Very High Temperature Reactor (VHTR) Safety Assessment

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VHTR System Safety Assessment

In November 2013, the GIF Risk Safety Working Group (RSWG) was tasked with assessing the high-level safety design attributes for all six Gen IV nuclear energy systems. To help with this task, the VHTR SSC was asked to prepare a safety assessment document focusing on the VHTR concept. This document compiles VHTR safety-related information according to the "Table of Contents" proposed by the RSWG.

1. General overview of the performance goals

Gen IV nuclear energy systems share common goals, including improved sustainability, safety & reliability, economics, proliferation resistance & physical protection. The goal of safety & reliability was established in the roadmap of 2002 [1] and was further consolidated in the roadmap update of 2014 [2] mainly by the following definitions:

- excel in safety and reliability;
- have a very low likelihood and degree of reactor core damage;
- eliminate the technical justification for offsite emergency response.

For performance reasons, Gen IV systems are expected to operate under more demanding conditions than existing NPPs, e.g., VHTR coolant outlet temperatures in excess of 1000°C or the Super-Critical Water Reactor with coolant pressures in excess of 20 MPa. Thus, Gen IV systems need further improvements to achieve the required high level of safety & reliability.

2. Historical review of, and feedback from, past construction and operation experience

Gas Cooled Reactors (GCRs) have a long history dating back to the early days of nuclear energy development in the 1950s. Commercial GCR experience began with the U.K.'s Calder Hall plant, the subsequent 26 Magnox GCRs and 14 Advanced Gas-Cooled Reactors (AGRs), along with GCR NPPs in France, Japan and Spain. The commercialization of early GCRs and AGRs demonstrated the excellent performance of the main System Structure Components (SSCs) including the gas circulator, the steam generator, the graphite, and plant design schemes despite using different coolants and fuel forms than the current High Temperature Gas-cooled Reactor (HTGR).

Development of HTGRs began in the 1950s with the initial goal of improving thermal efficiency then focusing on safety at a later stage. HTGRs utilize graphite as moderator, and helium as coolant, and a fully ceramic fuel and core structure able to tolerate high temperatures under all anticipated normal and accident scenarios. In today's HTGRs, TRISO-type coated fuel particles are dispersed in a graphite matrix and formed into cylindrical compacts for a prismatic block type reactor or into spherical fuel elements for a pebble bed reactor. In recent tests, this fuel morphology is proven to retain fission products at burnup and temperatures well in excess of bounding operational and accident maximum values. In the 1960s, the prototype HTGR plants, i.e., Dragon, AVR and Peach Bottom, were built in the U.K., Germany and the U.S., respectively. Follow-on HTGR development included two demonstration plants, the Fort Saint Vrain (FSV) in the U.S. and the THTR-300 in Germany. Despite the decline of the nuclear industry in the mid and late 1980s, the successful operation of these HTGR NPPs not only demonstrated the successful development of components, graphite, plant design etc., but also provided initial experience with high performance TRISO-type fuel, coolant and material chemistry, and the response of the plant to faults and leaks. Process heat applications with coolant outlet temperatures as high as 950°C were also demonstrated.

During the 1980s, the modular HTGR concept was developed, primarily in Germany and the U.S, with the conceptual designs of the HTR-MODUL and the MHTGR, respectively. This concept utilizes physical characteristics to arrive at a design wherein safety goals are achieved through the inherent response characteristics of the system. Considered to be a safe, efficient, environmentally acceptable and economic heat source for the cogeneration of electricity and process heat, the modular HTGR has been developed over the past two decades. Major developments include the construction and operation of the HTR-10 in China and the HTTR in Japan, with technology development efforts associated with the construction of the HTR-PM in China, the PBMR project in South Africa, the GT-MHR jointly developed by the U.S and Russia, and other design efforts [3]. Through the operation history and safety demonstration tests of two modular HTGR prototype plants, namely, the HTR-10 and the HTTR, inherent safety features incorporated in modular HTGRs were demonstrated and the performance of various systems are continuously being improved.

