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# Key Issues Affecting Advanced High Temperature Gas Reactors

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## FOREWORD

From time to time, INSAG will make available certain reports of high quality and interest that have been developed by staff and/or contractors for INSAG discussion. These reports in some instances may eventually provide the foundation for a formal report in the INSAG series. In the meantime, however, we conclude that our mission of seeking to advance international nuclear safety will be furthered if we make some of these background documents more widely available. This report on issues relating to advanced High Temperature Gas Reactors (HTGRs) is an example of a background document that we conclude should be made available to the interested public.

This study was prepared in order to provide a framework for INSAG's discussion of the issues that concern the safety and licensing of an HTGR. The approach taken in the report is to array the various issues and challenges that must be confronted in a safety review of an advanced HTGR. The discussion is presented at a relatively high level to facilitate discussion, categorization, and prioritization. It is not intended as a comprehensive analysis or examination of the important decisions and policies that must be addressed, but rather as a starting point for debate and discussion. As will be apparent, the report is based largely on work undertaken by the Office of Research of the US Nuclear Regulatory Commission.

Although most of the world's nuclear operating experience has been achieved with water-cooled reactors, there is significant experience with gas-cooled reactors in the United Kingdom, Germany, China, Japan, and the United States. Specifically, the United Kingdom has operated carbon-dioxide-cooled advanced gas reactors (AGRs), which are different than HTGRs, but which provide experience with graphite moderation. The United States has experience with the Fort St. Vrain HTGR, which was helium cooled and graphite moderated and used a coated particle fuel design. Germany has had about 20 years experience with a small (45 MWt) and about 4 years experience with a large (750 MWt) HTGR, both of which utilized helium coolant and coated fuel particle designs. Japan has operated a research HTGR (30 MWt) that employs helium as a coolant and coated fuel particles and China is currently operating a small (10 MWt) pebble bed HTGR research reactor. This experience was considered in the evaluation of advantages and challenges associated with HTGRs in the following report.

We hope that this report will be of value to Member States and others interested in the HTGR technology.

Richard A. Meserve

Chairman, INSAG

## 1.0 Purpose and Approach

The purpose of this study is to provide a framework to facilitate discussion of the issues that have and likely will emerge with the safety and licensing of an advanced High Temperature Gas Reactor (HTGR) by an International Atomic Energy Agency (IAEA) member state.

The approach taken was to assemble the issues and challenges that need to be addressed in a safety review of an advanced HTGR at a sufficiently high level to facilitate discussion, categorization, and prioritization. These issues are grouped relative to their potential to affect the ability to license the design. Section 3.0 addresses the considerable advantages of an advanced HTGR design, while section 4.0 list those challenges which are the furthest from resolution, or that provide the highest risk of presenting an issue that could affect the ability to license the design in a member state. Section 5.0 details those issues that require additional work and coordination, but will likely achieve satisfactory resolution in time. Section 6.0 highlights important policy questions that will influence the design, operation, and safety of the new generation of advanced high temperature gas reactors. It is not intended as a comprehensive listing of the important decisions and policies needed, but rather a starting point for debate and discussion.

The U. S. Nuclear Regulatory Commission Office of Research's Advanced Reactor Research Plan was relied on heavily in the development of this study and is cited as the first reference in Section 7.0. Section 8.0 provides a definition for all abbreviations used in the study.

## 2.0 Background

The large majority of the approximately 10,000 reactor-years operating experience accrued by the world's nuclear reactors has been achieved through operation of light and heavy water cooled reactors. As a consequence, the thermal-hydraulic and nucleonic codes, safety analyses, and even the regulations themselves have been to a large extent written for, tested, and refined during the design, construction, licensing and operation of water cooled reactors.

Fortunately, while the majority of the world's operating experience has been achieved with water cooled reactors, a significant amount of work has been done with gas cooled reactors in the United Kingdom, Germany, China, Japan, and the United States. Specifically, the United Kingdom has operated carbon dioxide cooled advanced gas reactors (AGRs), which are different than HTGRs but provide experience with graphite moderation. The United States has experience with the Fort St. Vrain HTGR, which was helium cooled and graphite moderated and used a coated particle fuel design. Germany has had about 20 years experience with a small (45 MWt) and about 4 years experience with a larger (750 MWt) HTGR, both of which utilized helium coolant and coated fuel particle designs. Japan has operated a research HTGR (30 MWt) that employs helium as a coolant and coated fuel particles and China is currently operating a small (10 MWt)

pebble bed HTGR research reactor. None of the HTGRs discussed utilize a Brayton cycle energy conversion system as is contemplated for the PBMR.

The experience gained from the HTGR operation described above, and the applicable water cooled reactor operating experience was considered in the development of the list of advantages and challenges that follow.

## 3.0 Advantages of an Advanced HTGR Design

HTGRs have always been attractive designs from a safety perspective. In particular, the low power density in the core, high thermal capacity of the graphite moderator, and slowly developing severe accidents provide the opportunity to present the world with a substantially safer generation of commercial nuclear reactors. Key advantages of the HTGR design are provided below.

