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THERMAL HYDRAULIC SAFETY CONSIDERATIONS, METHODS AND RESEARCH FOR HIGH-TEMPREATURE GAS-COOLED REACTORS (HTGRs)

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1. ABSTRACT

A literature review is conducted on the approaches that High Temperature Gas-Cooled Reactor (HTGR) vendors, designers and R&D engineers have used to justify their safety features or claims. Sources of information include the open literature, including, published journal papers, conference proceedings, technical reports, and Phenomena Identification and Ranking Tables (PIRTs) available and general websites. This report focuses on the thermal fluid aspects, with brief discussion of neutronic behaviours, of HTGRs and aims to provide information and guidance for vendors on their design and licensing efforts as well as researchers for future endeavours. The scope includes identification of the important phenomena, and a summary of the current understanding and prediction capabilities of the phenomena. The report also includes recommendations for further developments in modelling/prediction approaches for HTGRs.

2. INTRODUCTION

A high-temperature gas-cooled reactor (HTGR) is a type of gas-cooled nuclear reactor aimed at producing a very high core output temperature. Common to nearly all designs of HTGRs, uranium is used as the fuel, graphite as the moderation and helium as the coolant. The coated fuel particles are embedded in either a rod compact inserted into a stacked prismatic block or a spherical compact that constitutes a pebble, thus the reactor core can be either a “prismatic block” or a “pebble-bed” core (PMR or PBR). The most recent significant development is that two full-scale pebble-bed HTGRs, the HTR-PM, each with a 100 MWe of capacity, went into operation in China in late 2021[1].

The HTGRs have various specific design and safety characteristics. The graphite structure has large thermal inertia and structural stability even at high temperatures with characteristics of low-power density, high effective core thermal conductivity, and large thermal margins to fuel failure. The graphite in the core and reflectors acts as a thermal buffer, absorbing both fission and decay heat during transients. The helium coolant is single phase, inert, with no reactivity effects. The uranium carbide coated fuel permits high burn up and retains fission products. Negative fuel and moderator temperature coefficients of reactivity, in conjunction with the negative reactivity feedback of the fission product during LOFC (loss-of-forced circulation) events are also essential factors for safety. It has a design basis accident decay heat removal system, typically a passive system utilising a natural-convection-driven process (the Reactor Cavity Cooling System-RCCS); and a confinement-style reactor building structure. The high core-exit temperature permits generating electricity and co-generating hydrogen using the high-grade process heat.

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Despite those positive safety features, HTGRs still have major deign and safety-consideration challenges, which are mostly linked to the fact that they are very different to light water reactors (LWRs) and hence the tools and knowledge accumulated for LWRs need to re-evaluated and re-developed; that they have complex plena designs; and that they have significant passive features which are of low knowledge level in general.

3. METHDOLOGY

Sources of information include the open literature, including, published journal papers, conference proceedings, technical reports, and Phenomena Identification and Ranking Tables (PIRTs) available and general websites.

Constructing Phenomena Identification and Ranking Tables (PIRT) is a useful approach to identify key challenges and technology and capacity gaps. A few PIRTs have been carried out for HTGRs. The first PIRT for HTGR was carried out in 2008 (Ball et al [2,3]) for generic designs including both prismatic and pebble-bed designs. The work was funded by the Department of Energy (DOE) and NRC for the next generation nuclear plant (NGNP). Even though this was completed some time ago, it is still widely used as a reliable source and is referred to as US-NRC PIRT herein. More recently, Schultz et al 2017 revisited the above PIRT for some transients in the context of modular reactors [4]. At about the same time, a PIRT was carried out and published for the pebble-bed design HTR-PM of INET in China [5]. The key phenomena identified in all these PIRTs are very similar and the knowledge levels assigned are also similar, though the recent INET PIRT reflects somewhat slightly higher confidence. We mostly refer to the US-NRC PIRT in the document.

Safety considerations and modelling are discussed under the topics of the core, accidents and plena. In each case, the phenomena and key safety considerations are described, which is followed by discussion on recent modelling research and capacities. The topics are as follows:

Core heat transfer under normal operation;

Accident phenomena including loss of forced circulation (LOFC), air ingress (following depressurized loss-of-forced circulation, D-LOFC), water/steam ingress;

Connected upper and lower plenum flow, Reactor cavity cooling system (RCCS) flow.

4. RESULTS AND DISCUSSION

Reactor core at normal operation conditions

1. The determination of core temperature is important in that it provides initial and boundary conditions for transient fault studies. This is of course in addition to the fact that the knowledge is also valuable in its own right from design and operation point of view.

2. It should be recognised that the core heat transfer calculation for HTGRs can be quite different from that for LWRs because of the strong dependence on conduction across the core, the reliance on passive cooling, and the differences in the plena. Consequently, previously developed methods might not be directly applicable. There are two main-stream approaches for HTGRs, that is, unit cell and CFD, each with different advantages and disadvantages. The former is simpler and homogenised models, whereas the latter provides local behaviours but are more expensive and can only be applied to a small portion of the fuel blocks. Coupled/combined modelling would be valuable development. This can include coarse-grid CFD, which may naturally take the advantages of both methods.

