would be vigorously reinforced if a wider energy market was addressed. Industrial heat consumption is a good candidate to accomplish such a diversity of energy products. However, most of the industrial process heat applications require much higher temperatures than the operating temperatures of present Light Water Reactors (LWR). Besides, the amount of energy required is never more than a few hundred MW, while the present systems become competitive only for a thermal production of several thousand MW.

High Temperature Gas Reactors (HTGRs) can produce heat at high or very high temperature (HTR and VHTR, respectively) using a few hundred MW reactors. In other words, HTGRs has the potential to address a wide range of energy products: process heat, electricity and combined heat and power generation (cogeneration). The high temperature of the heat source, allows considering applications like the CO_2-free hydrogen production or oil refining, which require high quality energy, and others like desalination or district heating, which demand much lower temperature heat.

Such a potentiality of HTGRs has reawakened the interest worldwide of investigating specific aspects of these systems that still need further development. A good deal of such a research is being conducted in Europe through different ventures, like the INNOHTR project, the HTR Technology Network and, more recently, the RAPHAEL project (Hittner et al., 2007). Being HTGR inherent safety one of the most important assets of this technology, specific actions have been launched to elaborate a safety approach acceptable to Safety Authorities and to have qualified computation tools to carry out safety analyses (Pirson et al., 2006).

This paper reviews the main inherent safety features of HTGRs and discusses how they affect the implementation of the defense-in-depth approach. Additionally, the overall framework of accident analysis is presented and some specifics concerning the advances in PBMR licensing are highlighted.

HTGR TECHNOLOGY BACKGROUND

The high-temperature gas-cooled reactor concept evolved from early air-cooled and CO_2-cooled reactors. Since the first use of helium as coolant in a 5 MW experimental reactor project (Daniels, 1956), several prototypes were constructed and operated worldwide. Table I compiles the main characteristics of those reactors. Even though presently shutdown, they may be regarded as predecessors of present and upcoming HTGR technology. The main value of the DRAGON reactor was to provide information on fuel, material and component behavior under high-purity helium conditions. The AVR reactor was the first exponent of the pebble-bed type HTGR category and it was mainly used to perform tests related to HTGR performance and safety. The Peach Bottom Unit 1 was the first HTGR demonstration of a prismatic block type reactor and it brought up the need of modifications in the fuel particle design and an additional buffer coating was introduced. These small prototypes eventually resulted in a larger unit of each type, respectively: the THTR-300 (Thorium High Temperature Reactor) and the Fort St. Vrain one.

Tables II and III summarize some of the main characteristics of the prismatic-block type (GT-MHR, ANTARES, HTTR and GTHTR300) and the pebble-bed type reactors (PBMR, HTR-10, HTR-PM), respectively. They include both experimental reactors under operation (HTTR and HTR-10) and prototypes foreseen to be built in the near future. All of them share some features, like being cooled by helium (He) and moderated by graphite or the coated particle fuel design, known as TRISO (Triple ISOtropic layers). It is foreseen

	Dragon	Peach bottom	AVR	Fort St. Vrain	THTR
Location	UK	USA	Germany	USA	Germany
Operation years	1965-1975	1967-1974	1967-1988	1979-1989	1985-1989
Power (MWt/MWe)	20/-	115/40	46/15	842/330	750/300
Power Density (MW/m^3)	14	8.3	2.3	6.3	6
T_{He} (in/out) (ºC)	350/750	377/750	270/950	400/775	270/750
Pressure (bar)	20	22.5	11	48	40
	Fuel features				
Element form	Cylindrical	Cylindrical	Spherical	Hexagonal	Spherical
Coating	TRISO[1]	BISO[2]	BISO	TRISO	BISO
Kernel material	Carbide	Carbide	Oxide	Carbide	Oxide
Enrichment	LEU[3]/HEU[4]	HEU	HEU	HEU	HEU

Table I. Predecessors of HTGR systems (LaBar et al., 2004).
1 TRISO Three materials involved in the coating (low-density PyC, high-density PyC and SiC).
2 BISO Two materials involved in the coating (low-density PyC and high-density PyC).
3 LEU Low-Enriched Uranium (<20% U-235).
4 HEU High-Enriched Uranium (>20% U-235).

that, except the experimental reactors, the rest of systems attain thermal efficiencies notably higher than those of LWRs, essentially due to high He temperatures at the reactor exit and to power conversion units based on Brayton cycles (the Chinese HTR-PM being an exception).

As noted, the most outstanding difference between both reactor types is the fuel arrangement in the core. In the prismatic family, the TRISO particles are embedded in a cylindrical carbonaceous matrix (fuel compacts) which dimensions may be design-dependent; those compacts are then put into a hexagonal graphite block. In the pebble type, the TRISO particles are mixed in a graphite matrix and then pressed and sintered into a sphere of about 6 cm; these pebbles gradually move downward from top to bottom, where they exit and return to the top pushed up by a pneumatic system (on average, pebbles are supposed to make 6 cycles before being considered spent fuel).

