Very High Temperature Reactor (VHTR) Risk and Safety Assessment White Paper

J. Tong, F. Li, M. Fütterer, H. Gougar, C. Sink and D. Petti

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The purpose of this paper is to investigate the feasibility of applying the Integrated Safety Assessment Methodology (ISAM) to the VHTR system. Developed by the Generation-IV International Forum (GIF) Risk and Safety Working Group (RSWG), ISAM consists of a set of tools to aid in safety design of advanced reactor concepts. This paper summarizes the ISAM tools, provides an overview of VHTR technology, and the preliminary assessment of ISAM tools in application to VHTR safety design.

1. Short recall about the assessment methodology

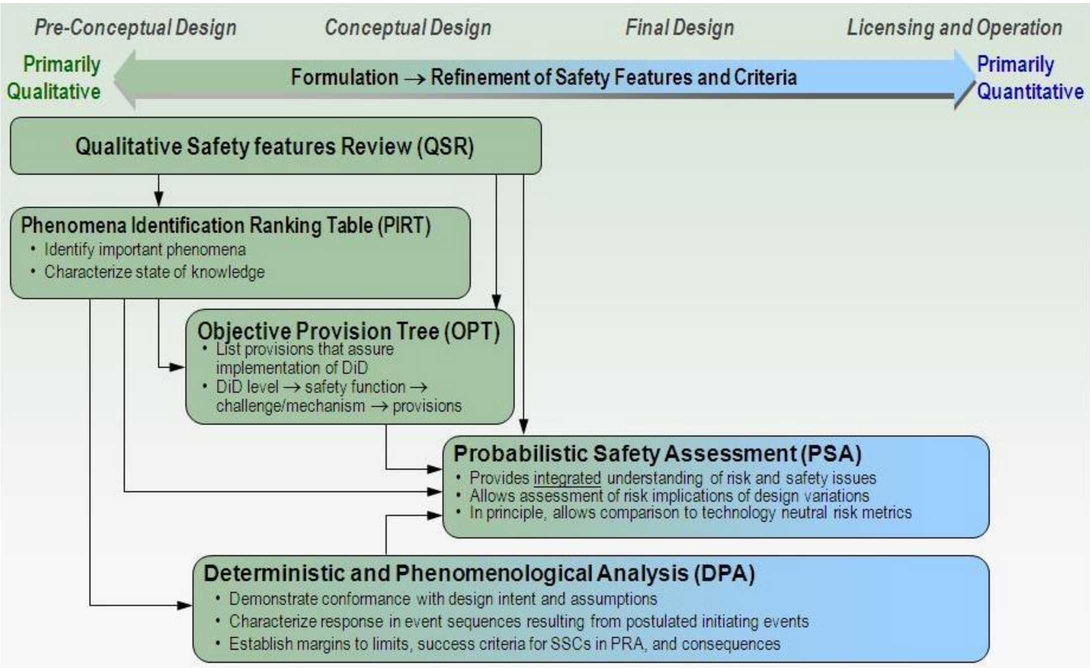
The RSWG has developed a methodology, called the Integrated Safety Assessment Methodology (ISAM), for use throughout the Gen IV technology development cycle. The risk and safety assessment white paper with respect to the Very High Temperature Reactor (VHTR) pilots the ISAM methodology in order to demonstrate its validity and feasibility. ISAM methodology will draw the feedback for further improvement in some aspects.

The ISAM consists of five distinct analytical tools (Ref. 1) which are intended to support achievement of safety that is “built-in” rather than “added on” by influencing the direction of the concept and design development. These tools are the followings:

- Qualitative Safety Features Review (QSR)
- Phenomena Identification and Ranking Table (PIRT)
- Objective Provision Tree (OPT)
- Deterministic and Phenomenological Analyses (DPA)
- Probabilistic Safety Analysis (PSA)

Figure 1 shows the overall task flow of the ISAM and indicates which tools are intended for use in each phase of Generation IV system technology development.

Figure 1: GIF Integrated Safety Assessment Methodology (ISAM) Task Flow



Each of the analysis tools that are part of the ISAM is briefly described here:

- ***Qualitative Safety Features Review (QSR)***

The Qualitative Safety Features Review (QSR) is a new tool that provides a systematic means of ensuring and documenting that the evolving Gen IV system concept of design incorporates the desirable safety-related attributes and characteristics that are identified and discussed in the RSWG's first report entitled, "Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems", as well as in other references (e.g. the INPRO Safety methodology). Although this element of the ISAM is offered as an optional step, it is believed that the QSR provides a useful means of shaping designers' approaches to their work to help ensure that safety truly is "built-in, not added-onto" since the early phases of the design of Gen IV systems. Using a structured template to guide the process, concept and design developers are prompted to consider, for their respective systems, how the attributes of "defense in depth", high safety reliability, minimization of sensitivity to human error, and other important safety characteristics might best be incorporated. The QSR also serves as a useful preparatory step for other elements of the ISAM by promoting a richer understanding of the developing design in terms of safety issues or vulnerabilities that will be analyzed in more depth in those other analytical steps.

- ***Phenomena Identification and Ranking Table (PIRT)***

The Phenomena Identification and Ranking Table (PIRT) is a technique that has been widely applied in both nuclear and non-nuclear applications. As applied to Gen IV nuclear systems, the PIRT is used to identify a spectrum of safety-related phenomena or scenarios that could affect those systems, and to rank order those phenomena or scenarios on the basis of their importance (often related to their potential consequences), and the state of knowledge related to associated phenomena (i.e. sources and magnitudes of phenomenological uncertainties).

The method relies heavily on expert elicitation, but provides a discipline for identifying those issues that will undergo more rigorous analysis using the other tools that comprise the ISAM. As such, the PIRT forms an input to both the Objective Provision Tree (OPT) analyses, and the Probabilistic Safety Analysis (PSA). The PIRT is particularly helpful in defining the course of accident sequences, and defining safety system success criteria. The PIRT is essential in helping to identify areas in which additional research may be helpful to reduce uncertainties.

- ***Objective Provision Tree (OPT)***

The Objective Provision Tree (OPT) is a relatively new analytical tool that is enjoying increasing use. The International Atomic Energy Agency (IAEA) has been a particularly influential developer and proponent of this analysis tool. The purpose of the OPT is to ensure and document the provision of essential "lines of protection" to ensure successful prevention, control or mitigation of phenomena that could potentially damage the nuclear system. There is a natural interface between the OPT and the PIRT in that the PIRT identifies phenomena and issues that could potentially be important to safety, and the OPT focuses on identifying design provisions intended to prevent, control, or mitigate the consequences of those phenomena.

- ***Deterministic and Phenomenological Analyses (DPA)***

Classical deterministic and phenomenological analyses, including thermal-hydraulic analyses, computational fluid dynamics (CFD) analyses, reactor physics analyses, accident simulation, materials behavior models, structural analysis models, and other similar analysis tools collectively constitute a vital part of the overall Gen IV ISAM. These traditional deterministic analyses will be used as needed to understand a wide range of safety issues that guide concept and design development, and will form inputs into the PSA. These analyses typically involve the use of familiar deterministic safety analysis codes. It is anticipated that DPA will be used from the late portion of the pre-conceptual design phase through ultimate licensing and regulation of the Generation IV system.

- ***Probabilistic Safety Analysis (PSA)***

Probabilistic Safety Analysis (PSA) is a widely accepted, integrative method that is rigorous, disciplined, and systematic, and therefore it forms the principal basis of the ISAM. PSA can only be meaningfully applied to a design that has reached a sufficient level of maturity and detail. Thus, PSA addresses licensing and regulatory concerns and is performed, and iterated with a beginning in the late pre-conceptual design phase, and continuing through to the final design stages. In fact, as the concept of the “living PSA” (one that is frequently updated to reflect changes in design, system configuration, and operating procedures) is becoming increasingly accepted, the RSWG advocates the idea of applying PSA at the earliest practical point in the design process, and continuing to use it as a key decision tool throughout the life of the plant or system. Although the other elements of the ISAM have significant value as stand-alone analysis methods, their value is enhanced by the fact that they serve as useful tools in helping to prepare for and to shape the PSA once the design has matured to a point where the PSA can be successfully applied.

It is intended that each tool be used to answer specific kinds of safety-related questions in differing degrees of detail, and at different stages of design maturity. As indicated within the Ref. 1 it is envisioned that the ISAM and its tools will be used in three principal ways:

- A use throughout the concept development and design phases with insights derived from the ISAM serving to influence the course of the design evolution.
- A punctual implementation of selected elements of the methodology which are applied at various points throughout the design evolution to yield an objective understanding of risk contributors, safety margins, effectiveness of safety-related design provisions, sources and impacts of uncertainties, and other safety-related issues that are important to decision makers.
- An application in the late stages of design maturity to measure the level of safety and risk associated with a given design relative to safety objectives or licensing criteria.

