NUCLEAR ENERGY SERIES REPORT

DESIGN FEATURES TO ACHIEVE DEFENCE IN DEPTH IN SMALL AND MEDIUM SIZED REACTORS (SMRs)

FOREWORD

There is a continued interest in member states in the development and application of small and medium sized reactors (SMRs). In the very near term, most new NPPs are likely to be evolutionary water cooled reactor designs building on proven systems while incorporating technological advances and often the economics of scale, resulting from the reactor outputs of up to 1600 MW(e). For a longer term, the focus is on innovative designs aiming to provide increased benefits in the areas of safety and security, non-proliferation, waste management, resource utilization and economy, as well as to offer a variety of energy products and flexibility in design, siting and fuel cycle options. Many innovative designs are reactors within the small-to-medium size range, having an equivalent electric power less than 700 MW(e) or even less than 300 MW(e).

Broad incorporation of inherent and passive safety design features has become a 'trademark' of many advanced reactor concepts; including several evolutionary designs and nearly all innovative SMR design concepts. Ensuring adequate defence in depth is important for reactors of smaller output because many of them are being designed to allow more proximity to the user, specifically, when non-electrical energy products are targeted.

Upon the advice and with the support of IAEA member states, the IAEA provides a forum for the exchange of information by experts and policy makers from industrialized and developing countries on the technical, economic, environmental, and social aspects of SMR development and implementation in the 21st century, and makes this information available to all interested Member States by producing status reports and other publications dedicated to advances in SMR design and technology development.

The objective of this report is to assist developers of SMRs in member states in defining consistent defence in depth approaches regarding the elimination of accident initiators/ prevention of accident consequences by design and the incorporation of inherent and passive safety features and passive systems into safety design concepts of such reactors. Another objective is to assist potential users in member states in their evaluation of the overall technical potential of SMRs with inherent and passive safety design features, including possible implications in areas other than safety.

The report is intended for different categories of stakeholders including the designers and potential users of innovative SMRs, as well as officers in the ministries or atomic energy commissions in member states responsible of implementing nuclear power development programmes or evaluating nuclear power deployment options in the near-, medium-, and longer term.

Main chapters of this report present the state-of-the-art in defence in depth approaches based on the incorporation of the inherent and passive safety features to the design concepts of pressurized water reactors, pressurized light water cooled heavy water moderated reactors, high temperature gas cooled reactors, liquid metal cooled fast reactors, and non-conventional designs within the SMR range. They also highlight benefits and negative impacts in areas other than safety, arising from the incorporation of such features.

The annexes provide descriptions of the design features of 11 representative SMR concepts, used to achieve defence in depth and patterned along a common format reflecting the definitions and recommendations of the IAEA safety standards. The annexes were prepared firsthand by the designers of the corresponding SMRs.

The IAEA officer responsible for this publication was V. Kuznetsov of the Division of Nuclear Power.

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1. INTRODUCTION

1.1. Background

1.1.1. Rationale and Developments in Member States

According to the classification adopted by the IAEA, small reactors are the reactors with an equivalent electric output less than 300 MW; medium sized reactors are the reactors with an equivalent electric power between 300 and 700 MW [1].

Small and medium sized reactors (SMRs) do not attempt to benefit from the economics of scale. In most of the cases, deployment potential of SMRs is supported by their ability to fill niches in which they would address markets or market situations different from those of currently operated large-capacity nuclear power plants, e.g., the situations that value more distributed electrical supplies or a better match between capacity increments and investment capability or demand growth, or more flexible siting and greater product variety [2, 3].

It is important that small or medium sized reactor does not necessarily mean small or medium sized nuclear power plant. Like any nuclear power plants, those with SMRs can be built several-at-a-site, or as twin units. In addition to this, innovative SMR concepts provide for power plant configurations with 2, 4, or more reactor modules. The units or modules could then be added incrementally in time taking benefits of the effects of learning, timing, construction schedule (see Fig. 1), and creating an attractive investment profile with minimum capital-at-risk.