SAFETY-RELEVANT DESIGN FEATURES

HTGR designs do not require active safety systems or prompt operator actions to prevent significant fuel failure and fission product release. Plants are designed in such a way that their inherent features provide adequate

	GT-MHR (LaBar, 2004)	ANTARES (Gauthier, 2006)	HTTR (Shiozawa, 2004)	GTHTR300 (Kunitomi, 2004)
Location	Russia	France	Japan	Japan
Plant design life (years)	60	60	20	60
PCU[1]	Direct Brayton cycle	Combined (Brayton-Rankine) Cycle	-	No-intercooled direct Brayton cycle
Power (MWt)	600	600	30	600
Power density (MW/m^3)	6.5	-	2.5	5.8
Net plant efficiency	48%	>45% (electricity) (47-89%, cogen)	-	45.6%
Core ($d_{in}/d_{out}/h$) (m)	Annular (2.6/4.8/8)	Annular	Hexagonal (2.3/2.9)	Annular (3.6/5.5/8)
T_{He} (in/out) (ºC)	490/850	400/1000	395/850	587/850
Pressure (bar)	70	50	40	70
	Fuel features			
Number of fuel columns	102	102	30	90
Fuel hexagonal blocks per fuel column	10	10	5	8
Chemical compound	UCO	UCO/UO2	UO2	UO2
Kernel dimension	650 - 850 μm	500±40 μm	600 μm	550 μm
Enrichment	15.5% (avg)	15 – 19.9%	6%	14%

Table II. HTGR systems with prismatic-block fuel.
1 PCU Power Conversion Unit.

	PBMR (Venter, 2007)	HTR-10 (Seker, 2003)	HTR-PM (Zhang, 2006)
Location	Republic of South Africa	China	China
Plant design life (years)	40	20	60
PCU	Direct Brayton cycle	-	Indirect Rankine cycle
Power (MWt)	400	10	458
Power density (MW/m^3)	3.25	2.0	4.75
Net plant efficiency	>41%	-	40%
Core ($d_{in}/d_{out}/h$) (m)	Annular (2/3.7/10)	Cylindrical (1.8/-/1.97)	Annular (2.2/4.0/1.1)
T_{He} (in/out) (ºC)	500/900	250/700 (300/900)	250/750
Pressure (bar)	90	30	70
	Fuel features		
Number of pebbles	452000	27000	520000
Number of TRISO particles per pebble	15000	8000	-
Chemical compound	UO2	UO2	UO2
Enrichment	9.6%	17 %	9.08 %

Table III. HTGR systems of a pebble bed type.

protection despite operational errors or equipment failure.

Coated fuel particles (CFP)

Ceramic nature of HTGR fuel enables it to withstand very high temperatures. TRISO particles are small, typically ~1 mm diameter. The fuel kernel is coated with four layers of three isotropic materials: a graphite-made porous buffer to accommodate fission gases released without over-pressurizing the particle; and then an inner and an outer layer of dense Pyrolytic Carbon (PyC) in between which there is a ceramic layer of Silicon Carbide (SiC). The last three layers form a miniature pressure vessel that prevents fission products from leaving the particle up to elevated temperatures. Variations in CFP design are primarily in fuel type (typically LEU-oxide or -oxycarbide or Pu-oxide), kernel size, buffer and coating thickness and microstructure (some advanced fuels could use zirconium carbide (ZrC) instead of SiC).

The CFP barriers form the primary line of defense against fission-product release. In normal operation a particular concern is the diffusion of Ag-110m through the intact SiC layer at high operating temperatures, as well as the capability of some fission products (e.g. cesium) to diffuse through intact PyC and graphite matrices. In accident conditions, CFP time/temperature history during the event dominates the fission product release rate. Diffusive release of several fission product species appears to begin at about 1600°C (Verfondern, 1997). Release rates increase markedly for time-dependent exposures in the 1700–2000°C range, and SiC degradation by chemical decomposition begins at approximately 2100°C. Hence 1600°C is typically chosen as a conservative limit on peak fuel temperature under accident conditions. Figure 1 shows this close relation between CFP failure and temperature (Methnani, 2006).

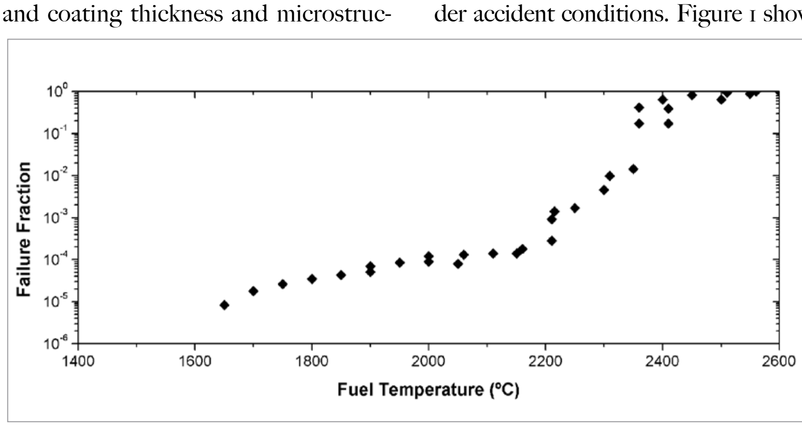

Figure 1. Fraction of CFP failed as a function of temperature.