2. Overview of Technology

The Very High Temperature Reactor (VHTR) is a helium-cooled, TRISO coated particle fuel, graphite-moderated, graphite-reflected, metallic-vessel reactor plant with the capability for the generation of electricity using a turbine cycle, with possible co-generation of process steam and high-temperature process heat for chemical process and hydrogen co-production. Various versions of the VHTR are under development in several countries that are members of the Generation IV International Forum (GIF), including the People’s Republic of China, France, Japan, the Russian Federation, Republic of South Africa, Republic of Korea, and the United States of America. The major VHTR design options can be categorized as follows:

- Prismatic versus pebble fuel
- Direct versus indirect power conversion cycles
- Water versus air cooled Reactor Cavity Cooling System (RCCS)
- Filtered confinement versus low leakage containment
- Underground versus above-ground nuclear islands

The two VHTR basic design concepts are the Prismatic VHTR and the Pebble Bed VHTR. Technology summaries for each vendor-proposed design option can be found in the reference 3 and the respective references.

2.1 Prismatic VHTR

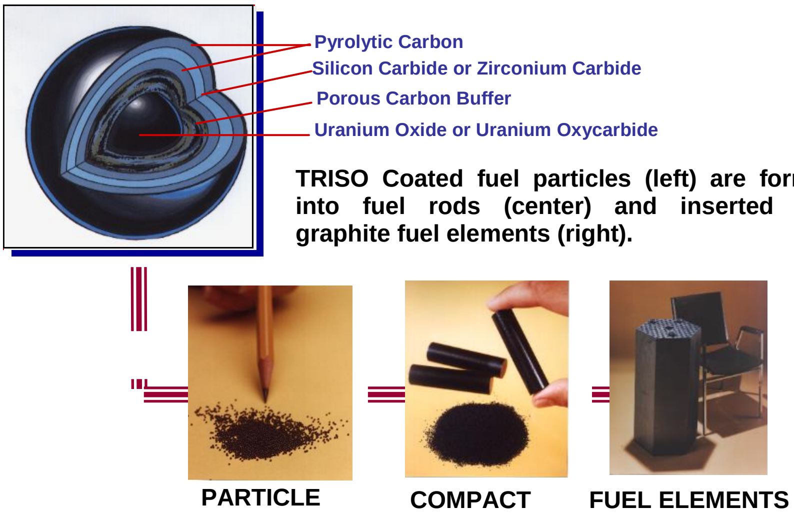
There are currently five concepts for the prismatic VHTR under consideration by different GIF nations. The first two of the following have the generic features of low-enriched uranium (LEU) and plutonium-fuelled block-type cores. Except for the second concept discussed below, the prismatic VHTRs are being designed assuming the initial use of a once-through LEU fuel cycle.

- **U.S.**: The General Atomics (GA) prismatic-fuel, direct or indirect cycle, air-cooled RCCS, filtered confinement Gas-Turbine Modular Helium-cooled Reactor (GT-MHR) or Modular High-Temperature Gas Reactor (MHTGR). The GT-MHR and MHTGR are 350-600 MW-thermal reactors with options for cogeneration of electricity and process heat. In the frame of the US DOE NGNP project different designs have been assessed. A cooperative agreement was put in place with the GA team to perform conceptual-level design work for the NGNP project based on the prismatic block reactor concept. The GA team prepared a Conceptual Design Report containing a summary-level description of their 350-MW(t) Steam-Cycle Modular Helium Reactor (SC-MHR) Demonstration Plant. The SC-MHR has a nominal reactor outlet helium temperature of 725°C and is designed to produce steam at 585°C and 16.5 MPa at the exit of the SG. Additionally, AREVA Nuclear Power completed a pebble bed reactor design assessment based on the existing HTR Modul design. From 2008 to 2013, the NGNP Project engaged the NRC in a series of pre-licensing interactions to develop a licensing framework for the modular HTGR. The intent of these interactions was to obtain further clarification from the NRC Staff regarding various issues associated with licensing the modular HTGR based on regulations developed primarily for LWRs. The NGNP project submitted 11 white papers to NRC on key licensing topics and engaged in 18 public meetings with the staff.
- **Russia**: In cooperation with GA and the U.S. Department of Energy (DOE), National Nuclear Security Administration (NNSA), the Experimental Design Bureau of Mechanical Engineering (OKBM) in Nizhniy-Novgorod with partners at the Kurchatov Institute (KI) and the A.A. Bochvar All-Russian Scientific Research Institute for Inorganic Materials (VNIINM) in Moscow is designing a Russian version of the GA GT-MHR to dispose of excess weapon-grade plutonium; however, OKBM is also analyzing alternative fuel cycles for the Russian GT-MHR. The deployment of the Russian GT-MHR is subject to DOE/NNSA joint funding to complete necessary research and development.
- **France**: The AREVA prismatic-fuel, indirect cycle, water-cooled RCCS, filtered confinement Modular High-Temperature Reactor (HTR) (designated ANTARES) which benefited for the R&D partnership with other EURATOM participants in the High Temperature Reactor-Technology Network (HTR-TN). The ANTARES Modular HTR is also envisioned to be a 600 MW-thermal cogeneration plant; the pre-conceptual design of ANTARES and the corresponding R&D have been completed, with in particular irradiation of TRISO fuel in the ATR reactor. R&D is pursued on plate-type IHX, in the frame of the European ARCHER project.
- **Japan**: The Japan Atomic Energy Agency (JAEA) continues development work that begun under the former Japan Atomic Energy Research Institute (JAERI) on the Gas Turbine High Temperature Reactor 300 for Cogeneration (GTHTR300C), which will scale up the technology from the JAEA 30 MW-thermal High Temperature Test Reactor (HTTR) into a 600 MW-thermal configuration that shares design features with both the GA GT-MHR and the AREVA ANTARES Modular HTR except for being coupled to a horizontal turbine-generator for electricity production; however, deployment of the GTHTR300C is not envisioned until after 2030.
- **South Korea**: The Korea Atomic Energy Research Institute (KAERI) is pursuing the Nuclear Hydrogen Development and Demonstration (NHDD) Project; the NHDD reactor is to be limited to 200 MW-thermal (based on the maximum reactor vessel diameter, 6.5 m, that can be fabricated in-country) with no decision yet made as to fuel/core type (pebble bed or prismatic).

The baseline fuel design uses LEU as TRISO-coated particle fuel in a once-through fuel cycle; the Russian version of the GT-MHR will use excess weapon plutonium as TRISO-coated fuel particles with the addition of erbium containing 167Er to provide a neutron poison with a thermal neutron capture resonance to affect a negative moderator temperature coefficient of reactivity. The TRISO-coated particle fuel (see Figure 2.1) has a small-diameter (nominally 200-500 μm) spherical ceramic fuel kernel of either uranium oxide or uranium oxycarbide, or mixed oxides of other actinides. The kernel is coated with four coating layers consisting sequentially of low-density porous pyrocarbon, an inner high density pyrocarbon (IPyC), silicon carbide (SiC), and an outer high density pyrocarbon (OPyC). The coatings on the fuel particles serve as the primary containment preventing the release of fission products, and plant configurations and operating conditions are being designed appropriately to limit fuel temperatures during both normal operations and accident conditions so as to preclude the release of fission products. The coated particles are loaded into fuel compacts (sticks) held together by graphitized carbon. The fuel compacts are loaded into holes in hexagonal prismatic block fuel elements. Fuel elements are stacked in the reactor core with fissile and neutron burnable poison loadings tailored so that the power distribution is peaked toward the top of the core where the inlet cooling gas has the lowest temperature and the power density is lowest in the bottom of the core where the temperature of the outlet coolant is highest. The fuel and burnable poison loading patterns are set to keep the peak fuel temperature below the limit for normal operation, which is 1250ºC for TRISO-coated fuel particles with SiC coatings.

Spent fuel is retained in cooled storage containers that are embedded underground and located adjacent to the reactor cavity. Prismatic spent fuel, which is unloaded from the core during periodic refueling shutdowns, can be tracked remotely by cameras viewing the serial numbers on the fuel elements during handling and storage operations. Since each fuel element is loaded with less than 4 kilograms of LEU, the plutonium content at full burn-up (~120 GWD/T) will be small (~60-70 grams) and isotopically degraded compared to weapon-grade plutonium.

Figure 2.1: Illustration of Coated Particle Fuel in the Prismatic Fuel Element



The current concepts for the energy utilization from the prismatic VHTRs are based on:

- direct Brayton cycle for electricity generation,
- indirect steam generation for process heat and/or electricity generation,
- indirect heat transfer to process heat user (e.g. Hydrogen production).

