Technical Meeting on the Safety of High Temperature Gas Cooled Reactors and Molten Salt Reactors

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Digital Engineering applied to the Nuclear Safety of Molten Salt Modular Reactors

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To answer the wide variety of needs and usages of the future nuclear reactors, technical and commercial options will be very diverse.
The increasing number of Small and Modular Reactor (SMR) projects being developed on an international scale confirms this trend. These SMR reactors will mainly be installed in areas where the electric needs are limited or where the grid cannot absorb the full production of a larger scale unit.
Their clustered installation can also help solving one of the main issues that nuclear operators must address: the financing. In a clustered SMR concept, the construction of the last units can be financed by the production of the first ones.
In addition, small reactors clustering eases outage planning.
This being, to be relevant, SMR must produce flexible energy at a moderate cost, while giving assurance on their effective ability to operate safely and to obtain and keep the relevant authorizations.\\

Yet, building smaller units directly has an impact on the cost of the energy produced, as large units take advantage of some downsizing effects.
As a consequence, SMRs must create disruptions in the way they are licensed, built, transported, commissioned, operated, maintained and dismantled. Their integration into a more efficient fuel cycle must also be looked upon very closely. Meanwhile, some new uses of nuclear energy, such as efficient hydrogen production or various chemical industry processes require higher temperatures than the usual water reactors can reach.\\
Put together, these points lead to re-open the Molten Salt Reactor (MSR) or the High Temperature Gas Reactor (HTGR) concept files, with fully new design options. These new designs will require new materials, new manufacturing techniques and new operating schemes. All this will imply a new engineering approach, where digital techniques will be at stake. \\

At this point, nuclear safety and its place in the licensing process become key, as public acceptance of these "exotic" and widely spread reactors will probably be an issue.\\

Nuclear safety is sometimes seen as a barrier to innovation, in particular when the approbation of new techniques generates a complexity that might weaken both the reliability of the facilities and the coherence between safety objectives and design options. Still, some innovative techniques can help achieving a sound cost-efficiency balance, especially when they reduce the interfaces and the discrepancies that often go with nuclear safety issues. Digital engineering is one of these favorable innovations, in particular when it enables a global Model-Based Systems Engineering (MBSE) approach.\\

The INternational Council On Systems Engineering (INCOSE), defines MBSE as ''the formalized application of modeling to support system requirements, design, analysis, verification and validation activities beginning in the conceptual design phase and continuing throughout development and later life cycle phases". Applied to the nuclear field, the use of MBSE makes it possible to combine nuclear safety routines and design options at the very beginning of the process. Postulated initiating events, transient calculations or radiation protection simulations can be performed within the design model itself. This integration is sometimes designated as a "digital twin".\\

The merging process requires not only to describe the reactor in a model that integrates design data, but also to transfer the safety demonstration and its associated requirements into a systemic model. As the nuclear safety prescriptions and requirements usually come through documents, this stage can be looked at as a document to data conversion, in which Artificial Intelligence (AI) is a great help. Once the system and its regulatory environment reach a sufficiently detailed data-centric description, the merging of the two (facility and regulatory) models can be achieved efficiently.\\

The sooner this integration is made in the process, the better. For instance, site characteristics can be described as parameters of the facility so that they can be adapted to local constraints. The resulting flexibility will be crucial for the development of the SMR model as costly site implementation studies will be drastically reduced. As a result, digital design and engineering is not an option for SMRs. It has a key role in the performance these reactors will obtain, in a matter of efficiency, of reliance and of nuclear safety, which directly impacts the licensing costs.\\

To apply a MBSE process to a MSR SMR project, two parallel paths must be followed.\\

The first path concerns the safety and the production functions that the system must fulfill in order to maintain a safe operation. Typically, confinement, radiation shielding, heat removal, criticality control and their derived requirements are structured in a scheme that will later enable the allocation of the safety functions to the facility subsystems. These subsystems are obtained through a more classical Plant Breakdown Structure (PBS) and modeled in a digital mock-up that integrates at least the geometrical, mechanical, thermal and neutronical characteristics of each component. This digital mock-up is animated by a software that simulates the thermohydraulics and the neutronics of the reactor. This can use existing computational modules.\\

The second path is the construction of a synthetic nuclear safety regulatory model. This can be achieved by a mass processing of a large variety of data and feedback. The first step of this compilation can take advantage of AI techniques, that rely on the identification of relevant ontologies.\\
Various level of references can be combined through this approach, from high level international standards and recommendations to local regulatory or operator prescriptions. All are gathered and structured in a comprehensive database that enables their allocation to the PBS components.\\

As a consequence, the MBSE methodology can reduce the risk of an insufficient adaptation of regulations or recommendations. The logical loop that links design options with safety requirements is shorter and more reliable than in a classical process. This helps reduce technical discrepancies, while enabling the testing of a larger spectrum of technical options, including I\&C, as SMRs will often be massively automated and remotely operated. This also helps achieving the breakthroughs and innovations required by the SMR context and that can be costly when using existing codes and standards. To put it in other words, this contributes to demonstrating the achievement of the objectives without taking excessive margins, provisions and without adjusting existing standards.\\

As a conclusion, MBSE methodology makes it possible to put emphasis on nuclear safety, without introducing an additional factor of complexity or cost in the project.\\

The article takes as an example the systems engineering approach followed by Assystem to build the licensing case of a MSR eXtra Small Modular Reactor (XSMR), the NAAREA reactor.

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Safety of MSR-FUJI

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1) MSR-FUJI Safety Concept
MSR-FUJI is a graphite moderated MSR with fluoride salt, proposed by the authors based on ORNL study on MSBR. Safety concept of MSR-FUJI is described at first. The safety functions, which MSR must be equipped, are essentially the same as those for LWR. They are shutdown function, cooling function, and containment function.

2) Safety Analysis
In order to investigate a wide range of safety on MSR, all possible accidental scenarios must be considered. At first, the following two types of accidental scenarios are considered. The first one is the external cause accident, which is initiated by external events, such as earthquake, tsunami, flood, wind, fire, turbine missiles, terrorism and so on. Another one is the internal cause accident,

which is caused by internal events, such as over-pressure, over-heat and other mechanical failures of the primary loop boundary. The external cause accidents are generic to any reactors, and it is not described here. The internal cause accidents are sub-categorized as below.
(1) Power increase accident or RIA (Reactivity Initiated Accident), for example, control rod ejection accident.
(2) Flow decrease accident, for example, fuel salt pump stopping accident.
(3) Fuel salt leak accident, for example, rupture of primary loop pipes.
(4) Other accidents, mostly MSR specific, such as molten salt freeze accident.

