

2. MHTGR DESIGN CHARACTERISTICS AND FEATURES RELEVANT TO THE SAFETY CASE

2.1. MHTGR DESIGN SAFETY CONCEPT

The MHTGR's fundamental safety objectives, requirements and design guidelines are based on the specific design characteristics and inherent safety features noted below:

- High quality ceramic coated-particle fuel of proven design, which adequately retains its ability to contain radioactive fission products over the full range of operating and accident conditions.
- A single-phase inert coolant (helium), with no heat transfer limits that would be associated with phase change.
- Post shutdown decay heat removal achievable through conduction, natural convection and radiation heat transfer, limiting maximum temperatures to values consistent with coated fuel particle and structural design limits.
- Combination of low core power density, large reactor core and internals heat capacity, high core thermal conductivity and large fuel thermal margins, resulting in very long times (days) for evolution of response to loss of normal shutdown functions without protective actions.
- Fuel temperature margins and negative temperature-reactivity coefficients sufficient to accommodate any foreseeable reactivity insertions during startup and power operation without damage to the fuel

If successfully developed, the defining safety characteristic of the MHTGR will be that its primary defence against serious accidents is achieved through its inherent design features. Active safety systems or prompt operator actions are not required to prevent significant fuel failure and fission product release. The plant is designed such that its inherent features provide adequate protection despite operational errors or equipment failure. A primary design characteristic is the limitation of rated thermal power to a small fraction (on the order of 6 to 20%) of typical power levels for the large water reactors upon which the existing safety requirements are based. This is necessary to provide for removal of post shutdown decay heat using only passive means. Specific features, characteristics, and related safety issues are discussed in this section.

2.2. COATED FUEL PARTICLE

MHTGR fuel is a ceramic, and is therefore able to withstand much higher temperatures than can fuel elements with metallic cladding. The design of today's coated fuel particle (CFP) has evolved empirically over several decades from a single layer of anisotropic carbon, to BISO (buffered isotropic pyrolytic carbon) to the current TRISO (triple isotropic layers) design. TRISO CFPs are small, typically ~1 mm diameter. In the TRISO design, the fuel kernel (typically LEU-oxide or -oxycarbide or Pu-oxide), is surrounded by a porous buffer layer to absorb fission gasses. Next there is an inner pyrolytic carbon (IPyC) coating; a silicon carbide (SiC) layer (or zirconium carbide – ZrC – in some advanced fuels) layer, and then an outer pyrolytic carbon (OPyC) coating. Variations in CFP design are primarily in fuel type, kernel size, buffer and coating thickness and microstructure, and in methods for fabrication and quality control (QC) screening.