2.2 Pebble Bed VHTR

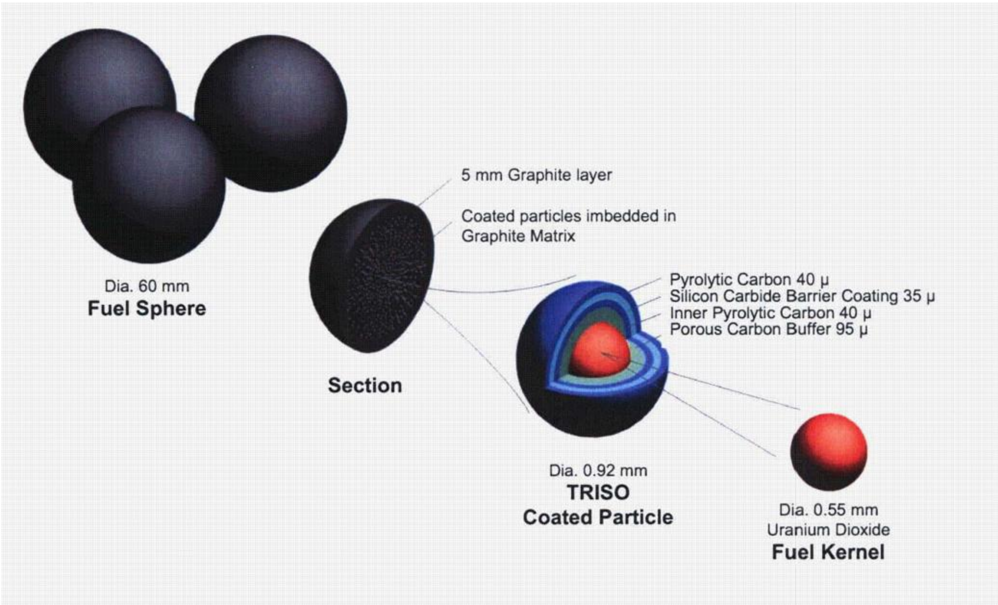
There are two national programs for a pebble bed VHTR.

- **South Africa**: The Westinghouse and PBMR (Pty) Ltd. pebble-fuel, water-cooled RCCS, filtered confinement Pebble Bed Modular Reactor (PBMR) that has been designed as a 400 MW-thermal direct Brayton cycle plant as an NGNP candidate was changed to a 400-500 MW-thermal steam plant utilizing two 200-250 MW-thermal reactors. The core for the 400 MW-thermal PBMR was to be annular with an inner cylindrical graphite reflector; the 200-250 MW-thermal core design would be cylindrical. Due to the financing dilemma, PBMR demonstration project was suspended indefinitely since 2009.

- **People's Republic of China (PRC)**: A consortium with the China Huaneng Group, the China Nuclear Engineering & Construction Group (CNEC) and Tsinghua University's Institute of Nuclear and New Energy Technology (INET) is developing and preparing near-term construction of the 250 MW-thermal, steam-cycle High-Temperature Reactor-Pebble-bed Module (HTR-PM); the HTR-PM demonstration plant, which builds on the success of the Tsinghua University's HTR-10 test reactor, is envisioned to be constructed in two module units producing 500 MW-thermal and 200 MW-electric. The HTR-PM core is to be cylindrical. HTR-PM demonstration plant began the construction in Dec, 2012. It is expected to be commissioned in 2017.

The pebble bed reactors share the same TRISO features as the prismatic VHTRs but have less excess reactivity due to on-load refueling. The LEU fuel for the pebble bed VHTRs is to be TRISO-coated particles compacted in small spheres as illustrated in Figure 2.2.

Figure 2.2: Illustration of Coated Particle Fuel in Pebble Fuel Element



The pebble fuel is not tracked individually by serial number as in the prismatic core, but elements are counted, characterized, and checked following each of multiple re-circulations until they achieve the target burn-up based on radioactivity measurements. Following several passes of each pebble through the core during on-line pebble recirculation, when pebble radioactivity indicates sufficient burn-up, the pebble is transferred to a storage container with a record kept of the number of pebbles transferred. Once pebble spent fuel is in the storage container, radiation monitoring is used to quantify by inference the amount of spent fuel present since, with no more than 0.12 grams of plutonium per pebble, it would take several tens of thousands of pebbles (or several metric tons by total mass and cubic meters by volume) to be diverted to constitute the basis for recovering a significant quantity of plutonium since a fresh PBMR pebble only contains 9 grams of LEU and a fresh HTR-PM pebble 7 grams. Further, at a burn-up around 90 GWD/T, the plutonium isotopic composition in the pebble spent fuel is degraded significantly from that of weapon-grade plutonium.

The 400 MW-thermal Brayton-cycle PBMR concept has the partially embedded reactor with the horizontal gas-turbine to the right of the reactor vessel and the associated spent fuel storage locations below-grade to the left of the reactor vessel. The 250 MW-thermal steam-cycle PRC HTR-PM has arranged the steam generator below to the left of the reactor vessel.

The key design parameters for each concept (both prismatic and pebble bed) are presented in Table 2.1.

2.3 Safety design principles

The safety basis for all the VHTR is to design the reactor to achieve passive safety to avoid release of fission products under all conditions of normal operation and accidents including most of the beyond design basis events. This passive safety aspect of the design should make the VHTR less vulnerable to a significant risk of "radiological sabotage" through malevolent acts.

Major safety functions of VHTR are usually realized in the following ways.

(1) Reactivity control

The inherent safety feature of VHTR usually plays significant and powerful role in the reactivity control function. In most of the cases, VHTR reactor can be inherently shutdown automatically due to the sufficient reactivity feedback and temporary Xenon-135 effect. However two diverse active safety shutdown systems are equipped according to most of the countries' regulation, by the means of control rods and other measures e.g. absorber spheres, in order to keep the stable cold shutdown state.

(2) Residual heat removal

The residual removal path based on passive system/feature is usually the primary path to remove heat, so it has to meet the safety related requirements during accidents. In order to increase the reliability and also the plant operation flexibility, a second diverse/redundant cavity or liner cooling system which is non-safety related can be found in the recent VHTR designs, so that more than one way can be used to remove the residual heat.

(3) Confinement of radioactivity

The use of gaseous coolant imposes the big challenge of containing the gaseous medium within the volume. As a result of the fact that TRISO fuel provides the excellent capability of fission product retention, VHTR designs are evaluating the choice of filtered confinement versus low leakage containment, because most of the radioactivity will be contained in the fuel during the normal operations and accident conditions as well as most of the beyond design basis accidents.

Key safety characteristics of VHTR can be summarized based on the current studies of the above VHTR programs.

(1) Unique safety features are presented due to the fission product retention of the coated particles.

(2) Longer grace periods during accidents are paid by large graphite masses.

(3) The RPV vessel concept (integrated or non-integrated) influences the possible accident scenarios of VHTR.

(4) Confinement concept is often used by VHTR, but it will need to cope with large breaks.

(5) Retention/filtering of graphite dust is one of the commonly concerned issues and needs be improved.

(6) Air and water ingress are main safety challenges.

2.4 Pending nuclear safety issues to VHTR

RSWG and VHTR SSC have cooperated to draw some key safety issues that VHTR shall be considered for further R&D programs based on the state of the art knowledge. These issues can be further discussed and expanded with the help of ISAM applications.

(1) Fission product confinement & Radiological source term

- Containment versus confinement: Need for containment (i.e., a pressure resistant barrier around the primary circuit) or confinement?
- The safety function achievement: the optimum share between the different barriers ensuring radio-nuclide retention (particle coating, primary circuit, confinement/containment)?

- Ability of TRISO coated fuel particles to contain fission products as a function of the fuel temperature.

(2) Consideration of severe core damage.

- Prevention versus mitigation of the consequences?
- Justification of severe core damage practical elimination.

(3) Credit for passive safety features: Importance of passive features in the safety demonstration?

(4) Stochastic behavior of the Pebble Bed Reactors (PBR): how to deal with margins associated with the inherent behavior of PBR (not dealt with)

(5) Combined safety assessment for modular reactors and for other site facilities

- Impact of several modular plants on the same site on safety?
- Impact from safety point of view of facilities linked with the nuclear plant?

(6) Materials codes and standards: Availability of codes and standards?