3) Transient Analysis Code
In order to perform numerical evaluation of MSR safety, a transient analysis code (DYMOS) is being developed. It can be used for transient/accident analysis, such as reactivity insertion accident, or pump trip accident, and so on. Verification of DYMOS code for the experimental MSR (MSRE) is performed.

4) Safety Design Guidelines
There are two principal guidelines on reactor safety for both reactor designer and licensing authority. One of them is safety design guidelines, which are used for conceptual design of MSR. They are categorized as Overall requirements, Protection by multiple fission product barriers, Protection and reactivity control systems, Fluid systems, Reactor containment, Fuel and radioactivity control, Salt systems and control, Other design requirements, and Additional design basis accidents (DBA).

5) Safety Analysis Guidelines
Another guidelines are used for safety analysis. Here, Abnormal Operating Transients (AOT) and Design Basis Accidents (DBA), which shall be evaluated in licensing, are specified with safety limits.

6) Freeze Valve Experiment
Freeze valve is an important passive safety system used in MSR, and the melting time of frozen salt is important. Experiments and numerical simulations were conducted to investigate the effects of important parameters on the opening time of freeze valve. The calculation model was verified by the experimental results.

3

Overview and Status of the Safety Basis Development for the Molten Chloride Fast Reactor in Support of DOE Authorization

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The Molten Chloride Reactor Experiment (MCRE) is a low-power reactor that is designed to measure key reactor physics and provide foundational knowledge to design and license a commercial Molten Chloride Fast Reactor (MCFR). A five-year contract has been awarded to support this effort via the United States Department of Energy (DOE) Advanced Reactor Demonstration Program. To support an aggressive timeline, the MCRE is proposed be built within an existing facility at the Idaho National Laboratory (INL), specifically within the renovated Zero Power Research Reactor cell, now referred to as the Laboratory for Operation and Testing in the United States test bed (LOTUS) [1]. As the reactor is being designed for operation as a research reactor on a DOE site, a DOE-specific authorization process is required to build and operate MCRE. Although the DOE process leverages some of the same guidance, it is entirely separate from the processes applicable to obtaining a license for a commercial nuclear plant through the United States Nuclear Regulatory Commission (NRC). The purpose of this paper is to outline the proposed approach and implementation details for developing a safety basis to support authorization.
At the highest level, getting DOE approval of a nuclear facility safety case requires meeting the

requirements of 10 Code of Federal Regulations (CFR) 830. This high-level document points to additional lower-level DOE standards, of most importance are DOE Order 420.1C, DOE Standard 1189, and DOE Standard 3009. These documents provide additional requirements and the documentation (and structure) needed for DOE authorization, such as a Safety Design Strategy (SDS) document and a Documented Safety Analysis (DSA). Because there is no guidance available that specific design and/or safety requirements for Molten Salt Reactors (MSRs), the approach adopted for developing the safety basis for MCRE is based upon the Licensing Modernization Project's (LMP) guidance in NEI 18-04 [2]. Although the LMP methodology was developed for application within the NRC framework, recent work in reference [3] on the Versatile Test Reactor has adapted the Risk Informed Performance Based (RIPB) approach embodied in NEI 18-04 to support meeting DOE authorization requirements. The interest in utilizing the RIPB approach is motivated by long term project goals; because this approach is what will be exercised in future commercial MCFR licensing efforts, developing competence in exercising the methodology for MCRE is highly useful. The outputs of the RIPB process include a systematic safety classification of the systems, structures, and components (SSCs), selection of safety basis events (SBEs), and evaluation of defense in depth (DID) adequacy.

Fundamentally, the RIPB approach utilizes a combination of probabilistic techniques to derive the likelihood of a given event sequence and combines that with deterministic methods to provide an estimation for the consequence of a given event sequence. A key element of this concept is the comparison of the quantitative results for each SBE to a frequency-consequence (F-C) target. For this project, a Probabilistic Risk Assessment (PRA) has been performed using software tools of the EPRI Risk and Reliability Workstation [4], and by following the ASME/ANS Non-Light Water Reactor Standard [5] for content development guidelines. To date, the focus of the PRA has been on the internal events hazard group, which is to serve as a basis for deriving all other internal and external hazards. Initiating events have been identified from an examination of a hazards analysis of the Aircraft Reactor Experiment [6], safety basis development of the Molten Salt Reactor Experiment (MSRE) [7], MSRE unusual operating conditions [8], recent industry-academia publications [9], and a Master Logic Diagram developed at TerraPower. With little operational and component reliability data available for MSRs, much of the data gathering efforts have been done utilizing available liquid sodium component data. For example, salt-wetted component data have utilized [10] while all other components have utilized [11]. Common cause events were quantified using generic common cause parameters from [12]. Event trees and fault trees have been developed on a safety functional basis to understand reactivity control, decay heat removal control, fuel pump trip, core offload, and prevention of fuel freezing.
To complement the PRA, deterministic analyses of SBEs will be performed to provide a quantitative determination of the consequence of a given event. This is accomplished with systems models developed in GOTHIC [13]. While not explicitly required, the approach for model development is leveraging guidance from the NRC's Regulatory Guide (RG) 1.203 [14]. The guidance taken is focused on model requirements development and identification of key phenomena via a phenomena identification and ranking table (PIRT); however, much of the guidance in RG 1.203 focuses on linking phenomena to experimental data, which are not yet available for MCRE. Modifications have been made to GOTHIC (to be released in version 8.4 in 2022) to handle specific flowing fuel MSR physics, such as delayed neutron and decay heat precursor movement within the liquid fuel. The methodology was developed in coordination with TerraPower and provides a modification to the point kinetics methodology by leveraging the tracer tracking feature already developed in GOTHIC.