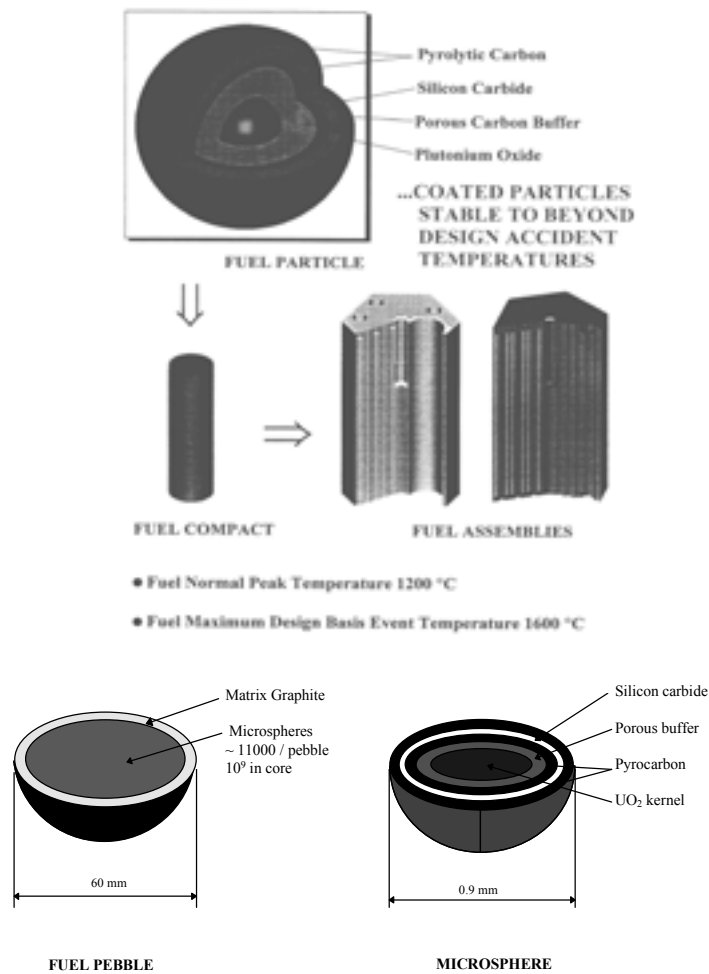


Fig. 1. TRISO fuel for pebble and prism designs.

Since the CFP barriers form the primary line of defence against fission-product release, good performance is essential to the success of the MHTGR design. For the most part, CFP designs have been arrived at empirically. A comprehensive analytical fuel performance model — accurately relating its (statistical) resistance to failure — has not been successfully developed due to the complexity of treating the combined effects of coating microstructure variations, variations in location and characteristics of microscopic imperfections, fission product chemical interactions along grain boundaries, fission gas pressure build-up, long term temperature and irradiation effects, etc. However, the empirical basis for CFP performance, a product of decades of development in many countries, is extensive. An IAEA Co-ordinated Research Project (CRP) on Validation of Predictive Methods for Fuel and Fission Product Behaviour was conducted from 1992 to 1996, with participants from China, France, Germany, Japan, the Russian Federation, the United Kingdom and the USA. The objectives of this CRP were to review and document the status of the experimental data base and of the predictive methods for gascooled reactor (GCR) fuel performance and fission product behaviour, and to verify and validate methodologies for the prediction of fuel performance and fission product transport. The results of this comprehensive international study of CFP performance are reported in IAEA TECDOC-978 [7].

CFP loss-of-function implies inability to retain fission products. Loss-of-function can range from long term diffusion of specific fission products (e.g. caesium) through the coating layers, to sequential or simultaneous coating layer structural failure. There are many factors affecting fission product retention capability of any given CFP, including as-manufactured dimensions, coating layer microstructure, and chemical impurities; irradiation flux and temperature history, and chemical attack. In normal operation a particular concern for gas-turbine (GT) designs is the diffusion of Ag-110m through the intact SiC layer at high operating temperatures. Silver deposition on turbine blades (and elsewhere) could lead to significant personnel exposure during maintenance, and possible material damage problems. Ag-110m precursor fission yields are over 50 times higher for Pu than for U, so it is more of a concern for Pu-burner designs. In accident conditions, CFP time/temperature history during the event tends to dominate the fission product release rate, particularly with regard to diffusion releases. Chemical attack from within (such as Palladium attack on SiC) or from without (such as via air or moisture from ingress events) may also be a factor. CFP compaction methods (prismatic core design compacts or PBR pebble elements) can also affect failure statistics.

Diffusive release of several fission product species appears to begin at about 1600°C, although heating tests of irradiated CFP show very little release in the 1600°C area even for relatively long periods (typical of times at or near the peak in long-term depressurization accidents). 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. It should be noted also that predictions of peak fuel temperatures vs. time analyses often neglect to mention that a relatively small portion of the core fuel is at or near the peak (3-D time-temperature percent-fuel failure models account for this effect in core release predictions).

2.3. HELIUM AS PRIMARY COOLANT

Helium gas pressurized to several MPa is employed as the primary system coolant. Helium is a single phase noble gas with no heat transfer limits associated with phase change. The absence of heat transfer limits (e.g. departure from nucleate boiling — DNB or critical heat flux — CHF) in addition to the core's large thermal inertia may eliminate any safety related need to monitor short term variations in core power and temperature distributions. For the same reason, large local temperature increases during anticipated operational occurrences are less likely to occur. This can offer major operational benefits such as elimination or simplification of safety related monitoring and protection systems, and related surveillance and in-service inspection requirements.