3. Overview of the Safety Architecture's characteristics and performances (*System's SSC/PMB in charge; RSWG in support*)

In this section, the feasibility analysis of applying the ISAM approach in case of VHTR will be discussed. The reference VHTR design is taken from the HTR-PM program. Although the ISAM final goal is to identify the provisions which, for each safety function, participate to the control, the management and the mitigation (if requested) of all the plausible abnormal situations, it is understood that the iteration process cannot be avoided, and the results shall be updated progressively. So in this white paper, the discussion focus is concentrated on the methodological feasibility rather than the result itself.

3.1 Qualitative Safety Features Review (QSR)

QSR is defined as the identification of safety related recommendations or foreseen characteristics helpful for a standard qualitative safety assessment. The basic idea is to provide the designer with a check list summarizing the good practices and recommendations which can be useful to verify that the design details are coherent with the recommendations which are available from different sources (Regulators, IAEA, RSWG), and applicable to the future nuclear systems. This ISAM element is offered as an optional step.

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Table 2.1: HTR or VHTR Major Reactor Design Parameters

Major Reactor Parameters	AREVA Modular HTR	General Atomics GT-MHR	Westinghouse & PBMR (Pty) Ltd PBMR	Huaneng Group & CNEC/INET HTR-PM	JAEA GTHTR300C	OKBM GT-MHR	KAERI NHDD
Thermal Power (MW-th)	600	600	400	250	600	600	200
Thermal Efficiency (%) in Electricity Generation	~45	~48	44.8 at a core coolant T_{outlet} of ~850ºC	40	~50 (inferred)	~48	None, H_2 production
Primary Coolant	Helium	Helium	Helium	Helium	Helium	Helium	Helium
Moderator	Graphite	Graphite	Graphitized Carbon with Graphite Reflector	Graphitized Carbon with Graphite Reflector	Graphite	Graphite	Graphite or Graphitized Carbon with Reflector
Power Density (MW/m^3)	~6.3 (inferred)	6.3	4.78	~3.22	5.4	6.3	2.27-3.0 pebble, 5.68 prismatic
Fuel Materials	$LEUO_2$ TRISO-coated particles	$UC_{0.5}O_{1.5}$ TRISO-coated particles; $LEUC_{0.5}O_{1.5}$ (19.8%) fissile and $U_{Nat}C_{0.5}O_{1.5}$ fertile	$LEUO_2$ TRISO-coated particles	$LEUO_2$ TRISO-coated particles	$LEUO_2$ TRISO-coated particles	$PuO_{1.8}$, LEUCO or mixed uranium-plutonium oxide (MOX)	$LEUO_2$ TRISO-coated particles
Fuel Element Type	Prismatic	Prismatic	Pebble Bed	Pebble Bed	Prismatic	Prismatic	Prismatic
Core Inlet Temperature/Pressure (ºC/MPa)	~400/~6.0	490/7.07	500/~9.0 (pressure inferred)	250/~7.0	586-663/6.9 (electrical production) & 594/5.1 (H_2 production)	490/7.07	490/~7.0
Core Outlet Temperature/Pressure (ºC/MPa)	or 850/7.0	850/7.0	900/~9.0	750/~7.0	850-950/6.9 (electrical production) 950/5.1 (hydrogen production)	850/7.0	950/~7.0
Neutron Energy Spectrum	Thermal peaking just below 0.3eV	Thermal peaking just below 0.3eV	Thermal peaking just below 0.3eV	Thermal peaking just below 0.3eV	Thermal peaking just below 0.3 eV	Thermal peaking just below 0.3 eV	Thermal peaking just below 0.3eV

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A practical problem during the application is that there are vast of sources (regulators, IAEA, RSWG, other international organizations and so on) which might be useful to compile the check list. The feasibility practice which intends to put these documents all in order, e.g. establishing the clear and consummate understanding about the differences and relationships among these documents, was found difficult, if we start from the very beginning. So this step is alternated by taking Appendix 2 of reference 1 – QSR Tables of Technical recommendations as the trial check list, including Table A2.1a – Generic & Technology neutral recommendations, Table A 2.1b – Detailed & Technology neutral recommendations, and Table A 2.1c – Detailed & Technology neutral recommendations applicable to a given safety function (decay heat removal). The trial check list can be refined successively along with the development.

For Class 1 technical recommendations and foreseen characteristics and features listed in Table A 2.1a, the items can be successfully applied for VHTR design since they are applicable to all the future reactors.

For Class 2 Detailed technical recommendations and foreseen characteristics and features in Table A 2.1b, most of the items can also be successfully applied for VHTR design since they are developed in order to cover all the future reactors. Issues which may need further discussion are summarized below.

1) Postulated Initiating Events (PIE) (see check list Item 1.2 and 1.4 under Class 2) shall be characterized by the pre-defined categories, which is not only useful for the PIE identification, occurrence minimization and consequence mitigation, but also necessary for the setup of the safety acceptance criteria in the future. It can be observed that different terminology frameworks are used by different regulators. For example, the term "design extension conditions" in Reference 1 corresponds to the scope of both the limiting events and beyond design plant states in European Fast Reactor (EFR) technology. While US NGNP project proposes the Licensing Basis Events (LBE) terminology framework to characterize the PIE categories. The difference, conflict or confusion might arise due to the inconsistent terminology. Hence, it would be a necessary and useful recommendation to have the item dedicated for PIE category definition during the Class 2 development.

2) The top-down logical approach pushes on the development of each level in the parallel way. The balance design among the defense levels haven't been well arranged to the appropriate position within the approach up to now. In fact, the safety feature of Gen IV reactors gives the possibility to adjust the "depth" of each level to achieve the optimum share between the different barriers. In case of VHTR, the fission product retention capability of TRISO fuel can increase the "depth" of the first, second and third defense levels inherently. The parallel development cannot consider the kind of benefit and impact in an integrated manner. In addition, the suggestion that accident management with more or less core degradation

should be considered at the fourth level of defence in depth, as recommended by ISAM (appendix 2 QSR table), is found worthy of further deliberation. The main reason that core degradation is tightly tied up with the severe accidents of Light Water Reactors(LWR), is the inherent fact that LWR cannot lose the coolant. In case of loss of coolant, LWRs will evolve to the stage of core degradation in a short time, and must actuate the emergent response quickly. It is a very rational requirement to consider core degradation in the fourth level of defense-in-depth for LWRs. However, VHTR reactors may not be necessary to persist in the same requirement on the fourth level of defence-in-depth as LWRs, because the inherent and reactor-specific safety features, e.g. enhanced TRISO barrier and longer grace time for accident transients for VHTR, can support the fourth level of defence-in-depth implicitly.

3) The ways of “confinement of radioactive materials” of Gen IV technologies might be quite different from each other. Therefore, some specific features, e.g. to prevent the containment failure (Item 4.4.1) and to conceive the containment provisions in order to keep the containment capabilities compatible with the objectives (item 4.6), might not be appropriate for VHTR in order to avoid major release of radioactive materials to the environment. There have been a lot of arguments on confinement/containment. The white paper deems it very necessary to recall the traditional concept of containment function in order to resolve the issue. Typically, the LWR containment undertakes not only the function to contain the radioactive materials, but also the additional important functions like external missile protection, supporting and protection of the component inside the containment, residual heat removal, and so on. Concerning the purpose of radioactive material retention, confinement solution can be the reasonable technology to meet the safety requirements, whilst the necessity and fulfilment of the other containment functions need to be justified. For example, the building structure should be designed to stand the most severe transient considered.

According to the ISAM approach, the intention of the Class 3 check list is to be technical neutral but specific for a given function. However, it is found that the intention might be difficult to be met strictly. For example, in Item 3.3, abnormal, accidental and severe accident conditions shall be identified in order to complete the condition list under each of the PIE category. Obviously, different reactor design may have different PIEs. So the newly developed list to describe these conditions in Class 3 phase will definitely have the reactor technology’s characteristics. For instance, the example item 3.3.2.2 in Table A2.1c «Primary pump faults (pump seizure and shaft failure) is not suitable for VHTR. VHTR has to identify its own abnormal, accidental and severe conditions.

From the subsequent trial of qualitative safety features review to the existing system and option characteristics and features, the interim conclusion for the qualitative safety feature recommendations which might have significant impact to the VHTR’s design are summarized as below.

1) The requirement or recommendation of the full range of conditions is one of significant features for Gen IV energy systems. The design basis for Gen IV energy systems should cover the full range of safety significant conditions. The historical notion of a single bounding design basis accident must be replaced by a “spectrum” of possible accidents that, while of low probability, represents with high confidence the range of physical events that could conceivably challenge the plant.

2) Specific efforts should be made for demonstrating the “practical elimination” of initiators, sequences or phenomena associated with the extremely low residual risk, including the avoidance of any cliff edge effect by design, to demonstrate the aptness of the concept for mitigating severe accidents and so on.