In 2000, the GIF signatory countries launched the Gen IV Roadmap Project which led to the selection of six Gen IV nuclear energy systems among which included the VHTR. This selection was based on an assessment of the potential of the concepts to meet the Gen IV criteria. Based upon the modular HTGR designs of the 1980's, the reference VHTR concept has a graphite-moderated and helium-cooled core with either prismatic or pebble fuel, with an outlet temperature in excess of 1000°C. A coolant outlet temperature range of 700-950°C can be achieved with existing materials in a first phase while long-term operation at higher temperatures requires qualification of new materials in a second phase. A large market of process heat applications requires less than 850°C outlet temperature and can be demonstrated during the first phase. During a second phase, the outlet temperature could be raised beyond 950°C to further enhance thermal efficiency and expand the application of nuclear energy to additional process heat markets.

3. Level of ongoing safety-related research and development

The early work of the RSWG focused largely on identification of high-level safety goals, articulation of a cohesive safety philosophy, and discussion of design principles, attributes and characteristics that may help to ensure optimal safety of Gen IV nuclear systems. In 2008, the RSWG published its recommendations on these and related topics in a report entitled Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems [4]. Within this document, the RSWG achieved a consensus regarding some of the safety-related attributes and characteristics to be achieved by Gen IV nuclear systems.

More recently, the RSWG has focused primarily on an integrated framework for assessing risk and safety issues for use throughout the Gen IV technology development cycle. In 2011, the RSWG published a second report entitled An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems [5]. This methodology consists of five distinct analytical tools and stages, which are structured around the last one, the probabilistic safety assessment. The tools/stages are the following:

- qualitative safety requirements/characteristic review;
- phenomena identification and ranking table;
- objective provision tree;
- deterministic and phenomenological analyses;
- probabilistic safety assessments.

Furthermore, the GIF RSWG developed the ISAM guideline[6] to clarify and enhance the application of the methodology. The ISAM is applicable for the safety assessment and performance evaluation of the VHTR. Some safety-related topics of the VHTR can be further clarified and elaborated (see section 4). The VHTR safety principles covering design, construction, commissioning, operation, and decommissioning are the same as those for current NPPs, which are mainly LWRs, but higher level of safety can be achieved by the VHTR. An IAEA Technical Working Group on Gas-cooled Reactors is developing Safety Design Criteria for High Temperature Gas-Cooled Reactors [7]. The SDC will provide designers with specific guidance for guaranteeing plant safety, particularly when designing for very high temperature (greater than 950°C) operation.

For example, the strategy of defense-in-depth will be applied to manage different categories of postulated accident scenarios. Most importantly, for VHTR the confinement of radioactive materials is achieved with a combination of high-performance coated fuel particles (CFPs), the graphite matrix, the primary coolant boundary and the Vented Low Pressure Containment (VLPC) or confinement. Nevertheless, there are great differences in the safety philosophy between the VHTR and the LWR, as the former strongly relies on the inherent safety features such as low power density, a large heat capacity of ceramic core internals, a core with a large height-to-diameter ratio, and CFPs acting as the primary barrier to radionuclide release. This combination of attributes allows the VHTR to reject decay heat passively and at a sufficient rate such that severe core damage cannot occur. This has been demonstrated in AVR [8], HTTR [9], and HTR-10 [10].

Sufficient experience has been accumulated during the design of both pebble bed reactors (HTR-MODUL, HTR-10, PBMR, and HTR-PM) and prismatic block type reactors, (HTTR, MHTGR and GT-MHR), which in turn were built upon the technologies demonstrated in the AVR, Peach Bottom, THTR-300, and FSV. Future VHTR designs will incorporate these lessons and practices.