Initial efforts on dose consequence have focused on developing a bounding consequence estimate to develop a preliminary understanding of the potential hazard of the system. This calculation is straightforward due to the relatively low fission product inventory based on the planned 1000-hour full power operation. A source term is derived from the output of burn-up calculations based on a full core MCNP [15] model. The bounding dose calculation considered the entire fuel salt and cover gas inventories escaping the fission product boundary . A challenge in this approach is selecting air release fractions (ARFs), as there is minimal experimental data on the fission product forms within the fuel salt. Thus, conservative values are selected, ARFs of 1.0 are selected for the volatile isotopes and ARFs of 0.01 are selected for everything else based on limiting conditions for liquid metal spills [16]. At present, utilizing these numbers is reasonable from the design perspective as the consequence of SBEs analyzed remain below regulatory limits. However, going forward in the MCFR development program these uncertainties will have to be reduced and a mechanistic understanding based on experimentation should inform the ARFs.
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4

ON THE SOURCE TERM FOR SEVERE ACCIDENT IN MOLTEN SALT REACTORS

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ON THE SOURCE TERM FOR SEVERE ACCIDENT IN MOLTEN SALT REACTORS

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Abstract

Molten Salt Reactors (MSRs) are rather category of reactors than a single concept. In majority of the concepts the fuel cycle performance profits from the consideration that selected gaseous, volatile, non-soluble fission products could be removed during the operation from the liquid fuel. Many concepts also foresee integration of fuel cleaning, or actually processing, unit in the same complex. Accordingly, the distribution or radiotoxicity in the MSR system may strongly differ from classical reactors with solid fuel. At the same time, the presence of driving forces can be eliminated (see Fig. 1).

FIG. 1. Illustration of source term distribution and presence of driving forces in LWR (left) and MSR with salt treatment unit (right).

In this presentation three topics related to the source term of MSR will be addressed. Firstly the distribution of radiotoxicity and decay heat will be presented for a reference MSR concept. Secondly, the source term locate in the fuel salt will be characterized. And finally some initial comments will be provided from the probability safety assessment.
The in-situ treatment of fuel salt will result in distributed localization of radioisotopes. For a reference Molten Salt Fast Reactor (MSFR) with blanket there are 7 major location of the source term listed in Tab. 1. For illustration the total ingestion radiotoxicity after very long operation in presented in the table for groups of nuclides with the same atomic number. These groups represent quite well the decay chains of fission products, when delayed neutron emission is neglected. Since it is instant radiotoxicity just after reactor shut-down, it is the highest in the off-gas system for majority of the presented atomic numbers. From this perspective, the source term left in the fuel is relatively low.
Assuming potential severe accident with fuel salt spilling on the bottom of the containment, the temperature increase, driven be decay heat, may result in release of the source term from the fuel salt. Such a scenario was simulated [1] by GEMS-Melcore coupled code cGEMS to quantify the total released activity in form of aerosols and vapors and its elemental break-down (see Fig. 2). The assumption was that during the hypothetical accident salt heats-up from 800℃ to 1500℃. As it can be seen the major activity carrying element is Zr. The usual elements like I, Cs and Sr, which are problematic in LWR are absent, because all of them have either gaseous (Xe) or metallic (Te) predecessors, which are removed during the continuous salt treatment. Nonetheless, it does not decrease the overall risk. It only means that the location of the I, Cs, and Sr is out of the fuel salt.
To understand what main be the main events with vessel damage preliminary probabilistic safety assessment was accomplished for graphite moderated MSR relying on salt draining system as a safety approach to avoid severe accidents [2]. Major conclusion was that first barrier or actually reactor vessel and piping failure is a consequence of double failure. Any of the trigger event like: steam generator tube rupture, blackout, pump trip, etc. (see Fig. 3) should be combined with draining failure to damage the vessel.

TABLE 1. Rough assessment of radiotoxicity division by atomic number after 200 effective full power years operation of MSFR for intensive gaseous and non-soluble fission products removal rate.

FIG. 2. Total released activity in form of aerosols and vapors (left) and its elementral break-down (right) during the hypothetical accident (salt heat up from 800℃ to 1500℃) [3].

FIG. 3. Main events with vessel damage identified by preliminary probabilistic safety assessment [2].

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5

Qualitative Risk Assessment for the Hermes Reactor

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The Hermes reactor is a non-power reactor currently under construction permit application review with the U.S. NRC. The reactor is the first iteration of the Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR) which is an advanced reactor technology developed in the United States over the last decade. Its fundamental concept is the combination of Tri-structural Isotropic (TRISO) particle fuel coupled with a molten fluoride salt coolant. This combination results in a high temperature, low-pressure reactor system with robust inherent safety characteristics. In support of this application, the Hermes accident analysis identifies potential events by the application of hazard analysis methodologies to evaluate the design of the facility and processes for potential hazards, initiating events, scenarios, and associated prevention and mitigation controls. [1]

A separate risk evaluation technique called the Qualitative Risk Assessment (QRA) is employed to validate the accident analysis event identification and prevention controls. The process leverages Probabilistic Risk Assessment (PRA) methods, requirements and techniques to ensure it is robust and comprehensive, but is a qualitative process in that no frequencies or probabilities are applied. It is right-sized for the following two outcomes in its support of the accident analysis:

1. **Postulated Event Validation:** Validate the completeness of the list of accident analysis postulated events
2. **Prevention Assumption Development:** Develop a set of design and operation assumptions that ensure sequences outside the design basis are prevented The process comprehensively catalogs the disturbances that may occur during Hermes operation through Hazard and Operability (HAZOP) studies, identifies those that represent a challenge to the plant using Initiating Event (IE) analysis and delineates the event sequences (ES) which could emerge from each IE. The sequence end states are then categorized as either mitigated or prevented to support the two outcomes.

The HAZOP studies are utilized to systematically identify process deviations using a guide word-based approach for key Hermes systems. This approach breaks down each major node in a given system into a set of parameters with a standard set of deviations being applied to each. Plant level effects are then postulated. The output from the HAZOP studies is a list of disturbances that is passed to the IE analysis for further inspection.

The IE analysis is where the raw information from the HAZOP is filtered into individual IEs and IE groups. IEs are identified based on whether the disturbance poses a challenge to the plant that requires successful mitigation to avoid release of material at-risk. An IE grouping procedure is then applied to produce a more manageable set of IE groups to analyze in the ES analysis. To facilitate IE grouping, attributes relating to operator performance, reactivity control, heat removal and barrier status are recorded for each IE.

The IE groups carried forward are then analyzed using ES analysis similar to that used in a PRA and result in end states that are either mitigated or prevented.