In general, the feasibility study of this White Paper supports the conclusion that the top-down functional approach to obtain the QSR list is effective and useful for VHTR.

Table 3.2.1: PIRT process illustration of VHTR trial

Steps	Step output	Notes
Step 1. Define the issue	The issue was defined as identifying the priority R&D issues related to the safety features specific to VHTR (HTR-PM) upon the DLOFC accident	
Step 2. Define the specific objectives	The specific objectives were defined as identifying phenomena and factors having a significant impact on DLOFC accident mitigation	
Step 3. Obtain database information	Database information sources be utilized included the preliminary safety analysis report, preliminary probabilistic safety analysis report, system design technical specifications, specific analysis reports for selected accident sequences and so on	HTR-PM has finished the preliminary design. There are relatively mature documents which can be collected and asked for
Step 4. Define Hardware and Scenario	(1) hardware: reactor, reactor protection system, safety shutdown systems (control rods and absorber spheres), residual heat removal system, confinement (ventilation system and filter) (2) scenario: a DLOFC accident	It is suggested to discuss and confirm the accident mitigation philosophy first, so that all the related hardware can be enumerated. OPT and PSA is found to be of help
Step 5. Establish the Figure of Merit	The Figure of Merit was defined as the source term released to the environment	
Step 6. Identify Phenomena	Accident analyst, PSA analyst, Source term analyst formed the expert panel to identify the phenomena	
Step 7. Importance Ranking	The importance ranking was rated. The ranking scales defined in Table 3.2.2 were applied.	Only one consensus scale was given
Step 8 Knowledge Assessment	The knowledge level was assessed only at the current time points by considering whether we have sufficient knowledge to simulate precisely the identified individual phenomena and state variables. The knowledge level ranking scales defined in Table 3.2.3 were applied.	Only one consensus scale was given

3.2 Phenomena Identification and Ranking Table (PIRT)

As applied to Gen IV nuclear systems, the PIRT is used to identify:

- a spectrum of safety-related phenomena or scenarios that could affect those systems, and to rank order those phenomena or scenarios on the basis of their importance (often related to their potential consequences), and
- the state of knowledge related to the associated phenomena (i.e. sources and amplitude of phenomenological uncertainties).

Formally the full elaboration of a PIRT in the ISAM approach has to go through 10 steps. All these steps are important for the final assessment. Nevertheless, priority is given to some of them, including step 1 (Define the issue), step 2 (Define the specific objectives), step 3 (Obtain database information), step 6 (Identify phenomena) and step 7 (Importance Ranking).

A VHTR PIRT trial is built on the depressurized loss-of-forced-cooling (DLOFC) accident in HTR-PM project, summarized as Table 3.2.1

Table 3.2.2: Importance ranking scales

Rank	Definition	Application Outcomes
High (H)	Phenomenon has controlling impact on figure-of-merit	Experimental simulation and analytical modeling with a high degree of accuracy is critical
Medium (M)	Phenomenon has moderate impact on figure-of-merit	Experimental simulation and/or analytical modeling with a moderate degree of accuracy is required
Low (L)	Phenomenon has low impact on figure-of-merit	Modeling must be present only to preserve functional dependencies.
Insignificant (I)	Phenomenon has no, or insignificant impact on figure-of-merit	Modeling must be present only if functional dependencies are required

Table 3.2.3: Knowledge level ranking scales

Rank	Meaning
4	Fully known, small uncertainty
3	Known, moderate uncertainty
2	Partially known, large uncertainty
1	Very limited knowledge, uncertainty cannot be characterized

Some lessons can be drawn from the application.

1) Expert panel formulation is found to be the most significant issue for the VHTR PIRT implementation, because all the PIRT works need the experts' deliberation, including the selection of Figure of Merit (FOM), the specification of scope, the phenomena identification and the phenomena ranking. The usefulness of PIRT technique lies in the panel's ability to identify and rank, relatively quickly and cost effectively, all of the phenomena in a complex reactor system. Independence of panel members and depth of their expertise on the subject matter are significant contributors to the success of the deliberations. Obviously, all the Gen IV systems will face the same issue due to the lack of experts, especially for the innovative features.

2) PIRT process has the significant brainstorming characteristics. It is suggested that OPT and PSA can be started during the PIRT process, e.g. together with the identification step, because these two tools can produce plentiful insights for the PIRT panel, and the insights are provided in a more systematic way. Hence the necessary of a good design management framework seems to be important, for which can make the effective and successful iteration among PIRT/OPT/DPA/PSA.

3) It is suggested that the expert panel builds up consensus breakdown criteria during the Identification of phenomena, characteristics and variables. Many phenomena, characteristics and variables can be expressed in different levels or ways. For example in the passive residual heat removal system, the water natural circulation capability and variance for the water cooling wall can be described as one key characteristic, but can also be divided into many factors: various resistance coefficients, hydraulic correlations, residual heat distribution and so on.

4) Importance Rankings have less overall impact on the conclusions of the PIRT. It may be due to the fact that we, being the honest researchers, can never or hesitate to declare that we have known everything perfectly. Another reason may be led by the insufficient knowledge to find some quantitative impacts to the FOM. To set up a set of interpretations to the "controlled", "moderate" and "low" scales according to the reactor technology characteristics might be helpful. In this application, although we attached more importance to the benefit from the identification step than the ranking step during this application, the

ranking still shows some inherent features of VHTR. For example, the importance ranking of passive residual heat removal system is set to L-Low, because failure of passive residual heat removal system will not influence the maximum fuel temperature during the accidents. The fission product retention capability of coated particles will remain the same. Hence, the R&D concerning the passive residual heat removal system will not lead to the significant update to the radioactivity release to the environment.

It must be recalled that this pilot PIRT intends to demonstrate the applicability of ISAM methodology, and the above result may not be valid for all the HTR or VHTR reactors. For instance, decay heat removal systems could be more difficult to conceive for VHTRs with higher thermal power. As VHTRs work at elevated temperature, thermal stress and creep fatigue interactions are phenomena difficult to model and still need additional research programs. This could largely change the safety demonstration. For instance, certain accident scenarios could lead to high temperature that would affect reactor structures, core geometry and influence coolant distribution and reactivity characteristics. High temperature could as well influence the fuel behaviour.

3.3 Objective Provision Tree (OPT)

The Objective Provision Tree (OPT) is a practical tool which should be applied on line to design and /or to assess the safety architecture of innovative plants coherently with the Defense in Depth (DiD) philosophy. This is done through visual presentation and systematic inventorying of the plant's safety capabilities, i.e. the systematic identification of the provisions which participate to the safety missions' achievement. Its use requires a minimal knowledge of the installation characteristics, the phenomenology associated with the abnormal situations, and the associated risks.

The OPT method allows driving the design and its assessment by integrating, in a preliminary and macroscopic way, concerns of provisions' performances and reliability without waiting for the PSA models; it allows to make confident that the provisions required at each level of the defense in depth exist and are correctly implemented.

With the due background concerning both the process under examination and the phenomenology involved under installation's abnormal situations, the OPT method is a top-down method with a tree structure which:

- for each level of DiD (normally level 1 to 5),
- and for each safety objective/function (in general, control of reactivity, removal of heat from the fuel, and confinement of radioactive materials),

identify:

- the possible challenges to the safety functions
- the plausible mechanisms which can materialize these challenges
- the provided provision(s) to prevent or control the challenges/mechanisms,

All this is done by expressing this hierarchy structure in a tree form. The availability of the OPT can greatly help and simplify the preparation of the PSA.

A VHTR OPT application is built on the HTR-PM project, taking the 3rd level of defense as the example. The process is summarized as Table 3.3.1.

Generally, the HTR-PM application supported the ISAM's conclusion that for Gen IV, the application of OPT at a very early stage will allow safety to be built-in the design concepts. Lessons learned from the OPT application of VHTR can be put on the following two aspects.

One of the difficult steps of OPT application is to identify the possible mechanisms of the challenge. It is helpful to include a risk analyst into the OPT team. PSA application will use a set of well-proven systematic technologies to identify the postulated initiating events as complete as possible, e.g. Failure Mode and Effects Analysis (FMEA), master logic diagram, HAZOP and so on. And then the initiating events will be grouped into several groups for the following event tree analysis. The PSA process actually does the same thing for OPT mechanism identification. The grouped initiating events can be used to form the required mechanisms of challenge.