In addition to these efforts, research and development continues in several countries focusing on key technical issues [11]. These include: the development and qualification of high-performance fuel, coupling of the reactor and the process heat plant, validation and verification of system analysis and computational fluid dynamics models, development and qualification of high temperature materials such as graphite and metallic alloys, and industrial process heat applications including hydrogen production. Under the GIF framework, research activities for the VHTR system are mainly performed under four projects: Hydrogen Production (HP), Fuel and Fuel Cycle (FFC), high temperature Materials (MAT), and Computational Methods, Validation, and Benchmarking (CMVB).

4. Achievement of fundamental safety function

4.1 TRISO Fuel

The coated particle fuel form is the primary barrier to fission product release. The coatings, in particular the silicon carbide layer, effectively retain the bulk of the radionuclide inventory under all anticipated operating and accident conditions. The fuel performance envelope was largely established in the first generation HTGR programs in the US and Germany which specified limits on power density, temperature, fluence, burnup, and packing fraction. Recent fuel fabrication and qualification efforts have further improved the performance with respect to particle failures and radionuclide release. The performance of TRISO fuel is demonstrated through qualification testing (irradiations and post-accident heating) under conditions that fully envelope those of the reactor under all anticipated operating and accident conditions. It is sufficient then to design and operate the core such that the fuel remains within this qualification envelope, i.e. retains its structural integrity under conditions of elevated temperature and chemical attack. The following sections describe the reactor design attributes that ensure that the fuel performance envelope is not breached.

4.2 Reactivity control

4.2.1 Control system

As with most modern nuclear plants, the VHTR features a negative temperature coefficient of reactivity which serves to suppress fission power in the case of accidents, even for inadvertent reactivity insertion. The large thermal inertia of the graphite absorbs the additional heat, even in the hottest fuel element if commonly accepted safety guidelines are followed in the design. The margin to failure in the TRISO particles is several hundreds of degrees above the temperatures achieved in the core.

The first protective action of a VHTR to any fault signal is to trip the helium circulator. The increase in core temperature ceases the fission reaction (due to strongly negative temperature coefficient) and the higher temperatures increase the rate of decay heat removal by conduction and radiation processes even in the absence of helium coolant. The balance between decay heat generation and rejection is such that peak fuel temperatures remain well under that which could cause severe core damage, even without auxiliary cooling. The reactor automatically reverts to a safe and stable low power state even in the worst case events where all station power is lost, the primary coolant system is depressurized, and the control rods somehow fail to insert.

In conformance to current safety requirements and design criteria, at least two independent reactivity control systems are planned for all VHTR designs. In pebble bed reactors, online refueling allows operation with minimal excess reactivity but separate control rod (CR) and small absorber sphere (SAS) systems are independently activated in the event of a fault condition. For prismatic reactors, burnable poisons in the fuel reduce the excess reactivity while two separate control rod systems would be deployed for short term reactivity control and shutdown. Nonetheless, under accident scenarios subcriticality can be achieved solely through the negative temperature feedback resulting in a relatively benign core heat-up transient that does not challenge the integrity of fuel, graphite or components. This has been demonstrated practically as tripping the primary coolant circulators was a routine procedure for powering down the AVR. .

Control rod ejection is not a credible scenario. The rods are suspended in block channels with some bypass helium flow moving past them in a downward direction to keep them cool. The structure of the control rod drive assembly and the limited space above the reactor pressure vessel prevents the excessive vertical displacement of the rod. This design feature was previously proposed for the General Atomics MHGTR and was accepted by the US Nuclear

Regulatory Commission [11]. In some designs, the control rods are not rigid but comprise of loosely connected vertical segments. This type of rod is not easily propelled as a unit from the channel. Inadvertent and uncontrolled rod withdrawal (usually a bank of rods) is a Reactivity Insertion Event that must be analyzed as part of a license application. Simulations of rapid control rod withdrawal indicate that the negative temperature feedback is strong enough to prevent significant temperature excursions. The fast temperature response of small size fuel particle is another advantage to countermeasure the fast reactivity accident. Nonetheless, depending upon the rate of withdrawal, either a high power or high temperature fault signal would be generated. The worst case of either of these scenarios would need to be simulated to show that core safety limits are not challenged.