Mitigated sequences are those analyzed in the accident analysis as postulated events. All mitigated sequences are mapped to the corresponding postulated event in the accident analysis as a validation step to ensure it represents the full spectrum of internal events.

Prevented sequences are those that are screened-out of the accident analysis. The Prevented Sequence Analysis examines each failure mode produced from the ES analysis and develops a set of candidate assumptions that are designed to combat each meaningful failure mode along a prevented sequence path. The assumptions can apply to design features, engineering programs and maintenance & testing that are important to keeping a given sequence prevented. Each assumption is periodically reviewed by a subject matter expert who proposes refinements, rejects or accepts the assumption.

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Evaporative release model in KP-SAM for source term evaluation

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SAM (System Analysis Module) is a systems code developed by ANL for advanced reactor system and safety analysis. It takes advantage of U.S.DOE's object-oriented application framework MOOSE, and modern software development environment and numerical methods to ensure the quality of the code. KP-SAM is the branch of SAM code with specific models for KP-FHR designed by Kairos Power, LLC. KP-SAM is primarily designed for heat transfer and single-phase fluid dynamics, but it has been recently enhanced to include the models for source term transport and release. One of the enhancements is a new evaporative release model, which is presented here.

The objective of the evaporative release model is to evaluate the release rates of source term materials (fission products, actinides, and activation products) through evaporation. For conservatism, thermal and mass transfer resistances on the molten salt side are ignored, i.e., the evaporative releases rate are determined by the gas side resistance only. Partial pressures of source term species are evaluated with Raoult's law with an activity coefficient to account for the non-ideality of the dissolved species in Flibe. The mass transfer coefficient between the liquid and gas is evaluated with correlations for various forced and natural convection flow conditions specified by KP-SAM users through inputs. The release rates are then functions of the mass transfer coefficient and the concentration differences of the species between the interface and the bulk gas space. Once the source term species are released from the liquid to the gas, they are assumed to be always airborne for conservatism, i.e., the deposition and resuspension of the species are ignored. The source term species in the gas are then transported by gas flows until released to the environment.

The evaporative release model allows KP-SAM to assess source term related phenomena and expands the code applicability from system analysis to source term related applications. Such a capability expansion helps support the analytical and licensing efforts of KP-FHR.

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Preliminary assessment of a reactivity insertion event in the Hermes FHR low power demonstration reactor

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Kairos Power, the leading developer of Fluoride-salt-cooled High-temperature Reactor (FHR) technology, has made significant progress in its modeling capabilities. In support of Kairos Power's PSAR submittal to the USNRC, the thermal hydraulics systems code KP-SAM was used to demonstrate the capability to model a reactivity insertion accident in a proposed FHR low power demostration reactor, dubbed "Hermes." This activity highlights progress made in the use and development

of KP-SAM, which is an internally developed variant of Argonne National Laboratory's advanced systems code System Analysis Module (SAM). This activity also exemplifies Kairos Power's ability to perform detailed transient calculations.

In this simulation, a model of the Hermes reactor undergoes a spurious control rod withdrawal from steady state. This scenario was selected as it is expected to be the bounding reactivity insertion event. In the first few seconds of the transient, reactivity is inserted as the most reactive control rod is withdrawn according to its rod worth curve. The reactivity insertion is partially offset due to inherent reactivity feedback effects in the reactor. Shut down rods are inserted after the reactor power rises above a set point, which also triggers a pump trip. The negative reactivity insertion initiated by the shut down rod insertion during scram quickly overwhelms the positive insertion from the control rod withdrawal, which is still in progress at this point. The system temperatures and flowrates drop quickly as the reactor shifts to decay heat removal mode, at which point the system is stable and the reactivity insertion event is over.

While the results are preliminary, this work demonstrates the advanced modeling and simulation capabilities that Kairos Power has developed. While KP-SAM is an in-house code developed at Kairos Power, many of the features used are available to the wider advanced reactor community in SAM. This work will be refined and similar work will be performed as the design of Hermes matures and modeling capabilities and validation basis are developed further.

8

Considerations on the new safety paradigm provided by liquid-fuelled reactors - Illustration on the MSFR concept

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Liquid-fuelled molten salt reactors (MSRs) may be different in terms of design, operation and safety issues compared to large water-cooled nuclear power plants (NPPs). This paper presents in a first part an analysis of the safety paradigm for such systems, based on the identification of safety-related innovations of such MSRs. The second part of the article is dedicated to an application to the Molten Salt Fast Reactor (MSFR) concept developed in France by CNRS to illustrate the safety principles application on a specific design representative of similar liquid-fuelled MSRs.

MSR technological specificities (mainly due to liquid fuel) and innovations have been identified by comparison with phenomena and usual safety features of large water-cooled power plants. The method used for identifying the safety issues is a top-down approach based on the analysis of generic safety concerns associated to the MSR concept, independently of the possible design solutions adopted by the designers and of any specific accident scenario. The application of the method raises questions about the implementation of safety concepts that may need to be adapted for MSR, notably defence in depth, severe accident consideration, etc. Currently, many liquid-fuelled MSR concepts are investigated worldwide. Some of the developers foresee commercial deployment within the next ten years, other designs are at an early stage of the development, and it is expected that new MSR concepts will be proposed in the future. Considering the various current and future MSR projects, the paradigm will have to be declined and adapted by the designers in safety approach and design specific issues in the frame of each project.

Based on these safety-related innovations of MSFR, the idea is to analyse how the fundamental safety principles (defence in depth, severe accident consideration (i.e., definition / prevention / mitigation) are applied to MSFR, taking into account the safety requirements defined in [1].

For 20 years, the French CNRS with national and European partners has focused R&D efforts on the development of a new MSR concept called Molten Salt Fast Reactor (MSFR) [2] selected by the Generation-IV International Forum for its promising design and safety features. The reference MSFR design, retained for illustration purpose in this paper, is a 3000 MWth breeder reactor containing 18m3 of a liquid circulating fluoride fuel salt. Even if no commercial deployment of MSFR is foreseen in the near future, this MSR concept is one of the most studied in the world up to now, with many publications and communications available including on safety issues [3,4,5]. The MSFR technological innovations related to safety will be detailed and a preliminary analysis of the safety principles application to MSFR is presented in this paper.