Cooling by helium

Compared to other gases, helium has several characteristics that make it a suitable coolant at high temperatures and pressures (i.e., higher specific heat and conductivity). However, the specific He features of safety importance are:

- Non-condensable gas behavior.
- Chemical inertia (i.e., fuel and other components will not undergo a He chemical attack).
- No significant reactivity effects.

The non-condensable behavior has two major implications: on one side, no large local temperature increase should be expected during anticipated operational occurrences (phenomena like departure from nucleate boiling cannot happen); on the other, effectiveness of a LWR conventional containment is substantially reduced. In the case of a primary system depressurization event, He retained would eventually behave as a carrier for radionuclides being released during a long term heat-up. In-containment high pressures would eventually raise leaking rates over nominal ones (typically ≤1%/day for existing reactors), this way setting up a driving force for primary circuit helium to enter containment. Thus, in many important scenarios a filtered vented confinement would result in a lower offsite dose than a conventional containment (Williams, 1994).

Passive decay heat removal

In the case of failure or unavailability of all active core cooling mechanisms, decay heat is removed passively from the core to the reactor cavity. Conduction, radiation and, if coolant is present, convection, transfer heat from the core to the non-insulated reactor vessel. Radiation and convection take over heat transmission to reactor cavity. This is feasible because of HTGR limitation of rated thermal power to 6 - 20% of typical power levels for present LWRs.

Nonetheless, a Reactor Cavity Cooling System (RCCS) is necessary to prevent overheating of the reactor cavity concrete during normal operation and to remove core decay heat under accident conditions. The RCCS may not be necessary to prevent overheating of the fuel during accident conditions, as its unavailability would only cause a slight increase in peak fuel temperature. However, it may be necessary to prevent long term overheating of the reactor vessel and possible damage to or failure of reactor cavity structural elements and reactor supports.

Large thermal inertia

The combination of a high heat capacity and a low power density in an HTGR core results in large thermal inertia. This feature together with the high effective thermal conductivity cause a slow heat-up of HTGR cores to events like transients, loss-of-forced convection or depressurizations. As a result, peak values of fuel temperatures are reached days after initiating event and, additionally, they are attained as the magnitude of the decay heat has been considerably reduced. Note that this delay extends the grace period for operational corrective measures to be taken.

Consequences of such a core heat-up delay may be illustrated through depressurization accidents phenomena. In these scenarios the small fission product release from the fuel would occur long after the depressurization was over. At this time, there would be no driving force to transport the fission products. In fact, once the maximum temperature is reached and the system begins to cool, the net flow would be inward.

Temperature margins and Temperatura-reactivity coefficient

A negative temperature-reactivity coefficient is attained over the full temperature range of concern. This along with the large margin between fuel operation and fuel damage temperatures and relatively low excess of reactivity, ensure that any foreseeable reactivity insertion during startup and power operation is accommodated without damage to the fuel.

IMPLEMENTATION OF THE DEFENSE-IN-DEPTH APPROACH

As may be concluded from the preceding section, HTGRs safety relies strongly on inherent features, with the confinement of radionuclides being accomplished with minimal or no reliance on active systems or operator actions. This results in specific peculiarities when implementing for HTGRs the defense-in-depth concept (IAEA, 1996): "a hierarchical deployment of different levels of equipment and procedures (Levels of Defense, LoD) to maintain the effectiveness of physical barriers placed between radioactive materials and workers, the public or the environment in normal operation, anticipated operational occurrences and, for some barriers, in accidents at the plant". This section illustrates how the above safety-relevant characteristics of HTGR influence the Defense-in-depth implementation.

The "barriers" that can be identified in an HTGR system are: the kernel (i.e. the fuel material), the three particle coating layers, the matrix (i.e. the graphitic material around the particles), the fuel element (i.e. pebble/fuel assembly block), the primary circuit, the plant confinement/containment and the filtering system. Particularly relevant in the radioactive containment function are the coating layers of the CFP. In order to get the right perspective, though, two key facts should be noted: the kernel coatings are a system of consecutive barriers operating in parallel among the billions of particles comprising a typical HTGR core, and each particle contains an insignificant amount of fission products.

Measures relative to defense in depth are ranked in five levels of defense (Figure 2). The first four levels are oriented towards the protection of barriers and mitigation of releases; the last level is related to off-site emergency measures to protect the public in the event of a significant release. The priority given in HTGR designs to accident prevention, by significant plant simplification and very robust LoDs, makes Level 1 and Level 3 acquire a strong emphasis.