### 3.1 Low Power Density[1]

The power density in a Pebble Bed Modular Reactor (PBMR) is an order of magnitude less than the power density in a typical light water reactor. This lower power density results in a lower decay heat flux that can be managed through radiation and conduction to the thermal mass of the moderator independent of the primary coolant. This allows the designer to employ passive safety systems that do not rely on operator actions or the availability of active systems.

### 3.2 High Thermal Capacity of the Moderator[2]

The graphite moderator and structural components of an advanced HTGR provide a sufficient thermal mass to accommodate post trip decay heat, and to substantially slow the rate of fuel temperature increases for certain accidents. The high surface to volume ratio of the core facilitates graphite cooling at a rate that can make achieving high fuel temperatures from decay heat unlikely. Graphite, if shielded from oxidizing agents, can withstand very high temperatures. The high heat capacity of graphite does not cause a sudden temperature rise in fuel kernels, resulting in a safe reactor shutdown.

### 3.3 Slowly Developing Accidents[3]

The low power density of an advanced HTGR coupled with the large thermal mass of the moderator substantially slows the rate at which off normal and accident conditions develop. For example, the high fuel melt temperature permits passive feedback mechanisms such as a relatively high negative temperature coefficient of reactivity to ultimately shut the reactor down during a loss of coolant accident coincident with a failure to scram.

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[1] Eskom Generation PBMR Safety Case Philosophy, October 2004
[2] South Africa Pebble Bed Modular Reactor Company Pty, Ltd Web page: www.pbmr.com/
[3] Eskom Generation PBMR Safety Case Philosophy, October 2004

The PBMR design does not rely on operator actions for several hours following an anticipated operational occurrence or accident. The negative temperature coefficient and the temperature tolerance of the fuel are such that imbalances in heat generation and heat removal can be restored without reliance on an active engineered safety system.

### 3.4 Fission Product Barrier

The tri-isotropic coated particle fuel (TRISO) design encapsulates each fuel seed in a silica-carbide shell that serves as a primary barrier to contain radioactive nuclides up to very high temperatures (see section 4.1). The large number of 1 mm fuel particles, each with its own silicon carbide fission product barrier and each with a very small inventory of fission products, provides additional assurance against a large release of fission products to the environment since any single failure of the fission product barrier for a 1 mm fuel particle would not result in a meaningful fission product release.

### 3.5 Helium Coolant

The use of helium as a primary coolant medium provides a number of advantages over water. In particular, the coolant will not change phase under all postulated operating conditions thus simplifying heat transfer and accident analyses. Additionally, the single phase primary coolant facilitates operating with primary coolant pressures much lower than those of a water cooled reactor lessening the likelihood and severity of a loss of coolant accident.

Dry helium will result in a more benign environment for the fuel, the moderator, and the structural components relative to corrosion and should lessen the radiological source term associated with activated corrosion products that are seen in light water reactors.

Helium has a small neutron scattering and absorption cross sections, resulting in a negligible moderator temperature coefficient. Doppler temperature feedback is then a dominant mechanism contributing to an overall negative reactivity feedback coefficient, an important characteristic in an “inherently safe” reactor design.

## 4.0 Significant Gaps and Challenges Associated with the Design of an Advanced HTGR.

The three issues developed below, fuel, materials, and security are considered to present significant challenges to the licensability of the design in some IAEA member states. These issues require additional analysis and a coordinated effort between the designers, owners, vendors, and regulators to assure that any gaps in knowledge and experience are effectively bridged.

## 4.1 Fuel[4] [5]

A considerable amount of research and development work has been performed over the past 30 years on tri-isotropic coated particle fuel (TRISO). The design evolved from a single anisotropic carbon layer, to bi-isotropic coated particle fuel (BISO) with a layer of buffered isotropic pyrolytic carbon, to TRISO with a porous buffer layer, an inner pyrolytic carbon layer, a silicon-carbide layer, and an outer pyrolytic carbon layer.

The coated particles can be suspended in a graphite matrix in the form of small spheres, for use in pebble bed reactors, or in cylindrical forms for use in HTGRs. In the spherical form, special challenges arise relative to structural-seismic interactions which can result in high point stresses between spherical fuel pebbles, and between fuel pebbles and the reactor vessel internals.

The manufacturing process for TRISO fuel includes vapor deposition and gel precipitation droplet formation, both of which involve some degree of inherent process randomness which results in a statistical distribution of attributes such as density, anisotropy, thickness, and microstructure. Rigorous quality control over the billions of individual fuel pebbles that will be manufactured will provide a significant challenge. Additionally, since individual fuel pebbles are not currently designed to have unique identifiers, tracking problems back to manufacturing process defects will be very difficult. The statistical distribution of fuel attributes will result in some additional uncertainties in the character of the core that will vary over time and will need to be addressed in the nucleonic, thermal hydraulic, and fuel performance codes.

A number of failure mechanisms have been identified for TRISO fuel such as internal overpressure, tensile failure of the silicon-carbide (SiC) layer, chemical attack, thermal decomposition, and mechanical overstress due to external loading. Failure modes that affect the shape or surface characteristics of a pebble can further complicate the predictions of pebble flow through the core. Currently, there is some consideration of altering the constitution of the SiC layer to reduce susceptibility to chemical attack from fission produced palladium. As problems are encountered, continued evolution of the fuel design and manufacturing processes can be expected.