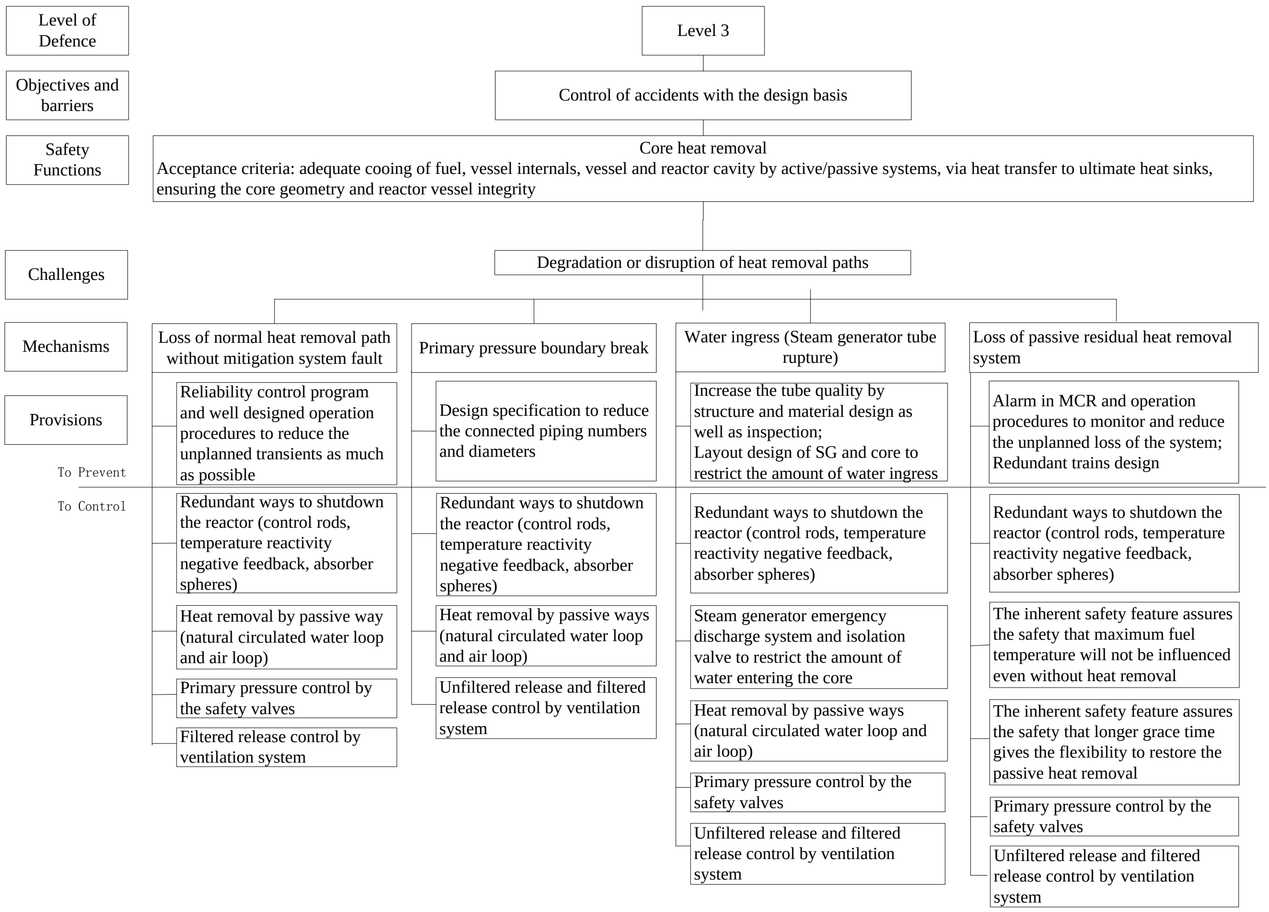
Table 3.2.4: Preliminary PIRT application result on DLOFC

System	Component	Phenomena/characteristics/State variables	IR	KL
Reactor	Reactor core	Power distribution	H	3
		Temperature reactivity coefficients (Fuel, Doppler, Coolant, …)	H	3
		Coolant temperature distribution through the pebble bed	M	3
		Coolant flow distribution through the pebble bed	H	2
		Fuel ball flow behaviour through the pebble bed	M	3
		Fuel temperature distribution spectrum throughout the pebble bed	H	3
		Thermal material property of fuel, coolant and graphite matrix	L	3
		Fission production diffusion coefficient in TRISO fuel	H	2
		Manufacture failure rate of coated particles	H	2
		Graphite dust deposition distribution in the core	H	1
		Nuclide deposition/combination/interaction behaviour against the graphite dust, coolant and the surface	H	1
Reactor Protection System	I&C	Actuation conditions for DLOFC related actions	L	4
Safety shutdown system	Control rods	Automatic and manual actuation conditions during DLOFC	L	4
	Absorber sphere	Only manual actuation condition	L	4
Residual heat removal system	Water cooling wall	Water natural circulation capability and variance	L	2
	Air cooling tower and the HX	Air natural circulation capability and variance	L	2
		environment temperature distribution	L	3
Confinement system	Ventilation blower	Capability (rate, operation temperature…)	M	3
		Actuation, stop and operation conditions	M	3
	Damper to the chimney	Actuation condition	M	3
		Reliability	M	3
		Capability (pressure limit, temperature…)	M	3
	Iodine filter	Filter efficiency	M	3
	Flow paths	Resistance	L	3

Table 3.3.1: VHTR OPT application illustration

Steps	Output	Notes
1. Team setting and defining the analyses scope	The objectives were defined as assessing the structure of safety architecture of the HTR-PM in a systematic, comprehensive and adequate manner based on the defense-in-depth philosophy.	
2. Data gathering	Database information sources be utilized included the preliminary safety analysis report, preliminary probabilistic safety analysis report, system design technical specifications, specific analysis reports for selected accident sequences and so on	The data can be shared with PIRT application and all other ISAM applications
3. Development of OPTs	OPTs were demonstrated by considering the residual heat removal safety function and the 3rd level of the defense-in-depth	It is quite similar with PSA steps such as identifying the initiating events and developing the event tree sequences
4. Documentation of the results	The developed OPT was illustrated in a tree structure (Figure 4.1)	

Figure 3.3.1: Example of OPT developed for HTR-PM safety function 2 (core heat removal) at level 3 of Defense-in-Depth



Developing the provisions is the other difficult task for OPT. The provisions are the measures to be implemented to prevent and/or control the mechanisms. During the implementation, the team had discussion on what are the prevention provisions and what are the control provisions. The team also felt uncertain about that whether the provisions about reactivity control and confinement shall be developed in the line of provision under the function of core heat removal. The final decision was made to develop the whole provision in order to give the clear picture about how we defend the challenge mechanism.

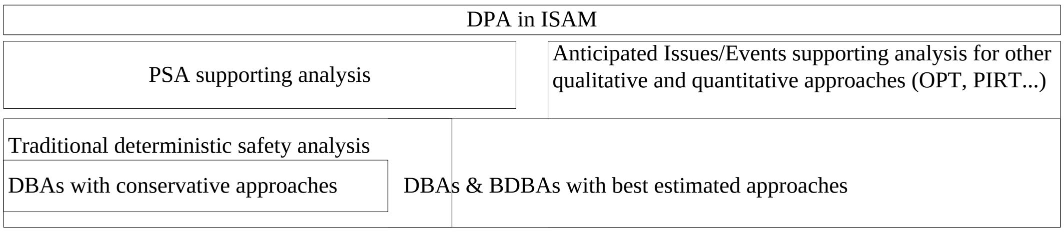
It is realized by the VHTR OPT team, from the application, that the current tree structure has the restriction to represent the multiple branches within the line of provisions. That is, not all the provisions in the same line shall be used all the time. For example, during the accidents of loss of normal heat removal path, normally the primary pressure will not increase to the set point of safety valve. Only the safety valves are challenged by the very rare occasions (additional failure occurs subsequently), filtered release control will be required. However, this graphical representation of the safety related design architecture will be very useful in the development of VHTR PSA models. In addition, the reliability data and expert judgments collected within this task will provide valuable input for PSA model quantifications.

3.4 Deterministic and Phenomenological Analyses (DPA)

Deterministic and phenomenological analyses, including thermal-hydraulic analyses, computational fluid dynamics (CFD) analyses, reactor physics analyses, accident simulation, materials behavior models, structural analysis models, and other similar analysis tools collectively constitute a vital part of the overall Gen IV ISAM.

It is clear, that at a later stage, and in particular during the licensing phase, a full scope of deterministic safety analysis may be needed to demonstrate that the plant as designed is capable of meeting any prescribed limits for radioactive releases and acceptable limits for potential radiation doses for each category of plant states. It is anticipated that people may draw an equal sign between DPA and the above mentioned deterministic safety analysis. It is interpreted by the VHTR application team that ISAM intends to give DPA an even broader functionality scope, since DPA will be used as needed to understand a wide range of safety issues that guide concept and design development, and form inputs into the PSA. Figure 3.4.1 illustrates the overview of DPA in ISAM.