Level 1: Prevention of abnormal operation and failures

The objective for Level 1 is the prevention of deviations from normal operation and, as a consequence, the prevention of failures so that safety systems would operate reliably if called upon at higher levels of defense. The essential means are the provision of a conservative design and high quality in construction and operation.

As the HTGR safety philosophy is based on control of releases primarily by retention of radionuclides within the CFP, it becomes of utmost importance to achieve high quality in manufacturing ceramic coated-particle fuel. Additionally, other HTGR safety-related features also affect this level of defense.

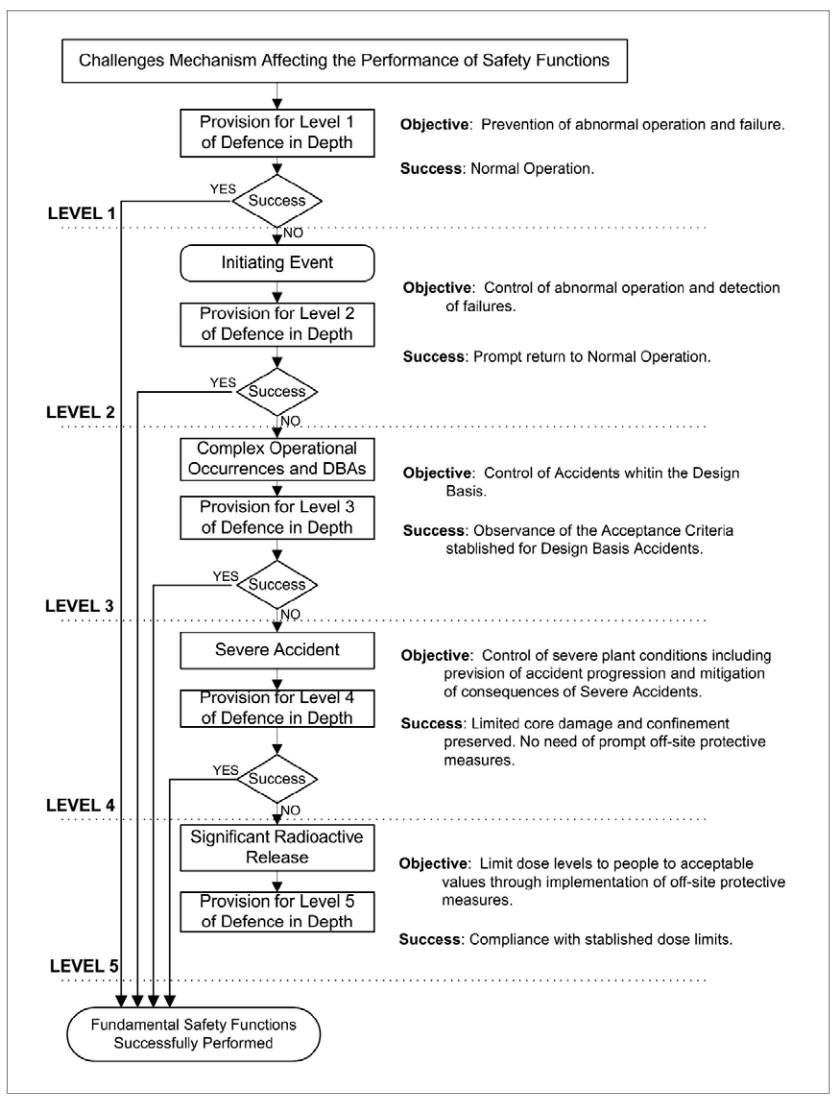

Figure 2. Logics of the Defense-in-Depth approach.

For example, HTGRs capability of passively removing decay heat reduces significance of pressure boundary integrity and the helium inertia eliminates the material demands stemming for coolant chemical attack. In addition, HTGR designs robustness (i.e., provision of adequate time for operators and the system to respond to normal events) is a consequence of single phase coolant, low power density, high core heat capacity and large temperature margins between normal operation and fuel failure temperatures.

Level 2: Control of abnormal operation and detection of failures

The objective for Level 2 is the control of abnormal operation and detection of failures. The essential means are control, limiting and protection systems and other surveillance features. The successful performance of Level 2 provisions will bring the plant back to normal operating conditions as soon as possible.

Level 2 incorporates inherent plant features, such as core stability and thermal inertia. The Reactivity Control and Shutdown System (RCSS) controls the reactor during normal operation conditions and is provided with an independent safety system to shut down the reactor if normal operating limits are exceeded. This system is supported by the negative temperature reactivity coefficient that limits the maximum temperature of the fuel.

Level 3: Control of design basis accidents

The objective for Level 3 is the control of accidents within the design basis. The essential means are inherent and engineered safety features and accident procedures. The measures taken at this level are aimed particularly at preventing fuel damage.