A substantial amount of research and operating experience with coated particle fuel has been gained from operation of the Arbeitsgemeinschaft Versuchreaktor (AVR) in Germany, the Japan Atomic Energy Agency (JAEA) HTTR in Japan, the High Temperature Reactor 10 (HTR-10) in China, Fort Saint Vrain in the United States, and the U.S. Department of Energy and the European Commission's research on fuel development and technology. This experience has included accident simulation tests involving ramping temperature to accident levels and holding it for an extended period. Additional tests are

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[4] USNRC Office of Research Advanced Reactor Research Plan (March 2002) NRC ADAMS Accession Number M L021570221

[5] Summary of NRC Delegation Visit to Germany on Safety Aspects of HTGR Design Technology August 21, 2001, NRC ADAMS Accession Number ML 012330131

needed that include rapid temperature cycling, post irradiation air and water intrusion, and reactivity excursions to further understand the fuel reliability and performance under transient and accident conditions.

### 4.2 Materials[6] [7]

In spite of the experience to date, the performance of new and familiar materials in high temperature helium applications is likely to provide substantial new challenges. In HTGRs, graphite is used as a moderator, reflector, heat sink, and structural material. Experience to date has shown that under temperature and irradiation conditions, graphite exhibits dimensional instability (both shrinking and swelling), changes in thermal conductivity, loss of strength, oxidation, creep, and fatigue. These effects may have some sensitivity to the purity of the raw material and the manufacturing processes used to manufacture and form the nuclear grade graphite, thus existing operating experience may not be fully applicable to newly manufactured or different physical forms of graphite.

Metals used in HTGR applications may experience aging (solid state transformation, precipitation, and embrittlement), sensitization of austenitic stainless steels, carburization, decarburization, oxidation, creep, fatigue, stress and crevice corrosion cracking from impurities in the helium coolant, and unknown long term effects of operating in a high temperature helium environment.

Concrete is known to degrade with prolonged exposure to temperatures above 260 degrees C (500 degrees F). With operating temperatures well in excess of this, special consideration will need to be given to concrete interfaces during normal and accident conditions. On line monitoring, an active reactor cavity cooling system (RCCS), and periodic inspections may be appropriate over the operating life of the reactor.

Non-destructive examination (NDE) and in-service inspection (ISI) will provide new challenges for advanced HTGRs. With the on-line refueling capability of a PBMR, the amount of shut down time to perform NDE and ISI will be reduced, and with fewer periods of time with the entire core off-loaded, the accessibility of some components will be more restricted. NDE techniques for nuclear grade graphite will need to be developed and improved, as well as remote methods to assess the surface condition and structural integrity of pebbles as they are examined before additional passes through the reactor.

The performance of materials in an HTGR environment is an important input to the PRAs that will be developed and may require a probabilistic fracture mechanics approach to deal with uncertainties.

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[6] USNRC Office of Research Advanced Reactor Research Plan (March 2002) NRC ADAMS Accession Number M L021570221

[7] Summary of NRC Delegation Visit to Germany on Safety Aspects of HTGR Design Technology August 21, 2001, NRC ADAMS Accession Number ML 012330131

### 4.3 Safeguards and Security[8] [9]

Safeguarding a facility such as a 10 module PBMR from diversion of fissile material creates some challenges not found at water cooled reactors due to the small size of the fuel and the nature of the nuclear cycle that involves multiple passes of the fuel through the core. The small size (about 60 mm in diameter) of the spent fuel pebbles make theft easier than with large water cooled reactor fuel elements due to the increased ease of portability and the ability to effectively shield small sources. Once stolen, a pebble would be relatively easy to use as an improvised radiation dispersal or exposure device.

Extracting usable quantities of fissile material from diverted fuel pebbles would be very difficult, considering the low enrichment at the front end of the fuel cycle and the high burn-up at the back end. However, cycling pebbles of depleted uranium through the core could provide a means to slowly build a usable quantity of plutonium 239.

Designing and building a new reactor in a post September 11, 2001 world provides an opportunity to harden the facility in ways that would be very difficult for existing facilities. In order to gain from the cost efficiencies of including the appropriate security requirements in the initial design, a number of fundamental questions need to be addressed. Specifically:

> Is the design basis threat a world-wide constant or is it country/region specific?
> Should the facility be designed to withstand multiple impacts by large aircraft?
> How should large fires and explosions be managed?
> How and where should spent fuel be stored?
> Is a containment necessary for security reasons?
> Should safety systems be bunkered?
> Should the control room be bunkered?
> Should plants be sited underground?

Security requirements for reactor facilities vary by country. A key contributor to these differences is the varying philosophies of the role of a nation's federal government relative to the facility owner/operator. At what point does protecting a facility become the responsibility of a nation's army, as opposed to a local security force? While achieving a consensus relative to the appropriate roles of private versus federal force to protect a facility will be difficult, it may be easier to achieve some consensus on the minimum set of design features that should be employed in a new nuclear facility to harden it against terrorism.