Figure 3.4.1: DPA in ISAM



As shown in Figure 3.4.1, DPA serves much more than the traditional deterministic safety analysis, typically the Chapter 15 of safety analysis report (the inner box in the left bottom corner) and the Chapter 19 of safety analysis report which has been required by the recent licensing processes(most portion of the bottom boxes) . For an innovative and new design, the wider spectrum of postulated accidents must be investigated before the DBAs and BDBAs can be selected and determined in line with the characteristics of representativeness, envelope, likelihood etc. Meanwhile, the implementation of ISAM elements such as PSA, OPT, PIRT and so on require the extensive support of deterministic and phenomenological analysis either. And in order to meet the specific purposes of PSA, OPT, PIRT and other elements, DPA may not be necessary to adhere to the usual conservative approach for all the cases and best estimated approaches would be preferred.

Table 3.4.1 summarizes some of the DPA applications done in HTR-PM. Performing the DPA is a complex task, which places significant requirements on analysts. There is no generalized and typical procedure for DPA. The typical DPA flow chart provided in the ISAM consists of several steps. Depending on the analysis objective, these steps are not always required: some can be carried out in parallel; some can be neglected; some can be separated for details.

Technological challenges arising from the application include the balance of conservative/best estimate assumptions, the modeling of phenomena under exploration, lack of empirical data to validate the computation code and so on. It is demonstrated by HTR-PM application that they can be improved gradually by the iterative process between PSA and DPA. In particular the deterministic analysis, together with PSA, provides a means to evaluate the efficiency of the provisions defined in the different Layers of Protection to ensure that the relevant safety functions are respected and all important phenomena are considered.

3.5 Probabilistic Safety Assessment (PSA)

The PSA is recognized as an effective means to identify accident scenarios that could occur for a given design and, with the associated assessment tools, as effective means to quantitatively assess the weight of the uncertainties associated with various aspects of those scenarios. The recommended ISAM approach intends to elaborate the integrated and essential role of PSA in the development, design, licensing and operation of Gen IV nuclear systems.

PSA works with the associated assessment tools to check the whole consistency of the safety architecture versus criteria such as:

- An exhaustive defense, i.e.: the identification of the risks, which leans on the fundamental safety functions, should look for exhaustiveness.
- A graduated, progressive defense; without that, “short” sequences can happen for which, downstream from the initiator, the failure of a particular provision entails a major increase,

in terms of consequences, without any possibility of restoring safe conditions at an intermediate stage.

- A tolerant defense: no small deviation of the physical parameters outside, the expected ranges, can lead to severe consequences (i.e. rejection of "cliff edge effects").
- A forgiving defense, which guarantee the availability of a sufficient grace period and the possibility of recover during accidental situations.
- A balanced or homogeneous defense, i.e.: no sequence participates in an excessive and unbalanced manner to the global frequency of the damaged plant states.

Table 3.4.1: DPA cases in HTR-PM project

Purpose	Typical cases
The spectrum of design basis accidents are analyzed to support preliminary safety review	A control rod withdraws spuriously
	Loss of offsite power supply
	Loss of main feed water
	Large break Primary Pressure Boundary (DN65)
	Small break primary pressure boundary (DN10)
	Steam generator tube rupture
	……
Selected beyond design basis accidents are evaluated to support the severe accident management design and PSA success criteria development	ATWS without offsite power
	ATWS without main feed water
	Steam generator tube rupture overlapping with the failure of steam generator discharge system
	Loss of main feed water overlapping with the isolation failure of main blower damper
	Loss of passive residual heat removal system
	Air ingress by chimney effect
	……

By the VHTR PSA application, an internal event PSA during the power operation of HTR-PM project has been completed. It has been included into the primary safety analysis report submittals, which has been reviewed and approved by the Chinese regulator. The consensus between the designer and regulator has been reached that the extended scope PSAs such as low power and shutdown, internal flooding, internal fire and external events especially seismic events and so on will be completed progressively as the system design evolves.

HTR-PM PSA has achieved some successful experiences in the following application activities:

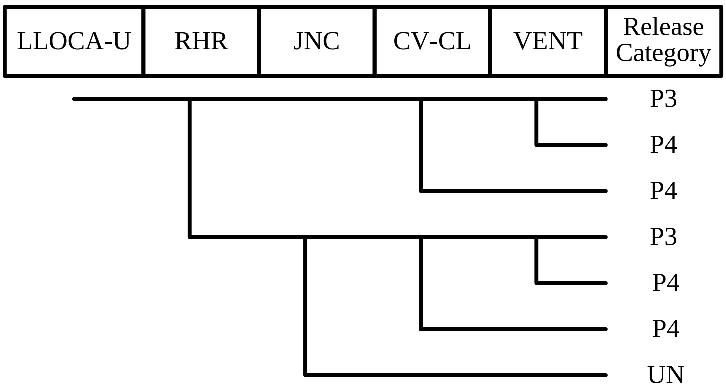
1) To demonstrate the current system design meets the safety objectives.
2) To support the HTR-PM plant operation states classification and the corresponding acceptance criteria establishment.
3) To support the selection of important event sequences which should be considered in the HTR-PM design as the representative beyond design accidents
4) To support the selection and determination of accident source term
5) To support the determination of emergency planning specification

The RSWG envisions that PSA should be used throughout the Gen IV system development cycle. However, the PSA value in the earlier stage of reactor technology development hasn't been proven and widely recognized. It is not easy to reach the consensus among the related communities (system designers, risk analysts, regulators and so on) that a full scope PSA with perfect quality is the final achievement, but not the essential basis for all the applications. Applications of the PSA results can be specifically associated with different stage of design development. Even in the earliest stages of conceptual design, simple PSA models can be useful in understanding how a design can be vulnerable to certain failures. As the design matures, based in part on an understanding of those vulnerabilities, the complexity of the PSA should also mature to yield more detailed insights into safety and risk issues associated with the design.

The importance and priority of the appropriate risk metrics determination for the candidate reactor technology is demonstrated by the VHTR PSA application. Given the diversity of different Gen IV reactor concepts, the traditional risk metrics that have been used widely for light water reactors will no longer be applicable or meaningful. In case of HTR-PM, the proposed metric is declared as: the cumulative frequency of all the accident sequences which may result in the consequence of individual effective dose exceeding 50mSv beyond? the site boundary shall be less than 1E-6 per reactor year. The proposed risk metric includes both the internal events and the external hazards. Both the designer and the regulator are involved to develop this metric, and this metric is finally issued as the Chinese regulator's official recommendation.

In order to implement such a risk metric evaluation, an integrated level 1 and 2 PSA framework has been used, since the source term information for each sequence must be obtained. Unique end states in terms of Release Category (RC) are defined for the event trees. Figure 3.5.1 illustrates the evolution process of HTR-PM PSA event sequences and the release categories. By using the integrated framework, all the accident sequences are categorized according to their release features. RC also provides the input to Level 3 PSA, because each RC is represented by a set of source terms.

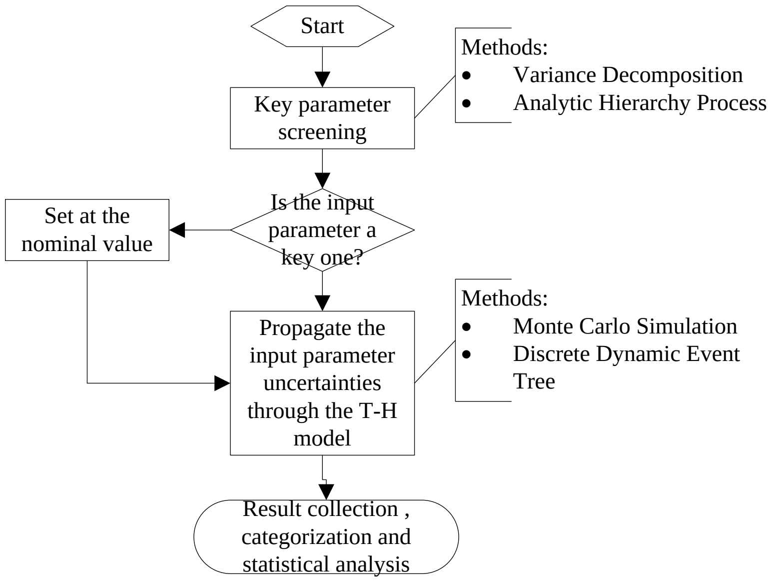
Figure 3.5.1 Evolution process of HTR-PM PSA event sequences and the release categories



Notes:
LLOCA-U: Large Break LOCA which cannot be isolated
RHR: Passive residual heat removal
JNC: Vessel support structure cooling
CV-CL: Confinement isolation valve reclosure
VENT: Filtered release control
P3: Release category for short-term unfiltered release and long-term filtered ventilation
P4: Release category for unfiltered release
UN: Bounding release category for unknown release phenomena

HTR-PM PSA experience indicates that most of the traditional PSA techniques can still be valid to VHTR reactors. Technical areas which may need special treatment are limited. One example is the methodology to estimate the reliability of passive system. Passive safety systems are commonly adopted by the Gen IV systems due to the advantages such as the simple structure, reduction of human interaction, reduction or avoidance of hardware failures and external power supply. When the reactor is in the abnormal state, passive safety system can operate on inherent safety characteristics of natural or physical principles. The elimination of hardware/active failures, which originally dominate the contribution of the safety system failure, brings forward the new question: what are the dominant factors for the passive system without hardware failures? It calls for the new technology development. The framework to evaluate the passive system reliability has been explored and shown in Figure 3.5.2. The proposed methodology is based on the Monte-Carlo simulation embedding the thermal-hydraulic (T-H) model which describes the physical process of the passive system under the different sets of possible parameter values. Since the large amount of running of the T-H model is very much time-consuming, some methods are considered to reduce the computation burden, e.g. key parameter screening.