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9

Some lessons learnt from the PRA of a gas reactor

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The paper presents the expert view on the main lessons learnt from the performing a full scale PRA levels 1 to 3 for a gas reactor- pebble bed type (PBMR).
The view is a in depth evaluation of the methods, results and lessons from the project of PRA for PBMR performed in the period 2002-2006, in which the author was the main senior expert and technical coordinator.
The time passed and the new views on nuclear safety after Fukushima showed the importance of many aspects encountered during the projects at technical, organizational and licensing level revealed by the study.
This is a view on past lessons still considered of high actual importance and significance for the safety evaluations of such reactors

10

GIF Risk and Safety Working Group Activities in Support of VHTR and MSR Systems

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The six non-water-cooled advanced reactor concepts been selected by the Generation-IV International Forum (GIF) to represent a diverse set of design and safety characteristics. The GIF Risk & Safety Working Group (RSWG) was formed to promote a consistent approach on safety, risk, and regulatory interfaces between six systems, to establish safety principles, objectives, and attributes based on GIF safety goals as input to R&D plans, and to support implementation of the technology-neutral Integrated Safety Assessment Methodology (ISAM) for design and licensing of specific design tracks.

Since its inception, the RSWG collaborated with GIF System Steering Committees (SSC) for joint development of:
• "white papers" to demonstrate applicability of ISAM for self-assessment of select Generation-IV design tracks,
• "safety assessment" reports as summaries of current state of high-level design attributes, challenges, and remaining R&D needs, and
• "safety design criteria and guidelines" for specific Gen-IV systems.

Generation-IV reactors are significantly different from the earlier generation of light water reactors. The overall success of the GIF therefore depends on, among other factors, the ability to develop, demonstrate, and deploy advanced system designs that exhibit excellent safety characteristics. While the RSWG recognizes the excellent safety record of nuclear power plants currently operating in most GIF member countries, it believes that progress in knowledge and technologies, and a coherent safety approach, hold the promise of making Generation IV energy systems even safer and simpler than this current generation of plants. Three specific GIF safety goals are the excellence in safety and reliability in normal operation, very low likelihood and degree of reactor core damage in accident conditions, and elimination of the need for offsite emergency response. This presentation will focus on the ongoing RSWG activities to achieve these high-level safety objectives in support of GIF VHTR and MSR systems.

11

Preliminary Assessment of a Station Black Out Event in the Hermes FHR Low Power Demonstration Reactor

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Kairos Power, the leading developer of Fluoride-salt-cooled High-temperature Reactor (FHR) technology, has made significant progress in its modeling capabilities. In support of Kairos Power's PSAR submittal to the NRC, the thermal hydraulics systems code KP-SAM was used to demonstrate the capability to model a station black out (SBO) event in a proposed FHR low power demonstration reactor, dubbed "Hermes." This activity highlights progress made in the use and development of KP-SAM, which is an internally developed variant of Argonne National Laboratory's advanced systems code System Analysis Module (SAM). This activity also exemplifies Kairos Power's ability to perform detailed transient calculations.
In this simulation, SBO is initiated by simultaneously tripping both the primary pump and the secondary flow controller. In the first few seconds of the transient, the loss of primary side flow results in an increase in core temperatures, leading to an insertion of negative reactivity suppressing reactor power. The reactor protection system detects high upper plenum temperature soon afterwards, resulting in the insertion of shutdown rods which brings the core power down to decay heat levels . Next begins the heat up phase of the event where in vessel natural circulation transports heat within

the system while vessel and internal temperatures continue to rise. Eventually, radiative heat transfer from the vessel to the decay heat removal system (DHRS) matches the decay heat load from the core, which marks the beginning of the cooldown phase of the event. The cooldown phase of the event continues until the mission time for SBO is achieved and the event is considered terminated.
While the results are preliminary, this work demonstrates the advanced modeling and simulation capabilities that Kairos Power has developed. While KP-SAM is an in-house code developed at Kairos Power, many of the features used are available to the wider advanced reactor community in SAM. This work will be refined and similar work will be performed as the design of Hermes matures and modeling capabilities and validation basis are developed further.

12

Verification of Design codes for assessment of HTGRs Nuclear Design safety

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Abstract
This abstract describes the verification of computer codes for HTGRs core neutronic Design. Verification of the computer codes for HTGRs covering several issues of neutronic design regarding the reactor core criticality, reactivity coefficients, control rods worth, neutron fluxes, and power distribution. The verification is performed by comparison with the results of several institutions conducted for the IAEA benchmark (IAEA-TECDOC-1382).
The 10 MWth High-Temperature Gas-cooled reactor-test Module (HTR-10) is the first reactor used in benchmark HTGR core design. It is a graphite-moderated helium-cooled reactor of spherical fuel elements with pebble bed type. A heterogeneous model for the reactor core was built as input for the MCNPX Monte Carlo code. The comparison of the results of MCNPX with VSOP and MCNP4B, DELIGHT, WIMS/D4, SRAC, TWOTRAN-II and CITATION, NJOY and MCNP4A, WIMS-D/4 and JAR-3D, PANTHER and VSOP, MCNP4B using ENDF/B library, SCALE 4.4 using ENDF-B IV library, TRIPOLI4, RODCIT and TOTMOS, and VSOP-PBMR codes for the neutronic design of the HTR-10 reactor core was performed determining the basic characteristics of HTGR verifying the codes for use in core design analysis. The neutronic parameters are calculated as the following:
• Critical height of the fuel pebbles in the core calculated in case of air coolant at 27 ℃;
• Effective Multiplication factor (Keff) and reactivity temperature coefficient at the full height of the fuel pebbles in case of helium coolant at 20, 120, 250 ℃;
• Control rod worth in case of helium coolant at 27 ℃ for one and ten control rods with a full height of fuel pebbles and ten control rods with a critical height of fuel pebbles;
• Control rod worth in case of helium coolant at 27 ℃ for ten control rods with a critical height of fuel pebbles.
The model results are in the range of the results of the aforementioned codes.
The 268 MWth Pebble Bed Modular Reactor (PBMR) is the second reactor used in benchmarking HTGR core design using a homogeneous model of the spherical fuel element. The verification of the MCNPX with VSOP and MCNP4B codes for the neutronic design of the PBMR reactor core was performed for the criticality, thermal and fast neutron flux, and reactor core power density.
Using the mentioned deterministic codes solving the neutron diffusion equation with models based on a macroscopic cross-section library generated versus the fuel burn-up or Monte Carlo code based on the same library can be concluded to be valid for the neutronic design of HTGRs core including the MCNPX code model. It's intended to use the verified model in the deterministic safety assessment of the small and modular HTGRs with different fuel compositions.