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[8] Trip Report D.A. Powers, HTGR Safety and Research Issues Workshop held October 10-12, 2001 NRC ADAMS Accession Number ML 020450645

[9] Exelon Position Paper Addressing Significant Conclusions in D.A. Powers' Trip Report Covering the HTGR Safety and Research Issues Workshop October 10-12, 2001, NRC ADAMS Accession Number ML 021640672

## 5.0 Issues Affecting the Design of an Advanced HTGR which Require Additional Work but are Likely to be Resolved

### 5.1 Regulatory and Licensing Issues

Design and licensing of new reactors provides an opportunity to revisit safety goals from the perspective of Level 2 and Level 3 Probabilistic Risk Assessments (PRAs) that consider containment performance, source term, and off-site consequences, and refine them based on what can be reasonably achieved as opposed to what was achievable using new PRA insights on old designs. While license conditions and operational limitations are established based on not exceeding safety limits, maintenance and surveillance requirements can be optimized based on risk insights. The PRAs developed for advanced reactors will likely show that passive systems have reduced the probabilities of core damaging events from internal events low enough that overall risk will be dominated by external events (fires, floods, earthquakes, etc) and human performance. The risk analyses done to support safety decisions for advanced reactors should address these areas.

Defense in depth is a concept that has served the nuclear industry well for the past 40 years, and demands a prominent place in new reactor safety analyses. Defense in depth is the appropriate place to deal with both known and unknown uncertainties. While we know to some degree the level of uncertainty in a PRA or thermal hydraulic analysis, we are simply unable to anticipate the combination of errors of omission, commission, random failures and malfeasance that may afflict an operating reactor. Thus, a blend of deterministic and probabilistic requirements that are performance based and incorporate the requisite degree of conservatism is likely the best path towards establishing requirements for advanced gas reactors.[10]

Substantial efficiencies can be gained through international cooperation and coordination of the safety review of new advanced gas reactor designs. At present, there are gaps in the spectrum of internationally accepted codes and standards dealing with nuclear grade graphite and the fabrication of graphite components for use in HTGRs; with thermal-hydraulic codes for use in the complex geometries of a Pebble Bed Modular Reactor (PBMR); nucleonic codes that can accurately predict the transient and accident response of a loosely coupled, statistically bounded pebble bed core configuration; and in the member nations' nuclear regulations themselves which may have been written, used, and optimized to assure the safety of water cooled reactor technologies. A set of internationally agreed upon basic licensing requirements that are independent of a particular reactor type would be a step forward in achieving an internationally accepted design review upon which member states could rely when conducting their country specific licensing reviews. Additionally, international cooperation and agreement on requirements for siting gas reactors (the acceptable environmental impact, the appropriate

[10] Trip Report D.A. Powers, HTGR Safety and Research Issues Workshop held October 10-12, 2001 NRC ADAMS Accession Number ML 020450645

source term, and the size of the emergency planning zone) would help address the issues that arise when a site is selected in the vicinity of an international border.

The existing operating experience data base will need to be mined to extract the germane lessons learned from an enormous, fragmented operating experience data base that was generated largely from the experiences of water cooled reactors. Further, gas reactors will not only present relatively new operating challenges, they will very likely be constructed differently than the current world-wide fleet of reactors. Specifically, modular construction techniques will result in key design, construction, and testing activities being conducted, perhaps simultaneously, in various parts of the world. New approaches to the regulatory model of inspection, oversight, and quality assurance will be necessary to manage a new and geographically diverse construction program.

A surge in design and licensing activities for advanced gas reactors will draw further from the already limited number of experienced, knowledgeable experts in the world's regulatory authorities. Succession planning that anticipates the needed knowledge and experience, and equips those experts with the appropriate tools and methods to perform the needed safety reviews should be a high priority.

Finally, the roles of the various organizations involved with assuring nuclear safety should be clarified to help assure that the research and development activities necessary to support design, construction, licensing, and operation of advanced gas reactors is performed efficiently and comprehensively.

## 5.2 Analyses[11]

The needed tools and methods to perform the analyses necessary to assure the safe operation of advanced gas reactors do not yet exist. Accident, transient, human performance, and design analyses will likely present challenges not yet seen or faced in the current fleet of water cooled reactors.

In the case of accident analyses and PRAs, the appropriate treatment of passive safety systems, selection of the bounding set of initiating events, the performance of graphite and metallic components in the high temperature helium environment, and the treatment of uncertainties will have to be addressed. For high temperature gas reactors, it is not clear that core damage frequency (CDF) and large early release frequency (LERF) are the right metrics to use to measure success. It will be difficult to obtain the failure data needed for components and materials with limited operating experience, and it will be challenging to model digital control systems and automated control rooms relative to human intervention. Assumptions will need to be made relative to the transient arrival rates associated with linking the primary plant to a Brayton cycle balance of plant. Since external events will likely dominate the overall risk, which external events are considered credible will play a key role in the overall risk of a facility. Additionally, risk has been historically calculated per reactor, but with as many as 10 modules at a site being

[11] USNRC Office of Research Advanced Reactor Research Plan (March 2002) NRC ADAMS Accession Number M L021570221

controlled from one control room, the question of site level PRAs with plant (module) interactions considered needs to be addressed. Management of severe accidents raises additional questions. Should air intrusion, water intrusion, and major graphite fires be required to be addressed?