Figure 3.5.2: Evolution framework for passive system reliability evaluation



Besides the passive system reliability, technical areas which may call for new techniques for Gen IV systems include the digital I&C system reliability as well as the human reliability evaluation under the digital environment and the longer grace time.

Another issue which will always cause the public interest is the reliability data quality issue for Gen IV systems. Gen IV systems will certainly have the inherent defect that It cannot have the operation experience of previous Generation reactors. However, it is illustrated by HTR-PM PSA application that the poor data problem is much less severe than people have conceived. In fact, the LWR experiences can still be valid for a large amount of components, e.g. the components in the auxiliary systems. Number of components which are completely new is very limited. Even for these new components, there are cues such as the common sense, engineering knowledge and expert opinion which can help to judge the data in the context of order of magnitude. Moreover, important safety functions require the redundant and diverse design. It reduces the impact of component data to the risk in another way. The sensitivity analysis of HTR-PM PSA reveals the fact that the exact component reliability data is relatively “less” sensitive to the risk. Very few components can directly cause the proportional change to the overall risk when the component reliability data varies. On the contrary, the unique design conception and functional features of the reactors can lead to more significant change to the overall risk. These features are usually influence the risk in a higher level, e.g. accident sequence level. For example, the choice of a passive residual heat removal system design versus the active design will have obvious impact to the risk result. In most of the cases, the rough quantitative information of the order of the magnitude is enough to support the decision making. As a matter of fact, what we are benefited from PSA is indeed the systematic way which PSA follows.

6. Current System Development Status

(N.B. The System’s SSC/PMB will be in charge of this section)

The design process follows an iterative stage: “design > assessment > feedback to the design” as they are planned by the designer. Safety concerns have to be taken into account in the “assessment” as early as feasible to answer the RSWG suggestion for a safety that will be “built in” rather “added on”.

The designer will indicate in this section, as far as feasible, timelines for key system development stages, if possible very high level GANTT chart, as well as the time schedule for the preparation of the different steps as they are identified within the §1 (PIRT > OPT > PSA).

7. Conclusions : System's Issues, Concerns and Benefits

(N.B. The System's SSC/PMB will be in charge of this section;RSWG will operate as support)

This section, will review the outstanding issues related to safety, the areas of known strength for the concept, and the future plans for improvement of identified weakness.

Conclusions on the implementation of ISAM based on results obtained shall be formulated. Indications are expected concerning the needs in terms of complementary R&D to be launched under the aegis of the GIF activities.

Based on the results obtained from the pilot implementation of ISAM to VHTR cases, can be summarized as follows.

1) The RSWG ISAM approach can be successfully applied to VHTR technologies in the general vision. The five distinct analytical tools (QSR, PIRT, OPT, DPA, PSA) can help to support achievement of safety that is “built-in” rather than “added on” by influencing the direction of the concept and design development.
2) ISAM methodology is not a standalone method. It is suggested that ISAM iterates frequently with the design and R&D work. Implementation of ISAM methodology can be more effective if there is a quite well defined design. However, it cannot be the realistic expectation that ISAM can highlight some specific issues that could challenge the safety of Gen IV reactors within every round of iteration. Although the ISAM proposed tools, i.e. QSR, PIRT, OPT, DPA and PSA, are not entirely new and have more or less been used in the design process. The pilot case prefers the lifetime control of the whole safety design with the help of ISAM tools' integration.
3) The basic idea of QSR is to provide the designer with a checklist summarizing the good practices and recommendations which can be useful to verify that the design details are coherent with the recommendations. The VHTR feasibility study supports the conclusion that the top-down functional approach to obtain the QSR list is effective and useful. It is suggested that the issue of considering the balance among the defense levels shall be mentioned in the approach. Because the safety feature of Gen IV reactors gives the possibility to adjust the “depth” of each level. In case of VHTR, the fission product retention capability of TRISO fuel can increase the “depth” of the first, second and third defense levels inherently. The parallel development of QSR approach cannot consider the kind of benefit and impact in the integrated way.
4) PIRT is used to identify safety-related phenomena or scenarios that could affect those systems, and to rank order those phenomena or scenarios on the basis of their importance and the associate state of knowledge, which helps identify the gaps in knowledge areas requiring additional research and data collection. Expert panel issue is found the most significant issue for the VHTR PIRT implementation, because all the PIRT works need the experts' deliberation. The usefulness of PIRT technique lies in the panel's ability. Since PIRT process has the significant brain storming characteristics, it is suggested that OPT and PSA can be started during the PIRT process, e.g. together with the identification step, because these two tools can produce plentiful insights for the PIRT panel, and the insights are provided in a more systematic way. Hence the necessary of a good design management framework seems to be important, for which can make the effective and successful iteration among PIRT/OPT/DPA/PSA.
5) OPT is applied on line to design and /or to assess the safety architecture of innovative plants coherently with the defense in depth philosophy. All this is done by expressing this hierarchy structure in a tree form. The availability of the OPT can greatly help and simplify the preparation of the PSA. As a matter of fact, OPT will benefit from PSA either, especially in the task of identifying the possible mechanisms of the challenges and developing the provisions. It will be a good arrangement to include risk analyst in the OPT team. The application supported the ISAM's conclusion that for Gen IV, the application of OPT at a very early stage will allow safety to be built-in the design concepts.

6) DPA in the ISAM approach is more than the traditional deterministic safety analysis. An even broader functionality scope has been assigned to DPA. DPA will be used as needed to understand a wide range of safety issues that guide concept and design development, and form inputs into the PSA. Technological challenges arising from the application include the balance of conservative/best estimate assumptions, the modeling of phenomena still under exploration, lack of empirical data to validate the computation code and so on. It is demonstrated by HTR-PM application that they can be improved gradually by the iterative process between PSA and DPA.

7) The recommended ISAM approach intends to elaborate the integrated and essential role of PSA in the development, design, licensing and operation of Gen IV nuclear systems. However, the PSA value in the earlier stage of reactor technology development hasn't been proven and widely recognized. The importance to reach the consensus that a full scope PSA with perfect quality is the final achievement, but not the essential basis for all the applications, shall be emphasized during ISAM implementation. Applications of the PSA results can be specifically associated with different stage of design development. Even in the earliest stages of conceptual design, simple PSA models can be useful in understanding how a design can be vulnerable to certain failures. Gen IV technologies will impose new technical challenges to PSA. However, most of the traditional PSA technologies are still valid. The new required techniques shall be improved gradually by the iterative process among the ISAM tools and matured as the design evolves. Hence, it is highly recommended to start the PSA as early as possible.

8) From a more general point of view, both internal and external hazards shall be included in the ISAM framework. It is suggested to include the specific technical instructions on the external hazards evaluation in a revision of the ISAM method, since we have much less knowledge on the effect led by severe external hazards.

9) ISAM methodology is not a standalone method. Safety objectives and the so called Top Level Regulatory Criteria (TLRC) do not explicitly appear of importance in the ISAM method. As a consequence, the ISAM method does not allow knowing whether the final design meets the safety objectives or not. For instance for the VHTR, the practical elimination of core melt down seems to be a safety objective and implementation of ISAM method should ensure that the safety demonstration of the practical elimination is comprehensive. Well defined safety objectives and TLRC will also be helpful to the implementation of the ISAM tools such as PIRT, DPA and PSA.

8. References

1. An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems – RSWG; Version 1, June 2010. https://www.gen-4.org/gif/jcms/c_9366/risk-safety
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