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Maximum Hypothetical Accident for the Hermes Reactor

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Introduction

The Maximum Hypothetical Accident (MHA) for the Hermes non-power reactor is a deterministic event where hypothesized conditions result in a conservatively analyzed release of radionuclides that bounds a potential release from other postulated events ("Hermes Non-Power Reactor Preliminary Safety Analysis Report Rev. 0" 2021). The MHA analysis is consistent with the fission product release accident analysis required for the 10 CFR 100.11 determination of exclusion area, low population zone, and population center distances. The MHA is a bounding event with conservative radionuclide transport assumptions that challenge the important radioactive retention features of the functional containment.

MHA Narrative

The reactor is tripped due to an unspecified transient. Pre-transient diffusion of radionuclide from the fuel kernels are hypothetically and conservatively not modeled to maximize available fuel inventory release. The system disturbance is detected directly or indirectly by the reactor protection system, which initiates control and shutdown elements insertion, fulfilling the reactivity control function. The highest worth element is stuck out and does not insert. The reactor decay heat removal system performs its function to limit reactor temperature and fulfill the heat removal function. A hypothetical temperature curve is used to bound expected system temperatures and thus conservatively model radionuclide movement through the credited barriers:
Radionuclides diffuse from all TRISO cohorts (e.g., intact, failed silicon carbide (SiC), exposed kernels). This fuel release is hypothetical given the steady state diffusion assumptions.

- Tritium desorbs from in-vessel graphite and steel.
- Radionuclides evaporate and degas from the Flibe driven by natural convective forces in the cover gas.
- Non-gaseous radionuclides (i.e., Salt-soluble Fluorides, Noble Metals, and Oxides) evaporate from the Flibe free surface in the reactor vessel.
- Gaseous radionuclides, including tritium, bypass the Flibe.
- Airborne radionuclides conservatively bypass the cover gas space and directly enter the facility air.

All radionuclides that have been mobilized in the facility air are then transported by dispersion to the site boundary on the basis of conservative analysis with unfiltered ground level releases.

Material at Risk and Modeled Retention Barries

The MHA analysis must identify the both the sources of material at risk (MAR) and barriers to release for each source of MAR. The sources of MAR for the MHA are:

- the TRISO fuel;
- the Circulating Activity within the Flibe coolant;
- Argon gas activated in open graphite pores;
- tritium retained inside graphite grains, steel structures, and suspended in Flibe.

The primary barrier to release are the fuel (i.e., kernel and TRISO barriers, the Flibe. To a lesser extent, the reactor building and atmospheric dispersion are also expected to contribute to the overall RF. Once in the facility air, the MAR is transferred the atmosphere and subsequently dispersed to a receptor at the Exclusion Area Boundary and Low Population Zone.

Conclusion

Through the use of functional containment of the robust TRISO fuel and the Flibe coolant, the dose consequence results for the Hermes MHA are maintained below site dose limits in 10 CFR 100.11(a)(1-2) at the EAB and LPZ with significant margin.

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Mechanistic Source Term Methodology for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor

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Introduction

Kairos Power is pursuing the design, licensing, and deployment of Fluoride Salt Cooled, High Temperature Reactors (KP-FHR) including a nuclear test reactor and commercial power reactors. To enable these objectives, the development of a technology-specific source term evaluation methodology is required. This methodology is applicable for evaluation of the KP-FHR mechanistic source term to be used to calculate radiological source terms for anticipated operational occurrences (AOOs), design basis events (DBEs), and design basis accidents (DBAs). These source terms will be used to calculate a boundary dose for the exclusion area and low population zone, which are anticipated to be at the site boundary or a boundary that is less than 1200 meters.

General Approach

KP-FHR technology is fundamentally different from the existing light water reactor (LWR) technologies because the fully-ceramic Tri-Structural Isotropic (TRISO) fuel used in KP-FHRs has large thermal margins to damage, and the KP-FHR coolant is operated at low pressures, has high chemical stability, and the capability to immobilize solid fission products. These characteristics make the safety case of the KP-FHR technology fundamentally different from LWRs. Due to these inherent safety features of the KP-FHR, the design relies on a functional containment approach to retain radioactive materials. The functional containment approach is further simplified because the majority of the radioactive material at risk for release (MAR) is retained within the TRISO fuel used in the KP-FHR. During normal operation, the small fraction of fission products that are released from the TRISO fuel will diffuse into the molten fluoride salt reactor coolant (Flibe) or are released into the gas space of the reactor vessel or the pebble handling and storage system, depending on the location of the fuel pebbles. There will also be activation products that will be created as a result of normal operation. This abstract provides the methodology to calculate the source term for licensing basis events (excluding beyond design basis events) based on the fission products and activation products generated from normal operation of a KP-FHR.

KP-FHR Material at Risk and Release Fraction Formulation

The KP-FHR uses a functional containment approach in which the RN retention capability of the robust TRISO fuel and the RN retaining Flibe coolant obviate the need for a traditional structural containment. Similar to the scoping source term approach used by NGNP (Petti et al. 2013),the KP-FHR source term demonstrates the robustness of its functional
containment through the use of as a series
of release fractions RFs associated with each barrier, j, and RN group, i.

This approach to source term quantification is consistent with that laid out in Enclosure 1 ("Enclosure: Technology-Inclusive, Risk-Informed, Performance-Based Methodology" 2019, 2) of SECY-19-0117 ("Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors" 2019).

The Release Fraction (RF) is a function of the mechanical and thermal loadings applied to the barrier at that given time. Conversely, MAR is an integral quantity that is impacted by generation (S), decay into and out of the group, and release of RN across barriers. Once event sequence modeling begins (t_o), MAR generation quickly stops, or at least the integrated generation of MAR is negligible compared to the quantity of MAR generated during normal operations.

\begin{equation}
\text{MAR}^{i,\text{ND}}j(T)=\text{MAR}^{i,\text{ND}}_j(T = t_0) + \int\limits{t_0}^{T} \sum_{k\neq j}^J ({\text{MAR}^{i,\text{ND}}k(t) \text{RF}{k\rightarrow j}^{i,\text{ND}}(t)}) - \text{MAR}^{i,\text{ND}}_j(t) \text{RF}_j^{i,\text{ND}}(t) dt
(\#eq:MARESNB)
\end{equation}

where the superscript i, ND is the non-decayed version of RN group i.