Human factors analyses and man-machine interface assumptions will provide new and different problems. For example, human reliability analyses for slowly developing accidents will need to be addressed, as will the role of the operator during normal, transient, and accident conditions. Will the operator be in an oversight and action verification role, or will the requirement for immediate actions to be taken by the operator be acceptable? Appropriate staffing levels for multiple module control rooms will need to be established. How many modules are too many to be controlled from one control room? What is the appropriate role of automation in routine, transient, and accident operations? How will complex decision making software and modern control theory methods be validated? What are the appropriate standards for operator training and qualifications in a digital automated operating environment? What is the appropriate role of procedures in this environment? And finally, how can the human-system interface be optimized to enhance safety? These types of challenging human factors issues may be the limiting cases for PRAs when the tools are available to model human errors of omission and commission in the complex operating environment of an advanced high temperature gas reactor.

Design analyses will require development of appropriate thermal-hydraulic codes to deal with the complex geometries and uncertain core configurations of the pebble bed design. Fuel performance and the performance of metallic and graphite components will need to be understood under transient and accident conditions to serve as inputs to the PRA. Of particular note will be the digital control room design analyses and the fundamental questions of the acceptable level of automation in a nuclear control room. Should automated control be permitted for the primary plant? Should it be permitted for the balance of plant and switchyard? Should automated startups and shutdowns be permitted? Should automated transient and accident response be permitted? Further, the extent to which modern control theory (neural nets, fuzzy controllers, etc,) can be employed to accomplish this automation will need to be justified through analysis, testing, and simulation.

### 5.3 Waste

Management of spent fuel for advanced HTGRs, especially the PBMR will provide new challenges simply due to the form of the fuel. In the case of a 10 module PBMR facility, it will discharge over 1,000,000 spent fuel spheres per year. These spheres will occupy about 128 cubic meters, and create some challenges relative to inventory and criticality control. In the case of inventory, each sphere lacks a unique identifying mark, making unit level inventory difficult. A coarser volumetric approach would likely be taken, but considering the public concern generated in the United States over a misplaced fuel rod at the Millstone facility, it is not clear that control of inventory at anything less than the component (or fractions of component level for damaged fuel) will be acceptable. In the

case of criticality control, the fuel is inseparable from the graphite moderator, thus requiring the presence of poison within the spent fuel storage facility. The statistical nature of the distribution of fuel could also conspire to make one section of the spent fuel array particularly reactive. Criticality control events at fuel fabrication facilities have shown that processes in place to exclude moderator from an area occasionally fail, as do geometry and quantity controls, thus, care will need to be exercised in the management of criticality during the storage, transportation, and disposal of spent fuel pebbles.

Long term disposal options will need to be rethought relative to the disposal of coated particle fuel. Can and should it be separated from the graphite matrix binder? Will the water cooled reactor long term geological repository solutions bound the new fuel types for advanced HTGRs? Is reprocessing an economical option?

### 5.4 Research

Successful deployment of advanced HTGRs will depend to a large extent on the scope and quality of the research done to anticipate and address the large number of challenges that the introduction of any new technology presents. As the various vendors, utilities, regulatory bodies, universities, and other research institutions take up this challenge, it is important to provide a means to coordinate these efforts so that duplication is minimized and on-going efforts benefit from related work that preceded it.

Clarifying the roles of international organizations relative to the coordination of research activities may help provide this needed coordination. The IAEA, OECD/NEA, and WANO are examples of the types of international organizations that could effectively coordinate the world's research supporting advanced gas reactors.

Generally, research needed to support advanced HTGRs can be grouped under the following general topics:

1. Transient and accident analysis

2. Materials applications in high temperature environments

3. Fuel performance

4. Digital control systems and automation

5. Application of risk technologies

## 6.0 Important Policy Questions

Considering the gaps, challenges, and issues listed in Sections 4.0 and 5.0 of this report, a number of themes emerge as key technical or policy issues, the resolution of which will have a direct impact on the safety, design, and operation of advanced HTGRs. A list and brief discussion of these key technical and policy issues follow:

### 6.1 Should new safety goals be developed that will drive advanced reactors to be substantially safer than the current generation of light water reactors?

Many of the design requirements and guiding principals for the current generation of nuclear reactors were based on conservative judgments, deterministic methods, and defense in depth philosophies. As probabilistic methods were brought to bear, safety goals were established, and imposed on an already existing reactor design. These safety goals and the probabilistic methods permitted a clearer understanding of the potential weaknesses in designs and facilitated the enactment of safety modifications.

At this point, the opportunity exists to take a fresh look at the world's safety goals with mature probabilistic safety analysis tools to establish what should be achieved as opposed to what can be achieved. Safety goals can become a principle design input for the new generation of advanced reactors. Level 2 PRAs which consider containment performance, and Level 3 PRAs which consider the source term and consequences of an accident will be needed along with rigorous PRA quality standards to provide the basis for much of the regulatory and safety decision-making. Consideration should also be given to risk assessment at the site level. With current plans to co-locate as many as 10 PBMR modules at one site controlled from a common control room, the integrated site risk and module interactions should be considered relative to safety goals.