Similarly, seismic disruption of the core must be assessed and shown not to result in excessive fuel temperatures. In pebble bed reactors, shaking can lead to a mild compaction of the core and a reactivity insertion (due to both an increase in fuel density and to what is effectively a withdrawal of control rods (the bed drops down). Mechanically, the control rod channels are designed so that motion of the surrounding reflector blocks does not prevent rod insertion. This must be demonstrated to the satisfaction of the designer. The SAS used in some pebble bed designs are not susceptible to 'jamming'.

4.2.2 Risk of Re-criticality

In Reactivity Insertion Accidents (RIAs), such as what may follow a CR system malfunction along with a loss of forced cooling, the core will quickly become subcritical. If control elements are not inserted, the core would eventually become re-critical as the core temperature and xenon concentrations decrease. Depending upon the size and power output of the core, re-criticality may occur after tens of hours or even days. If control rods are not inserted, the long-term reactor behavior is characterized by low magnitude oscillations in the core power (a few percent of nominal full power) reflecting well-understood dynamics resulting from changes in fuel temperature and xenon concentration. Eventually, the core assumes a low power state (< 1% nominal power) with the rate of fission heat production equal to the rate of decay heat rejection to the ultimate heat sink. The core temperature remains well below that at which any significant quantities of radionuclides could be released from the fuel. As discussed in the next section, this is a very low probability event and the large thermal inertia of the core allows considerable time for operators to take action.

4.3 Air and Water Ingress

It is the reality of industrial plants that seals leak and pipes break such can also be expected in the VHTR. In the event of an unplanned depressurization of the primary system, air can enter the core through breaks in pipes and other primary system components. Hypothetical large breaks (in coolant pipes) are considered as Design Extension Conditions for VHTRs, but smaller breaks occurring in instrument lines and other parts of the pressure boundary can be expected at some time in the life of the plant. Even if a large break is postulated, one must remember that the helium coolant is not considered critical for plant safety. In a circulator trip or a severe fault condition that results in a loss of convective cooling, heat is can be safely removed by conduction and radiation. The High Temperature Test Facility under construction at Oregon State University was commissioned to investigate the behavior of the core and surrounding atmosphere after such a large break as well as numerous small break scenarios. Using data from these and other experiments, numerous simulations that have indicated no significant core damage resulting from a loss of pressure and flow, and the extremely low probability of a massive cross-duct rupture, the equivalent of a large break LOCA demonstration, is considered to be a design extension condition.

Nuclear grade graphite does not burn but will oxidize if oxygen is available continuously and in sufficient concentration. A continuous leak of oxygen into the primary loop may result in oxidation of graphite structures closest to the location of the break, usually the lower plenum. In smaller quantities, the oxygen is consumed by the graphite closest to the break. Modular VHTRs are designed so as to prevent large amounts of air and water from entering the primary loop (relief valves, pipe diameters, steam generator design). The inventory of helium in the primary circuit is large compared to that of the air within the VLPC and the gas released from the core would be vented and filtered in the event of a severe depressurization accident. The reactor building itself is designed to take advantage of the density difference between air and helium to minimize the air concentration near the break.

While nuclear grade graphite does not burn (i.e. rapid and energetic combustion), it does oxidize at elevated temperatures and with a continuous supply of fresh air [12]. Consequently, the most exposed graphite components may eventually lose structural integrity. The oxidation process itself generates volatile combustible compounds such as carbon monoxide. The reactor should be designed to minimize these phenomena. The reactivity effect of air ingress is negative due to neutron capture in nitrogen.

For plants driving a steam-based (Rankine cycle) power conversion system, a rupture of a steam generator tube will result in water entering the core. The additional moderating effect of the water will initially increase reactivity but the moisture ingress will be limited by opening of passive pressure relief valves and ‘dumping’ (draining) of the steam generator secondary side. In typical HTR designs, the primary loop pressure is upwards of 7 MPa while the secondary side is as high at 25 MPa, which would quickly drop below the primary side pressure in the event of a rupture. The reactor control systems are generally designed to compensate for the reactivity insertion which, theoretically, can be as high a few thousand pcm if a large amount of the secondary side coolant is allowed to enter (no pressure relieve or SG drainage). The core will heat up in response to the reactivity insertion but, as with other loss of convection scenarios, the core will go subcritical while mitigating measures are taken.