3. Bypass flow is the most important factor when estimating the peak fuel temperature for the core heat transfer calculations. There is no direct measurement of the bypass flow fractions, which may vary with small changes in geometry over the lifetime of the reactor. Hence uncertainties linked to bypass flow are large and is an area that needs further work to improve the confidence in core temperature predictions.

Plena and RCCS

4. Generally speaking, the lower plenum is of high importance and low knowledge level. Analysing coolant flow entry into and mixing in the lower plenum is important for identifying hot spots and investigating material or mechanical concerns due to excessive heating. The lower plenum flow distribution impacts on/may cause possible hot streaks and the stress distributions in the plenum and downstream components.

5. The upper plenum is important only during P-LOFCs (Pressurised-Loss of Forced circulations), when the flow direction is reversed and the top plenum receives flow from the core in the form of hot jets/plumes. This can lead to undesired hot spots, thermal striping, and material cracking in the severe conditions. Hence non-isothermal/isothermal single/multiple jets mixing are relevant phenomena that need to be studied.

6. Due to the complex nature of the plena, the traditional, typical, nuclear codes is not suitable, and often CFD models are used for detailed heat transfer calculations, which would in turn need system codes such as RELAP to provide boundary conditions. CFD is powerful but its prediction may vary from case to case and hence it needs careful validations for specific designs.

7. RCCS is inherently safe and reliable as a passive safety system utilizing thermal radiation and natural convection in HTGRs. RCCS is also a key component for heat removal in many fault transients. And the latter needs more careful study. For example, the decay heat removal capability of RCCS is required to maintain core design limits during LOFCs. Under such fault conditions, heat transfer mechanisms are thermal radiation and natural convection. Emissivities under various conditions such as irradiation are key parameters for radiation, whereas better thermos-fluid model is required for natural convection.

Fault transients

8. Heat conduction is the primary heat removal mechanism within the core in Depressurised-LOFCs, and the effective core thermal conductivity is the major factor on the peak fuel temperature. In addition, RCCS is the key method to transmit energy from core to the surroundings and hence its performance is the major factor for the vessel temperature (see above). For both pebble-bed and prismatic cores, core thermal conductivities are uncertain due to difficulty in comprehensive measurements and its variation with irradiation. Calculation of RCCS requires modelling of natural convection, which poses significant challenges.

9. The chimney effect in P-LOFC increases the core and the vessel temperatures near the top, therefore the prediction of the upper structure temperatures is crucial for P-LOFCs. Buoyancy-driven flow is mostly encountered in P-LOFCs. P-LOFC accidents are often considered as a single variation of D-LOFC and are treated as a check of the method.

10. CFD of a section of a graphite block has been an important for LOFCs. However, CFD modelling of buoyancy-driven flows needs to be further investigated to identify the appropriate turbulence modelling approaches (RANS, URANS, etc.), boundary conditions, and near-wall treatments, which crucially may change from design to design.

11. Air ingress may occur under extreme D-LOFCs when there are multiple/large breaks. Graphite and support structure oxidation, fuel oxidation and fission product release are the main challenges in air ingress. Most

studies of air ingress to date use simplified models and do not represent the specific HTGR geometries and operating conditions. As parameters such as air ingress rate and onset time for natural circulation are significantly affected by the pressure, temperature and geometry, further experimental and modelling studies on more representative geometries and operating conditions are needed to understand the small and medium break accidents.

12. The occurrence of water-steam ingress depends on the system pressure and temperature, and equilibrium should be maintained and monitored during operation. Small amounts of water-steam ingress may damage the operating ability of many components, such as the primary system structures, and affect reactivity/core power oscillations leading to reduced control rod worth; control rod mechanical interference, etc. Predicting the local break and system level water-steam transport is challenging because it requires knowledge of many factors, including critical flow, moisture levels and locations of condensation sites, stratified flow, gas production, decay heat levels, and phase change heat transfer.

5. CONCLUSIONS

The significant phenomena in the thermal fluids area include the primary system heat transport, such as, core thermal and flow behaviours which relate to the power-to-flow ratio and thus impact peak fuel temperatures and fuel performance in many events; postulated air ingress accidents that, however unlikely, could lead to major core and core support damages; the reactor cavity cooling system (RCCS) performance which impacts on fuel and component temperatures. The most significant reactor physics phenomena are often related to feedback coefficients, power distribution for normal and accident conditions, etc. Besides those identified aspects, fission-product transport and dose, high-temperature materials, graphite dust and process heat for hydrogen production potentially are of high importance but not reviewed in this report. Major safety challenges include heat removal capability, reactivity control, confinement of radioactivity, and the control of chemical attacks. These challenges can largely be translated to the prospect of high core temperatures under normal operation, RCCS performance in loss of forced cooling (LOFC) scenarios, peak fuel temperatures in Depressurised-LOFC events, and the uncertainties in air ingress that could lead to significant core and core support damage.

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