Similarly, performance based standards and requirements can be developed which are independent of a particular technology. These technology neutral requirements can help achieve a consistent application of safety standards across both technologies, and nations.

### 6.2 What are the appropriate requirements for security, and should containments be required?

A new design provides the opportunity to establish optimal security provisions at a time when it is the easiest and least expensive to implement them. To achieve this efficiency however, key decisions need to be made early in the design process, and the greater the international consensus on these design inputs, the greater the cost savings. The types of questions that need to be answered are: Should the reactor be designed to withstand an aircraft strike? How big and how many aircraft? Should key safety systems be bunkered? Should the control room be bunkered? Is a containment necessary from a security standpoint? Should the facility be sited underground? What level of digital control and automation is acceptable, and how can cyber-security be assured? What degree of protection is appropriate for spent fuel?

Nations tend to define the threat their nuclear plants must face in their own terms and as a result of their own special circumstances and experiences. None the less, a core set of minimum requirements agreed to internationally will give the designers a starting point for the basic design that can be modified more efficiently based on the particular needs and requirements of the licensing authority.

The question of whether a containment or confinement provides the necessary barrier to fission product release may be settled by policy decisions regarding the need for security and defense-in-depth, rather than through fuel reliability and probabilistic analyses. If consensus is achieved to harden nuclear facilities against impact from a large aircraft for example, then a containment, or even multiple concentric containments would seem to be a likely outcome. Public opinion will play a role here as well. Plants with a containment may be more widely accepted by the general public, especially when the argument for confinement involves complex analyses regarding fuel failure, fission product transport, and probability, and the argument for containment rests with Chernobyl.

Safeguarding fissile material from diversion will provide some different challenges for the PBMR. The small size of a fuel pebble makes theft easier, and the large number of pebbles makes inventorying spent fuel at the fuel pebble level very difficult. Once stolen, a pebble could easily be used as a radiological dispersion device (dirty bomb) or an improvised exposure device.

Extracting usable fissile material from stolen fuel pebbles would be very difficult due to the low enrichment at the front end of the fuel cycle and the high burn-up at the back end. However, consideration would have to be given to alternate approaches of diverting fissile material such as cycling depleted uranium spheres through the core to breed plutonium.

### 6.3 Is this the right time to develop an international certification process to facilitate individual member states siting and licensing decisions for advanced reactors?

Individual safety and licensing reviews by member states will be expensive and to some degree duplicative. Is this the right time for the development of codes and standards that are, to the extent possible, technology neutral? Can a common set of safety standards be established that satisfy all member states' minimum safety standards for licensing? Can research, analytical methods, and safety reviews be coordinated or combined to prevent duplication and allow member states to rely on a common body of work for their safety and licensing decisions? Can a single comprehensive design basis be developed that member states can rely on to perform their licensing reviews?

The current realities are that a nuclear accident anywhere in the world will affect the nuclear programs everywhere, and the business of designing, building, and supporting operating nuclear plants has become multi-national. Thus there is clearly a shared responsibility for safety and a shared stake in the outcome. However, since the responsibility of assuring safety, protecting the public, and maintaining a viable energy infrastructure is so central to a nation's responsibility, it is unlikely that any nation would relax its sovereign authority in this area.

Ultimately, siting, licensing, and safety decisions must and should be made by the local regulatory authority. Much can be done however, to make these regulatory decisions more efficient and consistent world-wide. One such approach is a multi-national design approval program to facilitate the coordination of safety assessments, help harmonize

international standards and national safety regulations, and facilitate international trade by achieving increased compatibility of these standards,.

### 6.4 What level of automation and digital control is acceptable for advanced HTGRs?

Staffing, operator training and qualification, human-machine interface, digital control systems, and modern control theory are issues whose resolution will have a substantial impact on the control room design, operational philosophies, and economic viability of the advanced HTGRs. How many PBMR modules can be safely controlled from one control room? How many operators are necessary per control room per module? To what extent can modern control theory be applied to decision making for routine operations, transients, and accidents? Can control systems be connected to the internet? Should automatic generation of software for safety related applications be permitted? Can decision-making and safety related software be adequately validated and verified? Should on-line monitoring and self diagnostic systems be required?

These questions deal with the fundamental issue of the role of the human in the human-machine interface. As control systems inevitably move towards more precise control of increasingly complex processes, the use of smart controllers, advanced decision-making algorithms, and wireless connectivity will bring to bear the very difficult issue of cyber-security. Concerns about the transition to the year 2000's (Y2K) effect on computer systems, and the resulting realizations of how pervasive programmable decision-making has become utilizing, for example, electronic programmable read only memory (EPROM) chips is a good starting point for considering the effect of automation on the nuclear control room. The use of complex stand alone digital control systems in nuclear plants is already commonplace. As systems are put in place to facilitate interaction between control systems, such as radio frequency identification and detection (RFID) and wireless computing and connectivity, unanticipated interactions and the susceptibility to hacking and other malicious acts increases.

Control room staffing and decisions regarding work allocation will need to be addressed early, and are key factors in determining the amount of automation needed. Current philosophies vary internationally regarding how promptly operators should be expected to intervene during an off-normal condition. While decision-making by humans eliminates the ability for cyber-terrorists to act, in the case of Three Mile Island the lack of understanding of the physical processes at work caused the operators to intervene with the wrong action and cause the accident, and in the case of Chernobyl, once the conditions for the accident had been established, it occurred much too quickly for operator action to have any mitigative effect.