In addition, due to the inert characteristics of helium, no significant chemical attack on fuel and other components would be expected if the contamination levels are kept low. Also, helium has no significant reactivity effects, and a relatively low amount of waste is generated due to activation and/or transmutation of the coolant impurities and corrosion products.

On the other hand, it is relatively easy for helium gas to leak from the primary circuit, especially at the elevated temperatures and pressures (although helium leakage does not cause any important safety issues). Thus for operational purposes, careful consideration is required for the design, fabrication, inspection and maintenance of the primary circuit. A monitoring system to detect the leakage should be able to identify leakage locations.

Helium will not condense if contained in a structure at normal temperatures following depressurization. Thus the pressure would reduce somewhat in accordance with the ideal gas law due to cooling, but would remain relatively high until the helium leaks out of the structure. In contrast, steam released from a water cooled system will condense on structural materials and components, resulting in a relatively rapid decrease in pressure. This characteristic substantially reduces the effectiveness of a conventional containment structure for a helium cooled system relative to a water cooled system. By retaining the helium following a depressurization, the gas leaking from the containment (typically specified as $\leq 1\%$ /day for existing reactors) can serve as a transport mechanism for radionuclides which would be released from the fuel during a long term heatup. Thus in many important scenarios a conventional containment would result in a higher offsite dose than a filtered vented confinement design.

2.4. DECAY HEAT REMOVAL VIA PASSIVE MEANS

MHTGR designs typically rely on a passive ultimate heat sink system for removal of decay heat in the case of failure or unavailability of all active core cooling mechanisms. Under these conditions, core heat removal is accomplished via heat transfer from the core to the non-insulated reactor pressure vessel via conduction, radiation and (if coolant is present) convection, and from the vessel to the reactor cavity by radiation and convection. 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.

In typical designs the RCCS is fully operational during normal reactor operation, and there are no mechanical actions needed for it to function during a loss-of-forced-convection (LOFC) event. However, the operational mode may be different (e.g. transition from forced convection to natural convection RCCS cooling flow). Because of the multiple objectives and wide range of operational conditions, along with its necessarily massive size, the RCCS design and fabrication is challenging as well as crucial. In several instances, the performance of RCCS designs have been found to be difficult to predict with regard to local temperature distributions in the reactor cavity. Due to its location (in the reactor cavity), major repair and/or replacement may be very difficult.

The heat load distributions for depressurized and pressurized LOFC accidents are quite different and may affect RCCS design requirements. For the depressurized case, the peak core temperatures tend to be near the level of the core beltline, while for the pressurized case, peak temperatures and heat loads are near the upper part of the vessel due to convection heating effects.

Additionally, accident analyses of some loss-of-cooling events for some designs have shown that a total functional failure of the RCCS has remarkably little impact on predicted peak fuel temperatures. However, variations among MHTGR designs may significantly affect the functional requirements of the RCCS. For example, analyses have shown that for the higher power designs (~ 600 MW(t), RCCS operation is required during these accidents to protect the reactor pressure vessel from damage, while its failure does not necessarily lead to vessel damage for the lower power designs (~ 250 MW(t)). Over the past two decades there has

been a wide range of experimental and analytical work in this area in support of several MHTGR designs. CRP on Heat Transport and Afterheat Removal for Gas Cooled Reactors Under Accident Conditions was conducted from 1992 to 1997, with participants from China, France, Germany, Japan, Netherlands, the Russian Federation, and the United States of America. The objective of this CRP was to establish sufficient experimental data at realistic conditions, and validated analytical tools to confirm the predicted safe thermal response of MHTGR during accidents. The scope included experimental and analytical investigations of heat transport by natural convection, conduction, and thermal radiation within the core and reactor vessel, and afterheat removal from the reactor vessel. Code-to-code and code-to-experiment benchmarks were performed for verification and validation of the analytical methods. The results of this comprehensive international study of MHTGR passive decay heat removal are reported in IAEA-TECDOC-1163 [8].