The fundamental safety principle of HTGR designs is that there exists no physical process that could cause a radiation induced hazard outside the site boundary. This is achieved by a heat loss from the reactor vessel that exceeds the decay heat production in the post accident condition. As a consequence, the peak temperature reached during the transient is lower than the fuel and structures degradation temperatures. The main provisions for Level 3 are:

- Passive decay heat removal during accidents (heat conduction, radiation, natural convection) to simple surface coolers.
- Strong negative reactivity temperature coefficient and large fuel power and temperature margins provide a reliable inherent defense against positive reactivity insertion.
- No need for an intact primary coolant pressure boundary (pipe break) to prevent significant core degradation.

Additionally, RCCS and RCSS are independent and diverse systems, seismically designed that reinforce the above passive means.

Level 4: Control of severe accident conditions

The objective for Level 4 is the control of severe plant conditions, including prevention of accident progression and mitigation of the consequences (i.e., keeping radiation levels under the allowable radiological limits). Given that fuel melting is eliminated by design, severe plant conditions do not necessarily involve large releases from the fuel but they could challenge the structural integrity of the core and thus the ability to analyze the course of an event.

The main focus will be again to protect as much as possible the capability of CFP to retain fission products. The essential means are complementary measures and accident management. The large thermal inertia of the plant and characteristics of the fuel and reactor internal structures will provide considerable time to deal with these conditions. If necessary, additional measures and procedures may be provided. Ancillary and support systems, if employed, would be designed, manufactured, constructed and maintained consistent with the required reliability.

Level 5: Mitigation of radiological consequences

The objective for Level 5 is the mitigation of radiological consequences of significant releases of radioactive materials. The essential means are the off-site emergency response and the HTGR goal is to design a plant for which sheltering and evacuation measures are not necessary.

ACCIDENT ANALYSIS

The principal figure of merit for the accident analysis is the calculated onsite and offsite dose compared to dose acceptance limits. Therefore, the objective of the accident analysis is to demonstrate the compliance with regulatory dose and risk limits for both the public and the worker during accident conditions.

In order to carry out a thorough accident analysis of HTGRs, events are usually classified by frequency (or probability of occurrence) or by type. The former may be either initiating event frequency or event sequence frequency. The latter involves sorting each event into a specific group that is defined by a particular characteristic of the event (i.e., event initiator, phenomena involved, challenge to safety functions, etc.).

Event classification: frequency and type

For the purpose of accident analysis, all events are grouped into frequency based categories as follows:

- **Anticipated Operational Occurrence (AOO).** An AOO is a frequent event that may occur one or more times during the life of the plant. AOOs are typically associated with events having a mean frequency of occurrence higher than 10^{-2}/plant-year or higher.
- **Design Basis Accident (DBA).** A DBA is an infrequent event that is not expected to occur within the lifetime of the plant, but it may occur during the collective lifetimes of a large number of plants. However, the plant is specifically designed to mitigate this event. They are typically associated with events having a mean frequency between 10^{-2}/plant-year and 10^{-4}/plant-year.
- **Beyond Design Basis Accident (BDBA).** A BDBA is a rare event that is not expected to occur within the collective lifetimes of a very large number of similar plants. However, the plant is specifically designed to mitigate the event by taking credit for available safety related equipment and functional non-safety related equipment as well as creditable operator actions. BDBAs are typically associated with events having a mean frequency between 10^{-4}/plant-year and 10^{-6}/plant-year.

Very rare events which frequency falls below the cutoff for BDBAs generally need not be included within the accident analysis. However, sometimes events with frequencies around or even lower than 10^{-7}/plant-year, may be included if uncertainties affecting probability of occurrence are significant.

Note that the frequencies above are given by plant, regardless the number of independent reactor modules. Additionally, the frequency must include multiple failures due to common cause events and an account for all operating and shutdown modes.

Even frequency is closely related to dose acceptance criteria (to be defined by national regulatory authority). AOOs must have either very low or no dose consequences to onsite personnel or to offsite individuals or the surrounding population. Infrequent events (i.e., DBAs) are permitted to have limited consequences, while rare events (i.e., BDBAs) may be

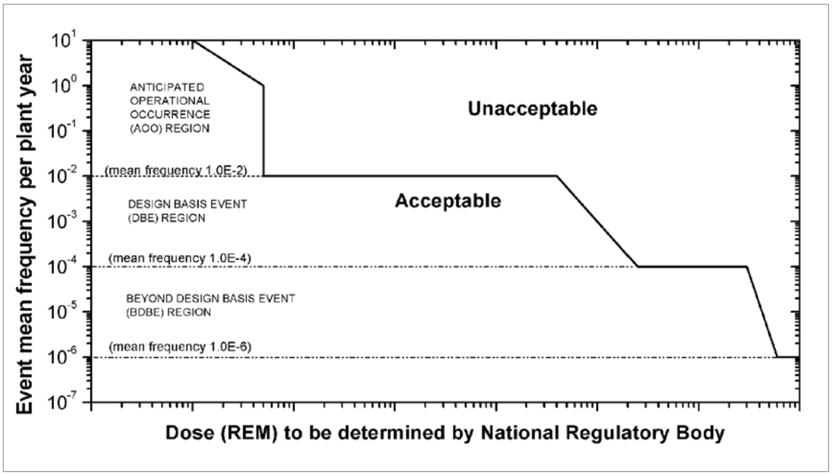

Figure 3. Relationship between frequency and dose.