### 6.5 Will the industry be able to safely manufacture and manage the very large number of fuel pebbles anticipated?

A ten module PBMR site, with about 360,000 fuel spheres per core, replacing a third of the core each year must manage an inventory of over 3.5 million fuel spheres in the cores

at any given time, with about 1.2 million spheres discharged as spent fuel each year. The reactors will be refueled with another 1.2 million new spheres during each year of operation. With even a small fleet of operating PBMRs, the number of fuel pebbles can easily reach tens or even hundreds of millions.

The fuel pebble manufacturing process involves inherently random processes such as chemical vapor deposition and gel precipitation droplet formation, thus resulting in statistical distributions of attributes such as density, anisotropy, thickness, and microstructure. Core physics will be constantly changing as the pebbles flow through the core, necessitating some statistical bounding of key parameters. Differences between the center of mass and the center of gravity for individual pebbles, and surface defects and irregularities may result in non-linear conditions governing pebble flow through the core which can make it impossible to reliably predict the transit time for any particular fuel pebble, or even the fuel pebble packing density within the core.

Considering the experience with fuel in light water reactors, and the anticipated failure modes for TRISO coated fuel spheres of internal overpressure, tensile failure of the SiC layer, chemical attack, thermal dissociation, and mechanical overstress, it is likely that a means of identifying and tracking the history of individual fuel spheres will become necessary. Tracing fuel failures or nonconforming conditions back to a manufacturing line or process will be necessary to establish the extent of the condition of emerging failures and to take effective corrective action.

The spent fuel waste solutions being developed for light water reactor fuel will need to be rethought for advanced HTGR fuel. Distribution of the fuel within the graphite moderator presents a different situation from light water reactor fuel. Consideration of the long term disposal of the fuel at the front end of the design and manufacturing process can ease the disposal problem in the long run.

### 6.6 Is the PBMR design ready for a full scale demonstration project or is a prototype needed to conduct necessary confirmatory research prior to commercial operation?

A considerable number of uncertainties exist relative to scaling up the results observed in the Chinese research reactor and applying the results of Germany's operation of the AVR to full scale operation of an advanced HTGR. The thermal-hydraulic codes and the core physics are examples of the analyses that may benefit from prototype testing. Pebble flow through the core under varying conditions, the effects of air and moisture intrusion into the coolant gas, and the effects of temperature and radiation on metallic and graphite components are also areas that may benefit from the enhanced testing environment of a prototype reactor. Additionally, confirmatory research is needed in transient analysis, nuclear physics, severe accidents, source term, fuel performance, material performance and failure mechanisms, digital control systems, application of modern control theory, fuel handling, waste management, and other areas.

Inservice inspection requirements and techniques may require a shift to on-line real time monitoring due to limited shutdown time and limited accessibility of some metallic and graphite components. Demonstration of the validity of monitoring techniques to detect insipient failures may be easier in a prototype environment. An integral approach to monitor the performance of equipment may be acceptable since the HTGR design has a very low radioactivity release due to the robust TRISO particle fuel and slow progression of transient events during operation and accident sequences.

The balance of plant for the PBMR also provides substantial new design and operational challenges that may benefit from a prototype. The Brayton cycle power generation system includes a turbine that has not benefited from extensive operating experience to inform judgments relative to overall reliability. It is likely that the transient arrival rate from an unproven balance of plant will be high and may complicate the testing and operation of the primary system.

### 6.7 Are the needed codes and analytical tools available to perform the required safety reviews?

Work remains to be done to provide conservative, validated design and licensing codes and analyses for an advanced gas reactor. Specifically, level 2 and 3 PRAs, fuel performance codes, thermal-hydraulic analyses, computational fluid dynamic simulations of severe accident progression are examples of analytical tools needing further development and refinement. Some of these tools require a better understanding of phenomena such as metallic and graphite material performance under long term operations and accident conditions. In the case of PRAs, core damage frequency and large early release frequency may not be the best measures of safety for an advanced gas reactor.

Relative to severe accidents, the events of September 11, 2001 make it likely that advanced HTGRs will need to consider mitigation of large explosions and fires. Additionally, the possibility of large air intrusion events and the resulting fire that could consume a fission product barrier is a likely candidate for severe accident consideration. Another effect unique to a PBMR is the high point stresses associated with fuel sphere interaction during a severe seismic event. The full scope of severe accidents that should be considered for advanced HTGRs is not clear, and will require carefully performed probabilistic risk assessments to guide our thinking.

The integrated application of digital control systems, automation, and modern control theory will also be faced for the first time in a nuclear application with the safety review of an advanced reactor. This review will likely require new tools, policies, and design criteria that can be used to assure the validation of safety related applications of “smart” technology and decision-making software.

### 6.8 Are the behaviors of materials under HTGR conditions sufficiently well understood?

Metals used at high temperatures in a helium environment can be expected to experience aging embrittlement, sensitization, carburization, oxidation, creep, fatigue, and unanticipated effects from conditions such as impurities in the helium coolant which could result in stress and crevice corrosion cracking. There is currently limited operational experience for nuclear application of metals at the temperatures being contemplated for a PBMR.