The major risk associated with a steam generator tube rupture is that the water can penetrate the graphite elements and ‘wash off’ radiological species previously trapped near the surface. This adds to the circulating inventory.

The probability of a relief valve opening is low because the helium inventory control system can suppress the increase in pressure and the helium inventory and purification system can mitigate the consequences of minor water ingress events. For multiple tube ruptures that do cause an opening of a primary pressure relief valve, the valve is assumed to fail ‘open’ after a few actuation cycles, thus depressurizing the primary loop and releasing this activity into the building. This is considered a bounding radiological release event and is analyzed accordingly.

Both AVR and FSV experienced massive water ingress events. In the AVR, the steam generator was located directly above the core, arguably making such a scenario inevitable. In both cases, the reactors entered extended outages to remove the water and affect repairs. In neither case was the reactor prevented from resuming operation. Improvements to the design of modular VHTRs were made from analyzing these events.

A full analysis of the behavior of nuclear grade graphite subjected to air and water ingress, including a retrospective view of the Windscale accident, is documented in an Idaho National Laboratory report [12] and in an IAEA TECDOC published in 2016.

4.4 Decay heat removal

4.4.1 Thermal inertia and grace period

The VHTR core has a low power density, a high heat capacity, and a slender core shape (large height-to-diameter ratio). The transients resulting from Loss of Forced Cooling (LOFC) events develop and occur over tens or hundreds of hours; a very long 'grace period' relative to LWRs. If forced cooling of the VHTR core is lost, decay heat is transferred naturally via conduction and radiation from the core, through the outer reflector and reactor pressure vessel (RPV), and on to the ultimate heat sink. For VHTR, the removal of decay heat does not require active cooling by the helium. Indeed, the first protective action of a VHTR responding to accident is to stop helium circulator as mentioned earlier. Modular HTGRs and VHTRs employ a Reactor Cavity Cooling System (RCCS) to absorb heat from the RPV outer surface and carry it via natural circulation of air or water to external cooling panels. This prevents the vessel from exceeding its designed temperature limit. In the event of RCCS failure (as is assumed in some Design Extension Conditions (DECs)), the reactor building and surrounding soil become ultimate heat sink which does not absorb heat as well as the working RCCS. The RPV will not fail but nonetheless may sustain damage and eventually require repair or replacement. For example, the steels proposed for near-term deployment in HTGRs (SA508/533) are allowed by the American Society of Mechanical Engineers (ASME) code to exceed the allowed limit for long-term operation (371°C) for limited periods of time[14]. Thus, an HTGR may be subjected to two or three unprotected LOFC events before requiring replacement. Failure of the RCCS may also lead to the overheating (spalling) of the surface of concrete reactor building, warranting extended shutdown for repairs.

Even with a non-functioning RCCS, massive fuel failure does not occur. If left unchecked, however, the higher fuel temperatures in the core after a DLOFC may drive sufficient release of fission products from some parts of the core such that, if not captured by the filters, dose limits to workers and at the site boundary may be challenged. Fortunately, the core heat-up and cool-down would take place over many days during which mitigating actions can take place. After the core cools, defueling and repair operations could commence.

The RCCS is designed mainly for the protection of the vessel and the surrounding concrete. HTR vendors have been very supportive of RCCS experiments and associated modeling efforts in order to characterize the uncertainties. The RCCS is considered an important safety and investment protection system.

4.4.2 Diversification, active and passive systems

The reactor core is cooled by convective transport of heat to the steam generator (or intermediate heat exchanger) during normal power and shutdown operations (loss of load, maintenance, or accident conditions). The RCCS is an additional system that can operate without station power to protect plant structures from overheating. As mentioned above, the fuel integrity is nonetheless ensured in case of failure of all cooling systems, including the RCCS.