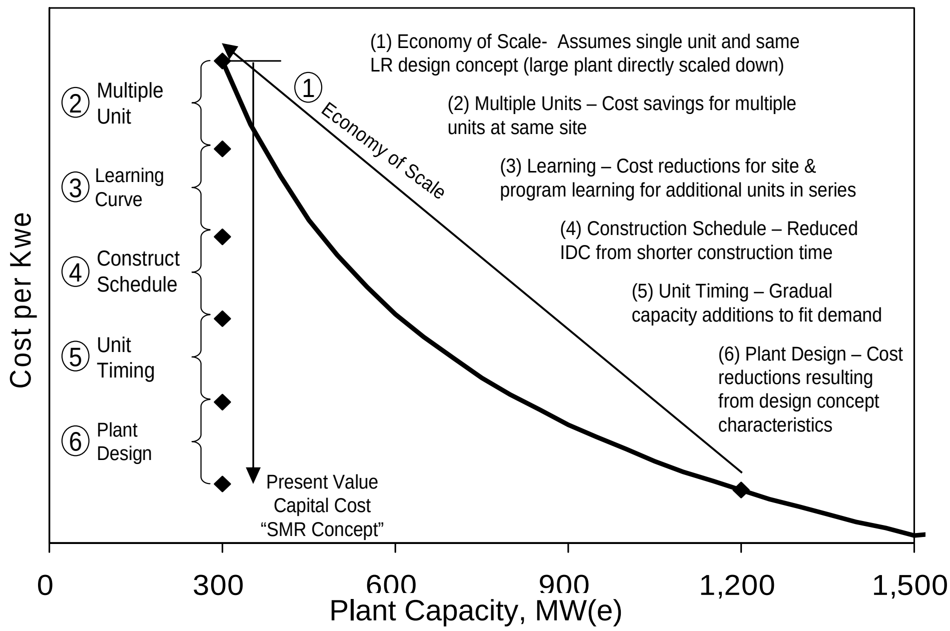

FIG. 1. A generic scheme illustrating potential SMR economic factor advantages (Westinghouse, USA).

Also, sometimes it is perceived that SMRs address the users in those countries, which currently either do not have, or have a small size of nuclear infrastructure, and are contemplating either introduction, or significant expansion of nuclear power for the first time. However, this is not the case – most of the innovative SMR designs are meant for a broad variety of applications in the developed and developing countries alike, no matter whether they have already embarked on a nuclear power programme or are only planning to do so [1, 2 and 3].

Finally, it could be emphasized that SMRs are not the only prospective nuclear option; it must be recognized that a diverse portfolio of reactors of different capacity and applications would be needed if nuclear power is to make a meaningful contribution to global sustainable development. The anticipated role of SMRs in global nuclear energy system could then be to increase the availability of clean energy in usable form in all regions of the world, to broaden the access to clean, affordable and diverse energy products and, in this way, to contribute to the eradication of poverty and, subsequently, to peace and stability in the world.

In 2008, more than 45 innovative[1] SMR concepts and designs have been developed within national or international research and development (R&D) programmes involving Argentina, Brazil, China, Croatia, France, India, Indonesia, Italy, Japan, Republic of Korea, Lithuania, Morocco, Russian Federation, South Africa, Turkey, USA, and Vietnam [2, 3].

Innovative SMRs were under development for all principal reactor lines and some non-conventional combinations thereof. The target dates of readiness for deployment ranged from 2010 to 2030.

Strong reliance on inherent and passive safety design features has become a 'trademark' of many advanced reactor designs, including several evolutionary designs [4] and nearly all innovative SMR designs [2 and 3]. Reactors with smaller unit output would need adequate defence in depth to benefit from more units being clustered on a site or to allow more proximity to the user, specifically, when non-electrical energy products are targeted and the user is a process heat application facility, e.g., a chemical plant.

This report is intended to present the state-of-the-art in the design approaches to achieve defence in depth in SMRs. Preparation of this report has been supported by the IAEA General Conference resolution GC(51)/14/B2(k) of September 2007.