2.5. LARGE THERMAL INERTIA, LOW POWER DENSITY, LARGE TEMPERATURE MARGINS

The combination of an MHTGR core's large thermal inertia (high heat capacity and low power density) typically results in long, slow core heatup (and cooldown) transients for loss-of-forced-convection and loss of coolant pressure events. These attributes, coupled with the core's high effective thermal conductivity attributes, tend to delay the occurrence of peak values of fuel temperatures for days, when the magnitude of the afterheat is considerably reduced. Very long response time also allows considerable opportunity for operational corrective measures to be taken.

The thermal response, in combination with the time-at-temperature effect on fuel fission product retention and the helium characteristics noted earlier, fundamentally alters the effectiveness of strategies for fission product containment. For example, in a depressurisation accident, the predicted small fission product release from the fuel occurs long after the depressurization is completed, even for relatively small leaks. 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 is inward. However, if the released gas is contained, with a small (e.g. 1%/day) leakage rate, the leakage flow and slowly decreasing pressure would provide a mechanism for fission product transport. Thus attempting to contain the leaking helium can result in a higher fission product release rate for some of the most limiting events. This effect was observed during the safety review of an earlier MHTGR design [9].

For annular core designs, the peak fuel temperatures in the depressurized accident scenarios (for a given total core power and vessel size) are reduced relative to those for a cylindrical active core. Increases in the core graphite conductivity, which can vary widely with irradiation and irradiation-temperature history, can also result in reduced peak fuel temperatures as the core graphite anneals, effectively increasing conductivity with increasing temperature. Thermal radiation (T^4) effects also tend to become the dominating heat transfer mechanism for both prismatic and pebble cores at the very-high (accident-range) temperatures.

2.6. TEMPERATURE MARGINS AND NEGATIVE TEMPERATURE-REACTIVITY COEFFICIENT

A negative temperature-reactivity coefficient can be attained in the MHTGR for the entire fuel cycle and over the full temperature range of concern, as seen in most of the other

types of reactors. In combination with the characteristics of large margin between fuel operation and fuel damage temperatures, and relatively low excess reactivity, as discussed below, power control and reactor shutdown can be ensured naturally. These characteristics significantly reduce the safety significance of the reactivity control and reactor shutdown systems.

In the pebble bed reactor, the reactor core can operate with low excess reactivity by adjusting the number of fuel balls introduced during operation. Protection and management of abnormal reactivity insertion conditions could be provided by inherent features, simplifying the design of active/passive protection or mitigation systems to assure safe shutdowns.

For the block type reactor, rather low excess reactivity can be attained by appropriate core design with burnable poison, optimized refuelling programmes, etc. Careful design and quality assurance/control would be required for the reactivity control and shutdown system, as well as countermeasures to the possible control rod housing failure causing rapid reactivity insertion (control rod ejection event).

2.7. FEATURES COMMON TO MHTGRs AND OTHER FUTURE REACTORS

Simplification and use of passive systems

MHTGRs make extensive use of passive characteristics that offer the opportunity to eliminate or simplify active systems that rely on a large number of safety grade support systems by applying the advantages of simple gravity driven or thermal gradient driven safety systems. The challenge is to demonstrate the capability and the reliability of these passive systems, in particular for the long time accident response.

Standardization, prefabrication and modularity

The standardization, prefabrication and modularity of the facilities that will likely be part of the design, construction and operation of MHTGR with evident benefits on the economics of a single unit, will also lead to a simplification of the licensing through a certification procedure, and reduction of the construction time and licensing costs.

Applicability of PSA and risk-informed decision making

Because of the extensive use of passive components, the safety of these reactors is primarily determined by initiating events of very low probability (e.g. structural failures due to extremely rare external events). The consequences of these events are determined by the direct phenomenological response of the plant to these events, rather than by a sequence of failures of systems, which individually have higher probabilities and which can be analyzed and modelled with much less uncertainty. This aspect will pose significant challenges for the development and application of PSA methodologies to address these concepts.