4.5 Confinement of radioactive materials

Multiple barriers preventing fission product release are designed into the VHTR, including the CFPs, the graphite matrix, the primary pressure boundary, and the VLPC. Of these, the particle coatings are the most effective because they stay intact under the anticipated range of temperature, power, and burnup conditions. There is a low rate of diffusion of certain species (e.g. cesium, silver, and europium) out of the particles at normal operating temperatures, but this increases as temperatures approach and exceed 1600°C as the particle failure rate increases. Even with anticipated particle fabrication defects, the small amounts released do not pose a radiological concern, except possibly for workers engaged in maintenance and repair of the cooler primary loop components upon which the species (especially silver) are likely to be deposited [14]. A dedicated helium purification system is designed to keep the circulating activity at extremely low levels.

Indeed, the 1600°C 'limit' is not a 'cliff-edge' s) for a control rod to be completely expelled from the core.occupied by the rod drive assemblies) for a control rod to completebut is a rule of thumb based upon historical data, in which the temperature at which the release rate of fission products from UO2-based TRISO fuel is observed to increase considerably. The fuel needs to exceed this temperature for extended periods of time (tens to hundreds of hours) and in significant volumes of the core before significant core damage with radionuclide release would be expected to occur. Some recent tests on modern TRISO fuel forms (UCO specifically) indicate the elevated release rates are not observed below 1800°C at burnups well in excess of 10% FIMA[16]. Failure rates for UCO TRISO have been demonstrated to be lower than plant designer requirements at burnups as his as 19% FIMA (in the US AGR tests.). Based on these experiences, other TRISO fuel design and manufacture should yield improved results.

Given the low release rates, the surrounding graphite matrix retains much of this release so that the activity in the helium is very low, even including the circulating graphite dust that accumulates during pebble bed reactor operation. 'Designed-to-Fail' tests of TRISO fuel have been performed in the AGR series of experiments and have generated data confirming the additional retentive capability of the graphite. Outside of the primary coolant boundary, the radiological releases are further controlled by the Vented Low-Pressure Containment System (VLCS) which captures the very low levels of fission products that might leak from the primary coolant loop. Actual radiation doses to workers and the environment are below actionable levels.

Under the relatively severe Depressurized LOFC condition (which entails a breach in the primary coolant boundary) and a heatup of the core. The elevated temperatures will drive certain species out of the graphite matrix (noble gases in particular but other mobile species as well). The helium and the circulating activity it contains can be discharged into the reactor building (where more of it will be deposited). If the building pressure limit is exceeded or if the VLPC is purposefully vented, the concentrations of radionuclides are low enough to be released into the environment without any filtration. No off-site measures are needed to protect the public. When the pressure balance is achieved between the VLPC and the environment, the negative-pressure ventilation will be re-established, recovering the filtration function.

5. Management of design extension conditions

The strategy of defense-in-depth is adopted to deal with different categories of accidents. The first three levels of the strategy reflect the principles of prevention, detection, and control of Design

Basis Accidents (DBAs). As there is no risk of severe core damage in the VHTR plant even under the most extreme scenarios, “prevention” and “mitigation” of Design Extension Conditions (DECs) focuses on building response and the VLPC system. Large-break induced depressurization without backup cooling, with or without failure of the RCCS, will lead to elevated fission product release rates through the graphite matrix (but not core melt in the commonly accepted sense). There are two phases to such an event: 1) ‘blowdown’ and 2) core heatup/conduction cooldown. Blowdown of the primary loop releases in the circulating inventory into the reactor building. Some of the inventory will be retained by the building itself (intelligent design of the building can inhibit releases) but much of it will be released by venting the building to the atmosphere but not through the filters and they would not be able to manage the flow rate. Circulating inventory will be low enough such that site boundary levels remain below actionable levels [15] after this first ‘puff’. After depressurization (“blowdown”), the vents are closed and filtering of the building atmosphere resumes for the heatup/cooldown phase. Due to the elevated temperatures attained in this phase, fission products are driven out of the graphite at a greater rate. Some will be released to atmosphere through filtered vents but the overall dose rate at the site boundary remains below levels at which measures would be needed to protect harm to the public.