allowed to have somewhat higher consequences. Figure 3 illustrates qualitatively this correlation (Makihara, 2006) at the exclusion area boundary.

Events may also be grouped by type based on the safety functions challenged (i.e., heat removal, reactivity control, radioactivity confinement and chemical attack control) and on dominant phenomena in the course of the event:

- **Helium Pressure Boundary (HPB) Breaks** (challenge to confinement of radioactivity). They result in a release of radioactivity from the primary system. These include all leaks greater than the normal operational leakage rate. All the events with a simultaneous challenge to reactivity control and/or heat removal are included in this grouping. Examples of this group are (whenever they result in a loss of radioactivity confinement): breaks with a loss of forced core cooling, air ingress events, water ingress events, etc.
- **Reactivity Transients** (challenge to reactivity control). All those events that result in a reactivity transient without failure of the HPB are included in this category. Event sequences with a simultaneous challenge to reactivity and heat removal are included in this grouping. For example a withdrawal of a set of control rods with a loss of forced cooling.
- **Heat Removal Transients** (challenge to heat removal). Event sequences resulting in a heat removal transient with no failure of HPB are considered. Examples are a turbine trip with decay heat removed by an active system (AOO) and, also, a turbine trip with passive decay heat removal (DBA). The short term pressure transients are treated as a subset of this group.
- **Other Activity Releases**. This category includes all those events associated with sources of activity other than the core and primary system. For example the fuel handling and storage system (for pebble bed reactors), the spent fuel storage tanks, the helium inventory control system and the waste handling system.

Consequence acceptance criteria

Consequence acceptance criteria can be split in two categories: basic and specific criteria. **The basic criteria** are related to the top level dose acceptance criteria (to be defined by the national regulatory authority). The safety analysis must show that the dose to the public and to onsite plant personnel is less than the regulatory dose limits for the event frequency category. For example (Makihara et al., 2006): AOO, 5 mrem offsite; DBA, 25 rem, BDBA, 300 rem. Additionally, it must be shown that the overall risk (all events considered) to the public does not exceed the specific risk guidelines established by the regulatory authority. The **specific criteria,** determined by the designer and approved by regulators, ensure defense-in-depth; so to say, they guarantee that structures, systems and components modeled in the analysis and related to accident mitigation, are capable of performing their safety functions. Examples of limiting variables are: fuel, maximum temperature; core, maximum dynamic load; reactor pressure boundary, maximum pressure and temperature, etc.

Methodology

Evaluation of doses is the main output from a safety analysis. This requires a series coupling of the analytical capabilities to assess all the safety related aspects of the reactor; essentially: neutronic, thermal fluid dynamic response, fuel performance, release and transport of fission products to confinement (containment) and, from there, to environment, their atmospheric dispersion and, eventually, onsite and offsite dose estimates.

Best estimate and/or conservative analysis may be applied, depending on regulatory requirements and event category. The use of best estimate computer codes is encouraged whenever it is complemented with sensitivity analyses or, preferably, with a thorough uncertainty analysis that ensures sufficient safety margins. Conservative analyses should be carefully set out since the existence of multiple acceptance criteria could make an initially pessimistic analysis with respect to one criterion not to be so with respect to others. In these cases, a separate analysis needs to be done for each acceptance criterion.

THE PBMR SAFETY APPROACH

The Pebble Bed Modular Reactor (PBMR) is scheduled to be constructed in 2009. Designed to be passively safe, environmentally acceptable, highly efficient and commercially competitive, the PBMR will be available as single-module or multi-module arrangements. A PBMR module consists of a graphite-moderated, helium-cooled reactor in which the gas is heated by the nuclear fission reaction and establishes a direct recuperative Brayton cycle. Even though, many of PBMR features have been encapsulated in Table III, those closely related to safety aspects are reviewed next.

Overview of safety features

According to the overall HTGR approach, the PBMR safety strategy relies on three fundamental principles:

- Utilization of proven technologies.
- Accomplishment of passive safety by inherent design features.
- Implementation of active systems for defense-in-depth.

The PBMR reactor is based on the HTGR technology that was originally developed in Germany, which main exponents were the AVR and THTR reactors. The pebbles, the bed core configuration and the steel reactor vessel were based on the AVR reactor (VDI, 1990), whereas some specific components and the reactor design rules were adapted from the THTR reactor. Further than that, the PBMR plans to use materials, components and processes that have a proven nuclear or industrial record to the maximum extent. All it without giving up conductance of research and development programs as required, particularly to extend its attributes to those of VHTRs.