Graphite is known to be susceptible to dimensional instability, thermal conductivity changes, oxidation, creep, fatigue, and loss of strength in nuclear applications. Sensitivity to these effects as a function of the graphite form (sleeves versus blocks), manufacturing techniques, and purity of the graphite are less well understood. Considering the planned use of graphite in structural applications, advanced non-destructive testing techniques may be required to be used in concert with real time in-service inspection to assure degradation of key components is appropriately monitored.

The effect of sustained high temperatures on concrete is also a potential material condition effect that will need to be understood relative to the specific design applications. Sustained exposure of concrete to temperatures above 260 degrees C (500 degrees F) can be problematic.

Considering the known problems and sensitivities of metals, graphite, and concrete to the expected operational conditions, and the sure knowledge that unanticipated service related material failures will occur, this is an area where anticipatory research can contribute greatly to the successful implementation of commercial HTGRs. On-line refueling, limited shut-down periods, and infrequent full core off-loads will also make it difficult to monitor and inspect components for known and unanticipated failure mechanisms. Improvements in NDE and ISI techniques and in real time monitoring can help to offset the reduced inspection opportunities.

## 7.0 References

1. USNRC Office of Research Advanced Reactor Research Plan (March 2002) NRC ADAMS Accession Number M L021570221.

2. USNRC SECY 01-0070 Plan for Pre-application Activities on the Pebble Bed Modular Reactor April 25, 2001, NRC ADAMS Accession Number ML 031000388

3. Presentation by Andrew C. Kadak, MIT Advanced Reactor Projects April 2005

4. USNRC SECY 05-0006 Attachment 1 Technology Neutral Regulatory Structure NRC ADAMS Accession Number ML 043350086

5. The Modular Pebble Bed Reactor Concept, Andrew Kadak, www.min.uc.edu/nuclear/kadak/sld001.htm

6. South Africa Pebble Bed Modular Reactor Company Pty, Ltd Web page: www.pbmr.com/

7. NUREG/CR-6844 TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables (PIRTs) For Fission Product Transport Due to Manufacturing, Operation, and Accidents.

8. Summary of NRC Delegation Visit to Germany on Safety Aspects of HTGR Design Technology August 21, 2001, NRC ADAMS Accession Number ML 012330131

9. Peoples Republic of China Pebble Bed Reactor Program December 20, 2002, NRC ADAMS Accession Number ML 031000325

10. Exelon PBMR Fact Sheet January 9. 2003, NRC ADAMS Accession Number ML 0302103559

11. Trip Report D.A. Powers, HTGR Safety and Research Issues Workshop held October 10-12, 2001 NRC ADAMS Accession Number ML 020450645

12. Exelon Position Paper Addressing Significant Conclusions in D.A. Powers' Trip Report Covering the HTGR Safety and Research Issues Workshop October 10-12, 2001, NRC ADAMS Accession Number ML 021640672

13. PBMR: Thermal Hydraulics Analysis September 6, 2001, NRC ADAMS Accession Number ML 012610169

14. Historical German Fuel Qualification Process Document, May 30, 2002, NRC ADAMS Accession Number ML 021550118

15. USNRC NUREG/IA-0137 S Study of Staffing Levels for Advanced Reactors

16. USNRC NUREG/CR-6749 Integrating Digital and Conventional Human-Systems Interfaces: Lessons Learned from a Control Room Modernization Project

17. USNRC NUREG/CR-6842 Advanced Reactor Licensing: Experience with Digital I&C Technology in Evolutionary Plants

18. USNRC NUREG/GR-0020 Embedded Digital System Reliability and Safety Analyses

19. USNRC NUREG-1338 Pre-application Safety Evaluation Report for the Modular High Temperature Gas Cooled Reactor (MHTGR)

20. Eskom Generation PBMR Safety Case Philosophy, October 2004

## 8.0 List of Abbreviations

| | |
|---|---|
| AGR | Advanced Gas Reactor |
| AVR | Arbeitsgemeinschaft Versuchreaktor |
| BISO | Bi-Isotropic Coated Particle Fuel |
| CDF | Core Damage Frequency |
| DOE | Department of Energy (US) |
| C | Celsius |
| F | Fahrenheit |
| HTGR | High Temperature Gas Reactor |
| HTR | High Temperature Reactor |
| IAEA | International Atomic Energy Agency |
| ISI | Inservice Inspection |
| JAERI | Japanese Atomic Energy Agency Research Institute |
| LERF | Large Early Release Frequency |
| LWR | Light Water Reactor |
| mm | Millimeter |
| MWt | Megawatt thermal |
| NEA | Nuclear Energy Agency |
| NDE | Non-Destructive Examination |
| OECD | Organization for Economic and Commercial Development |
| PBMR | Pebble Bed Modular Reactor |
| PRA | Probabilistic Risk Assessment |
| SiC | Silicon-Carbide |
| TRISO | Tri-Isotropic Coated Particle Fuel |
| UK | United Kingdom |
| US | United States |
| WANO | World Association of Nuclear Operators |

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