Unmitigated DEC may result in irreparable structural damage to the plant. A loss of forced cooling accompanied by a failure of the shutdown cooling system can lead to overheating of the reactor pressure vessel. The ASME code case for SA508/533 (LWR steel) permits a limited number of excursions in temperatures above 700°F/370°C to a maximum of 1000°F/540°C for limited periods of time (1000 hours) under Level A, B, C, and D Service Limits [14]. Depending upon the length and severity of the heatup phase, the vessel may be able to return to service.

The temperature attained in the vessel (and the concrete walls of the reactor cavity) is sensitive to the operation of the Reactor Cavity (or Vessel) Cooling System. If the RCCS fails, the ultimate heat sink becomes the concrete walls of the reactor building and surrounding soil. In this Design Extension Condition, irreparable damage to the RPV and concrete is likely to occur even though the fuel remains largely intact and radiological releases will still be limited.

A long grace period (up to days) allows considerably flexibility and for on-site accident management. External measures can be considered such as injection of nitrogen (to inert the building atmosphere) and backup (e.g. diesel) power replacement or refueling. The nitrogen would displace any air in the building that might enter the RPV through the break and degrade the graphite structures through oxidation. Backup generators would be used to power shutdown (auxiliary) blowers to facilitate decay heat removal and prevent structural damage caused by overheating. A site-specific management guide for DEC will be included as part of the licensing of the plant.

Future VHTR plants will be built to comply with regulations related to airplane crashes, through below-grade emplacement and suitable above-grade building structures.

5.1 Practical elimination of hazardous releases

For a VHTR, the source term is mainly from the DLOFC and air or water ingress scenarios which have a very low probability of occurrence. Even so, the radioactivity release to the environment should not exceed the permissible value due to the design features described above. Even in the DEC case (described above) of a loss of forced cooling and failure of the RCCS, the increase in fuel temperature is limited in both prismatic and pebble bed HTRs. In the prismatic fuel MHTGR [16], the peak fuel temperature was computed to be only 75K higher than in the comparable case with the operational RCCS. The peak RPV and concrete temperatures were computed to be 750°C and 650°C, respectively, significantly higher than achieved with a functioning RCCS. Likewise, in the pebble-fueled 200 MWt HTR Modul [17], the peak fuel temperature was computed to be only 100K higher than the case of the operational RCCS. There are no conceivable accident scenarios, even DECs, that lead to significant releases of radionuclides to the environment.

The total release of radionuclides in cases of all other postulated accidents is limited and is much lower than any LWR severe accident. Moreover, the release probability is even lower than the above-mentioned bounding scenarios. Accordingly, the target of practical elimination of hazardous releases can be achieved by the VHTR.

6. Safety of the fuel cycle

6.1 Type of fuel

TRISO-type CFPs using a UO2 or UCO kernel and a SiC layer are currently proposed for VHTR fuels. More advanced kernel and coating technologies, such as a C/ZrC/SiC sandwich layers, are being investigated. The flexible VHTR fuel kernel compositions allow a diversity of fertile and fissile materials to support a variety of fuel cycle strategies including high burnup and thorium cycles, LWR spent fuel reduction, and weapons material destruction. Some of these fuels from laboratory scale production have already been partly qualified.

Because coated particle fuel is a key element for the safety performance of the VHTR, production-scale fuel is subject to very stringent QA measures and qualification tests including irradiation testing to high burn-ups followed by the simulation of temperature excursions during accidents during which the fuel must retain its inventory.