Decay durring barrier transport is treated retractivity for each modeled isotope in the RADTRAD code (Arcieri, Mlynarczyk, and Larsen 2016) when calculating dose consequences. Conservative and prescriptively high-building leakage rates are used during building transport as part of the functional containment analysis.

Conclusion

The source term methodology is event-specific because it depends on the amount of MAR and conditions of the MAR in the locations affected by the licensing basis event. The normal operation conditions provide an initial amount of MAR, that is used to calculate the source term for licensing basis event conditions. Due to the functional containment provided by the robust TRISO fuel design and the RN retaining Flibe coolant, accident analysis for FHRs is not expected to model significant RN transport to the facility air, ensuring that acceptably low off-site doses can be demonstrated.

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GIF VHTR System Safety Assessment

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In 2018, the Generation IV Forum (GIF) Very High Temperature Reactor (VHTR) System Steering Committee and the Risk and Safety Working Group developed a VHTR System Safety Assessment report covering the general overview of the system performance goals, historical review of, and feedback from, past HTGR construction and operation experiences. The report focused on the specific safety attributes of the GIF VHTR design tracks and ongoing safety-related R&D activities. This presentation will provide a high-level summary of the GIF report to highlight the main safety features and possible challenges of the technology, to assess the status of GIF activities, and to identify future VHTR R&D needs.

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The MMR nuclear safety design and assessment

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The MMR™ system is a small modular nuclear energy system under design and licensing that delivers safe, clean and cost effective electricity and heat to remote mines, industry and communities. The energy system consists of two plants, the nuclear plant and the adjacent non-nuclear power plant. The Nuclear Plant is independent of the Adjacent Plant, requiring no supporting services for any event for its safe operation. The Adjacent Plant consists of the equipment and systems that convert the process heat to electrical power or other forms of energy as per client requirements. The Nuclear Plant would generate approximately 15 MW(t) of process heat that could supply electrical power and/or heat to a small community as the potential end user. The electrical power could also be supplied to the area grid, over an anticipated life span of 20 years.

The MMR design deploys TRISO fuel in the form of Fully Ceramic Microencapsulated (FCM) pellets. The FCM pellet has an additional SiC layer in the pellets retain fission products from failed TRISO particles and tramp uranium and serve as an additional release barrier. The SiC also protects the TRISO particles (especially exposed kernels) from chemical attack due to water or air ingress.

The overall philosophy guiding the design of the MMR may be stated as produce a safe, economical plant design which meets regulatory and user requirements by providing defense-in-depth through pursuit of four goals:

1. Maintain Plant Operation: Reliably maintain the functions necessary for normal plant operation, including the plant states of energy production, shutdown, and startup/shutdown operations.
2. Maintain Plant Protection: Assume that despite the care taken to maintain plant operation, failures will occur and provide additional design features or systems to prevent plant damage.
3. Maintain Control of Radionuclide Release: Provide additional design features or systems to ensure functional containment of radionuclides in the event that normal operating conditions cannot be maintained and/or plant protection is not assured.
4. Maintain Emergency Preparedness: Maintain adequate emergency preparedness to protect the health and safety of the public in the event that control of radionuclide release is not accomplished.

The unique aspect of the MMR is the approach which has been taken to achieve the functions of Goal 3. To accomplish this goal with high assurance, the design of the MMR has been guided by the additional philosophy that control of radionuclide releases be accomplished by retention of radionuclides within the FCM fuel pellets with no reliance on active design features or operator actions. The

overall intent here is to provide a simple safety case that will provide high confidence that the Goal 3 safety criteria are met.

There are two key elements to this philosophy which have had a profound impact on the design of the MMR:

1. The philosophy requires that control of radionuclides be accomplished with no reliance on active systems or operator actions. By minimizing the need to rely on active systems or operator actions, the safety case centers on the behavior of the laws of physics and on the integrity of passive design features. Arguments need not center on an assessment of the reliability of pumps, valves and their associated services or on the probability of an operator taking various actions, given the associated uncertainties involved in such assessments.
2. The philosophy requires control of releases by the retention of radionuclides within the fuel kernel, the coated fuel particle and the FCM pellet, rather than reliance on secondary barriers (such as the primary coolant boundary or buildings). The judgement made here is that the proof of functional containment is dramatically simplified if arguments can center on issues associated with fuel pellet integrity alone.

It is clear that the the other phenomena such as plate-out en deposition, or the remaining barriers such as the pressure boundary, filters and the citadel building can also significant contribute to the retention of fission products and this will still be evaluated as part of Defense-in-Depth and ALARA principles but not necessarily credited in the Deterministic Safety Analysis.

More information will also be shared on the overall safety analysis process that generally follows ANSI/ANS-53.1-2011; R2016 "Nuclear safety design process for modular helium-cooled reactor plants", the approach to Defense-in-Depth, Probabilistic Safety Assessment, Safety Classification, Deterministic Safety Analysis as well as the computer codes used to perform the analysis and ultimately deliver the Safety Analysis Report.

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Final outcomes of the IAEA CRP on Modular HTGR Safety Design

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The International Atomic Energy Agency (IAEA) Nuclear Technology Development Section (NPTDS) fosters information exchange and collaborative research and development on advanced nuclear reactor technologies needed to meet the increasing energy demands of the 21st century. The programme on technology development for advanced reactors and non-electric applications is in the IAEA Department of Nuclear Energy and division of Nuclear Power. It provides support to Member States to cooperate and share technology developments to facilitate near–term deployment of all these advanced reactors.

One of the important mechanisms is the use of co-operative research projects (CRPs). The selected research areas are defined by the IAEA based on the needs expressed by the member states on topics that are of common interest. Projects are typically for 3-6 years and include both technology holders as well as other countries and institutes less experienced in the topic.