PBMR inherent design features will achieve passive safety with no need of electric power, successful operation of active systems or operator actions. Three major design elements sustain

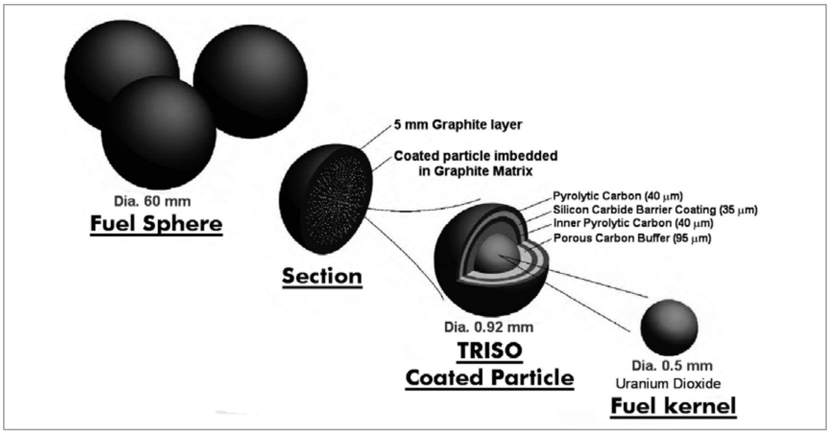

Figure 4. Schematic of a PBMR pebble.

this philosophy: the fuel, the materials and the geometry. The capability of TRISO particles to retain fission products up to very high temperatures and its chemical compatibility with helium, have been extensively proven (IAEA, 1997). Figure 4 shows a schematic of the TRISO particles arrangement in the pebble. The properties of graphite (i.e., high heat capacity, chemical compatibility with fuel and coolant, well characterized thermal conductivity, etc.) and helium (i.e., single phase, chemically and neutronically inert, low stored thermal energy, etc.), make these materials highly suitable in the system. And, finally, the tall and slender core results in a low power density and a large surface area that favors passive cooling; additionally, it also provides for damping Xenon oscillations.

Engineered active design features are aimed to meet reliability, availability and investment protection requirements and to provide additional prevention and mitigation as required in the defense-in-depth approach. Examples are the active elements of reactivity control systems, the Heating Ventilation and Air Conditioning (HVAC) system, the helium purification and inventory control systems, etc.

In summary, the PBMR design features (mostly generic to other HTGRs) make a big difference with respect to current Light Water Reactors (LWRs) concerning safety:

- Fuel, coolant and moderator are chemically compatible under all conditions.
- Fuel has very large temperature margins in normal and accident conditions.
- Core integrity does not depend on the presence of coolant.
- Reactor response times are very long (days).
- Inherent mechanisms for runaway power excursions are nonexistent.
- Three concentric and independent barriers act as radionuclide containment
- A LWR-type containment is neither advantageous nor necessarily conservative.

Licensing framework

The existing South African licensing process had to be adapted to take into account the developmental nature of the PBMR project. It entails the development of a set of PBMR specific regulatory documents on many regards, particularly on the assessment of the Safety Case (i.e., a comprehensive safety justification). These binding licensing documents comprise, besides those principles addressing good engineering practices, ALARA (As Low As Reasonably Achievable) and defense-in-depth, specific radiation dose limits (Table IV) (Bester et al., 2006).

A multi-staged process has been adopted. After a satisfactory regulatory review of the safety case by the NNR (National Nuclear Regulator), an initial Nuclear Installation Licence (NIL) will be granted to the applicant for the first stage of the process. A variation to this NIL will be requested by the applicant and issued by the NNR following its satisfactory regulatory review, at each subsequent licensing stages. The foreseen stages are:

- Stage 1& 2: Site preparation, construction and manufacturing phase.
- Stage 3: Commissioning and Start-up (nuclear fuel on site).
- Stage 4: Plant operation.
- Stage 5: Eventual decommissioning.

It is during the first stage when the NNR assess the proposed design to confirm that it meets the licensing requirements. Then, a comprehensive safety report must accompany and support the application for the initial NIL and for each subsequent application for the NIL variation. It is noteworthy that a broad international consensus has not been developed on general design criteria and rules for the PBMR (unlike LWRs). In other words, although HTGRs have been licenced and operated elsewhere, no international "off the shelf" package is available for defining the design basis and the safety case for the PBMR.

Therefore, in view of the absence of well researched and documented design criteria and rules, a structured process has been implemented to develop the safety case and demonstrate the PBMR compliance of the licensing requirements. The main components for the development and review of the PBMR safety case are:

- The Safety Case Philosophy (SCP); It provides the intellectual and philosophical arguments of how PBMR safety will be demonstrated to meet the safety requirements set by the NNR.
- The Safety Analysis Report (SAR); It is to provide a detailed justification of how the safety arguments will be demonstrated.
- The General Operating Rules (GOR); along with other documents, it refers collectively to safety related practices or programs that are applicable during the operational phase of the plant.