6.2 Management of waste (quantity, quality)

Nuclear waste of the VHTR mainly includes spent fuels and irradiated graphite. The EU has completed a research project named CARBOWASTE focusing on the treatment, decontamination, disposal or recycling of irradiated graphite. At laboratory scale, promising decontamination results have been obtained by fragmentation of the graphite followed by heat-up and exposure to a carrier gas with low oxidation potential [19].

Fragmentation of pebble fuel elements and exposure of the coated particles can be expected to occur to a limited extent. Broken pebbles are separated from undamaged pebbles and sent to storage and eventual disposal. Fuel kernels can be reprocessed using available technologies. However, under the current market conditions, the ‘once-through’ utilization of VHTR fuel is more economical than available reprocessing options. The retention characteristics of the fuel also make it a robust waste form.

There are, nonetheless, some technical issues to which special attention should be paid. These include but are not limited to: a very large volume of irradiated graphite, inventory and release of the 14C isotope, possibly additional steps for spent fuel reprocessing compared with LWR, and large volume of irradiated materials during the plant decommissioning phase. These do not pose critical safety concerns and thus can be addressed as VHTRs are deployed.

6.3 Radiation protection

Internal radiation levels in the VHTR plant are generally lower than those in LWR because of the low circulating activity in the coolant, mainly in the form of graphite dust on which some radionuclides may be adsorbed or otherwise carried. If the hydraulic system is not properly designed, dust deposits in the primary system may create hot spots which hinder in-service inspection and may pose a worker hazard during replacement and repair of components inside the primary circuit, e.g., blower, CR, SAS and Steam Generator (SG). This is potentially more of a concern for a pebble bed system in which dust is created during the movement of pebbles through the system during operation and because the online refueling of the PBR obviates the need for regular maintenance and refueling outages. Although dust did not inhibit the successful operation of AVR and FSV, increased understanding and mitigation of dust is being gained through ongoing R&D and confirmed through operation. Similarly, operational experience will improve the ability to manage the transport and release of tritium and 14C.

7. Other risks

7.1 7.1 Chemical risk

As a VHTR can be connected to a chemical plant, the dynamic coupling between the reactor and the chemical plant needs to be considered and addressed. A process heat plant must be collocated in relative proximity with the nuclear plant to minimize thermal losses during transport of the working fluid. The VHTR is uniquely suited for this. The potential radiological releases are so low that contamination of the process heat facility is negligible. The inert coolant and the large thermal inertia of the VHTR core effectively decouple the reactor dynamics from the process heat system dynamics. Nonetheless, the transport of reactive species from the chemical plant to the nuclear plant cannot be ruled out and has been the subject of investigation.

It is also important to verify the impact of accidents in the process heat end-user plant onto the nuclear reactor (e.g. explosion, fire, shock waves) and to ensure that the responsible safety authorities carry out combined safety assessments. An analysis of thermal fluid behavior of an HTGR coupled to various process heat plants was performed by General Atomics as part of HTGR training provided to the US NRC [18].

7.2 7.2 Radiation protection

Per the issue of dust-born contamination described above, operational feedback will be important in determining the magnitude of the radiological risk and for developing mitigation strategies. This is not considered a critical issue. A similar approach has been used in the development and deployment of Boiling Water Reactor technology. The radiological hazard faced by VHTR plant personnel is anticipated to be much lower than that faced by BWR personnel.

8. Others

Other technical issues not mentioned above still need to be defined.

9. Summary of progress needed

Currently, research activities proposed and performed by the four VHTR project management boards reflect VHTR R&D deemed essential to near-term deployment and specifically to licensing. The most recent summary together with an outlook is provided in [11]. Further research may focus on the following aspects: Qualification of alloys that can withstand even higher coolant outlet temperatures, development of graphite coatings that can better withstand oxidation, development of new kernel coating technology, development of process heat applications and power conversion systems, and experimental validation of models and computer codes used for plant simulation. Operational experience feedback from HTGR test reactors (HTR-10 and HTTR) and demonstration plants (HTR-PM) will be very valuable for safety demonstrations and for the optimization of performance and reliability of systems and components.

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