The strengthening of nuclear reactor safety features is a continuous process and advanced large LWRs today employs additional measures to prevent and mitigate the consequences of complex accident sequences involving multiple failures, and of severe accidents, as part of Design Extension

Conditions. The latest Safety Design Criteria (SDC) and designs of new nuclear power plants now explicitly includes consideration of severe accident scenarios including postulated core melt and strategies for their management and mitigation. This is also reflected in the IAEA Safety Requirements No SSR 2/1.

Modular high temperature gas cooled reactors use design principles that ensure no core meltdown or significant early large radionuclide release even for postulated severe accidents. The coated particle fuel contains fission products to very high temperatures and exhibit no common mode failures or cliff edge effects. This design and safety philosophy came from the late 1980's and was born in the aftermath of the Chernobyl accident.

Based on a proposal by Japan the members of the TWG-GCR requested the IAEA to launch a CRP to study the technology specific safety design criteria that may be required for this advanced reactor technology. In the CRP initiated in 2014 the nine participating Member States represented by designers (vendors) and research organizations investigated mHTGR safety design criteria.

Two approaches were followed and compared. Approach 1 limits the scope to qualitative, functional statements of how the top-level requirements are to be met for only systems, structures and components (SSCs) that are safety-related for public safety taking specific designs into consideration. This is then compared to Approach 2 where modifications to and interpretations of the IAEA SSR 2/1 requirements are proposed largely based on engineering judgement (based on multiple licensing efforts).

When comparing the two approaches the high-level requirements match well but much less emphasis are placed on the coolant (and cooling support systems) following Approach 1 since decay heat removal does not rely on the coolant being present or cooling systems. Additional requirements, for example on the protection against chemical attack (oxidation of graphite due to massive air ingress), is also proposed by Approach 1.

The project concluded in 2019 and the final outcomes were shared in the Joint IAEA–GIF Technical Meeting on the Safety of High Temperature Gas Cooled Reactors held from 9 - 11 December 2019 in Vienna. The main objectives were to discuss the working material of the coordinated research project and to share information on the implementation of SDG for VHTRs/HTGRs by designers of innovative concepts. The meeting also discussed the further joint development of the SDC and SDG for VHTRs. At that time two draft reports were made available to the participants and GIF. After consultation within the IAEA it was decided to combine the reports and this to rework the material into a single report. The status and final outcome of the these efforts will be shared.

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An integrated energy system based on Molten Salt Reactor: inherent safety, on-demand supply and comprehensive utilization

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In order to meet the growing energy demand while protecting the environment, Shanghai Nuclear Engineering Research & Design Institute (SNERDI) is developing an integrated energy system coupling the Molten Salt Reactor (MSR) and the thermal storage system with features of inherent safety, on-demand supply and comprehensive utilization.
SNERDI started investigations in MSR from the time of establishment and had conducted zero-power experiments in 1970s. In recent years, lots of research and design work about MSR has been done to support the "Thorium molten salt reactor nuclear energy system" project (TMSR). Nowadays, with 52 years' R&D experience in multiple reactors, an integrated energy system based on MSR has been proposed by SNERDI.
The typical integrated energy system consists of the reactor system, the thermal storage system

and the steam generation system. For the reactor system, the 125MWt fluoride-salt cooled high-temperature reactor with the cross-helix-shaped fuel has been adopted after comparing different design schemes. The lithium berylium fluoride salt is chosen as the reactor coolant flowing in natural circulation. "Solar Salt" serves as the thermal storage material, with operating temperature range from 290℃ to 560℃.
A R&D list has been identified for the initial development, including: development of cross-helix-shaped fuel, studies on reactor coolant and thermal-storage salt, evaluation of design analysis codes, also an early stage work performed on core design, fuel management, main component development, structural material development, tritium control technology, thermal insulation of energy storage system, system configuration and conceptual design.

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Conceptual Design of 400MWt small Modular Thorium Molten Salt Demonstration Reactor (smTMSR-400)

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smTMSR-400 is a 400MWth/ 168 MWe small modular Thorium Molten Salt Demonstration Reactor designed by Shanghai Institute of Applied Physics, Chinese Academy of Science. smTMSR-400 is designed as a thorium convertor and in situ burner driven by low enriched uranium. The nuclear heat supply system consists of reactor module, molten salt loop, heat storage system and heat utilization system. smTMSR-400 will be applied as a high temperature heat source, which not only can be used for electricity generation, but also can satisfy the energy diversified demands, such as seawater desalination, heat supply, supercritical steam supply for industry demands and hydrogen production etc. A molten salt heat storage system will help smTMSR-400 to serve Peak and valley regulation for hybrid-energy system. After offline batch reprocessing, thorium and uranium can be recycled into a new reactor in order to minimum the spent fuel mass as well as enhance the neutron economy. Most off-line pyro-processing techniques, such as, fluoride volatility, vacuum distillation and electrochemical reduction, will used in fuel cycle of smTMSR-400.
smTMSR-400 will demonstrate the highly intrinsic safety and engineering reliability of molten salt reactor, such as shutdown by negative temperature feedback and draining off fuel salt, passive residual heat removal systems etc. The aim of safety design is to cancel off-site emergency and increase siting flexibility. smTMSR-400 uses small and modular designs to reduce and simplify the R&D challenge and difficulty of MSR. Modular design will make MSR more flexible to deploy at different places with various demands. Modular manufacture and assembly can improve the quality of equipment and enhance safety and economy. Modular construction can also reduce the financing cost and risk.

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Hight temperature reactor research in Switzerland: general and material related safety aspects

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Switzerland is since the beginning member of the GIF very high temperature reactor system arrangement and specifically of the materials project management board, with the Paul Scherrer Instiut (PSI)

being the implementing institute. Besides these contributions, different other activities, which are relevant to the safety of gas cooled reactors, have been conducted at PSI.
The presentation will summarize the contributions in materials research, with a specific attention to safety aspects. The materials being looked at are metals and ceramics, and here especially oxide dispersion strengthened steels and silicon carbide based composite materials. Both materials which promise to extend the applicable temperature of some components, and therefore offering extended safety margins. Here a specialty of PSI is the extended study of creep, from the thermal exposure to the combined exposure in a radiation field. Besides the safety aspect under accidental conditions, with an increased margin, this also provides the long-term safety of these materials under normal operational conditions.
The paper will also highlight other PSI activities, which are not directly material related. Here work concentrated on the qualitative study of accident phenomenology in HTGR and especially performing simulations of pressurized and pre-pressurized loss of forced flow accidents.