At present, the PBMR SCP and SAR Rev1a have been submitted to and reviewed by the NNR (externally supported by a group of international consultants). As a result of discussions held with the applicant (ESKOM), the initial SCP has been considered credible and, as a consequence, it has been accepted. During the review process 25 key Licensing Safety Issues (KLIs) have been identified, most of which need to be addressed in SAR Rev2 in support of stages 1 & 2. The compilation of the SAR Rev2 is currently underway by the PBMR Pty (Ltd) and ESKOM. To do so PBMR has estab-

Event	Category	Credit (to)	Release
10 mm break	AOO	Forced cooling HVAC filtration	Initial
10-230 mm break at bottom of reactor or PCU	DBA	Passive heat removal	Initial + Delayed
10 mm break + loss of forced cooling	DBA		
10-30 mm break at reactor top	DBA		
30-230 mm break at reactor top	BDBA	Passive heat removal Reactor unit retention Confinement retention	
230-1950 mm break	BDBA		
Single control rod ejection	BDBA		
SSE +pipe break	BDBA		

Table IV. Dose and risk limits.
***The ALARA doses are required to be demonstrated by Best Estimate analysis.**

Event	Category	Credit (to)	Release
10 mm break	AOO	Forced cooling HVAC filtration	Initial
10-230 mm break at bottom of reactor or PCU	DBA	Passive heat removal	Initial + Delayed
10 mm break + loss of forced cooling	DBA		
10-30 mm break at reactor top	DBA		
30-230 mm break at reactor top	BDBA	Passive heat removal Reactor unit retention Confinement retention	
230-1950 mm break	BDBA		
Single control rod ejection	BDBA		
SSE +pipe break	BDBA		

Table V. Breaks requiring public and worker dose estimates (Bredin, 2006).

lished cooperation agreements with international organizations, an example of which is the PBMR-CIEMAT collaboration for the PBMR confinement analysis.

Accident analysis

PBMR initiating events are categorized according to the classifications presented above. So, they are grouped by frequency in AOOs, DBAs and BDBAs, and each of these categories are split into three sets: breaks, reactivity transients, heat removal transients and other radioactive releases from non-Main Power Systems.

Break type initiating events are the only ones that have the potential to result in accidental public and worker dose. The success criteria are compliance with the regulatory dose and risk limits shown in Table IV. The potential releases are divided in two phases: initial release and delayed release. The initial release includes the activity circulating with the coolant or deposited within the coolant circuit. The delayed release is the activity released from the fuel during breaks where there is a loss of forced cooling. The break type BDBAs considered in the safety analysis are: a large pipe break, a single control rod ejection, medium break at the top of the reactor and seismic event with pipe break, all with a loss of forced cooling and an unfiltered initial and delayed release.

The success criteria for reactivity transients and heat removal transients are that fuel, reactor and HPB limits are not exceeded. These limits are different for AOOs and DBAs, the former being more demanding since operation is assumed after the event. Anyway, they are set in such a way that ensure no large increase in the circulating activity, no change in reactor geometry and no rupture of the helium pressure boundary.

Table V shows a set of sequences requiring estimate of public and worker doses. The analysis is broken down into 5 major steps: pre-break phase (the potential source term available for the initial release is calculated); initial release (how much of the ex-core source term is release from the HPB); delayed release (only applicable to scenarios that include a loss of forced cooling and subsequent core heat-up); worker dose analysis (doses estimates from initial and delayed release, if applicable); and, public dose calculations.

The dual nature of the NNR safety criteria (i.e., dose and risk) implies that the safety analysis for demonstration of compliance of the Safety Case with licensing criteria for the PBMR have to comprise both deterministic and probabilistic analyses. PBMR has already defined a specific methodology to assess external and airborne radionuclide source terms and dose rates by linking a set of computational codes. Table VI shows the calculation platform such as it was by mid 2006 (Ramlakan et al., 2006). At the time, phenomena not explicitly included in the table, like production of H3 and C14 or dust-fission product interaction were calculated through analytical equations. Nevertheless, this should be understood as a living statement. Very recently, PBMR stated their intention to carry out the confinement modeling with the ASTEC-CPA code (Naicker et al., 2007). Hence other changes with respect to Table VI should not be ruled out.

SIMULATION AREA	CODE
Noble gases and halogens release into the HPB	NOBLEG
Metallic fission and activation products	GETTER
Transport and deposition of fission product and carbon dust	RADAX
Nuclide transport in the plant building	AMBER / CFD-PARDISEKO
Dispersion an impact of atmospheric releases	PC-COSYMA & ADMS

Table VI. Calculation platform at PBMR.

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