1.1.2. Previous IAEA publications

Direct predecessors of this report are the IAEA-TECDOC-1485 titled *Status of innovative small and medium sized reactor designs 2005: reactors with conventional refuelling schemes* [2], published in March 2006; and the IAEA-TECDOC-1536 *Status of innovative small reactor designs without on-site refuelling* [3], published in January 2007. These reports presented design and technology development status and design descriptions for the concepts of innovative SMR developed worldwide. Design descriptions of the SMRs in these reports incorporated descriptions of the safety concepts prepared according to a common outline. However, these descriptions were rather limited in detail because of a limited space provided by these reports, also dedicated to the presentation of other aspects of innovative SMRs, including descriptions of the design, economics, proliferation resistance and security, fuel cycle options, and innovative infrastructure provisions. More important is that the descriptions of the SMR safety

[1] IAEA-TECDOC-936 [5] defines an innovative design as the design "that incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice" and would, therefore, "require substantial R&D, feasibility tests and a prototype or demonstration plant to be implemented".

design concepts in these reports had not always been structured according to the recommendations of the IAEA safety standards, specifically, as comes to defence in depth strategy.

Another predecessor of this report is the IAEA-TECDOC-1487 *Advanced nuclear plant design options to cope with external events* [6], published in February 2006, which provided structured descriptions and explanations of the design features of 14 advanced nuclear power plants intended for plant protection against the impacts of natural and human induced external events. The designs considered in that report included several SMRs.

The present report, therefore, provides an in-depth description of the safety design features used to achieve defence in depth in 11 innovative SMR concepts selected to represent all major reactor lines with near- to medium- and to longer-term deployment potential. These descriptions are structured to follow the definitions and recommendations of the IAEA safety standard NS-R-1 *Safety of the Nuclear Power Plants: Design Requirements* [7] and also include some references to other IAEA safety guides and documents, including the NS-G-3.3 *Evaluation of Seismic Hazard for Nuclear Power Plants* [8], and the NS-G-1.5 *External Events Excluding Earthquakes in the Design of Nuclear Power Plants* [9], as well as recommendations of the International Nuclear Safety Advisory Group [10], [11], and non-consensus definitions suggested in the IAEA publications [12], [5]. The basic definitions recommended or suggested in the abovementioned IAEA publications are reproduced in Appendix 2 to this report.

In September 2007, the IAEA has published IAEA-TECDOC-1570 *Proposal for a Technology-Neutral Safety Approach for New Reactor Designs* [13]. Based on critical review of the IAEA safety standard NS-R-1 *Safety of the Nuclear Power Plants: Design Requirements* [7], the IAEA-TECDOC-1570 outlines a methodology/process to develop a new framework for development of the safety approach based on quantitative safety goals (a probability- consequences curve correlated with each level of defence-in-depth), fundamental safety functions, and generalized defence-in-depth, which includes probabilistic considerations. The direction for further elaboration of the IAEA safety standards suggested in reference [13] could facilitate further design development and safety qualification of several medium- and longer-term SMRs addressed in the present report; therefore, certain suggestions of this IAEA publication are referenced in Chapter 3, which presents design features of the selected SMRs. The limitations of the information provided by member states for this report did not make it possible to consider in full the recommendations of the IAEA safety standards and guides. Wherever possible, references to other recently published IAEA reports are included, where such recommendations may be considered in more detail, e.g. see reference [6].

1.2. Objectives

The report is intended for different categories of stakeholders including the designers and potential users of innovative SMRs, as well as officers in the ministries of atomic energy commissions in member states responsible of implementing nuclear power development programmes or evaluating nuclear power deployment options in the near-, medium-, and longer term.

The *overall objectives* of this report are:

(1) To assist developers of innovative SMRs in defining consistent defence in depth approaches regarding the elimination of accident initiators/ prevention of accident

consequences by design and the incorporation of inherent and passive safety features and passive systems in safety design concepts of such reactors;

(2) To assist potential users of innovative SMRs in their evaluation of the overall technical potential of SMRs with the inherent and passive safety design features, including their possible implications in areas other than safety.

The *specific objectives* of this report are:

- To present the state-of-the-art in the design approaches used to achieve defence in depth in pressurized water reactors, pressurized light water cooled heavy water moderated reactors, high temperature gas cooled reactors, sodium cooled and lead cooled fast reactors, and non-conventional designs within the SMR range;
- To highlight benefits and negative impacts in areas other than safety arising from the implementation of inherent and passive safety design features;
- To identify issues of performance reliability assessment for passive safety systems in advanced reactors, and to highlight further research and development needs arising thereof.

Designers of SMRs not considered in the present report (currently, not less than 45 innovative SMR concepts and designs are being analyzed or developed worldwide [2, 3]) could benefit from the presented information, which is structured to follow the definitions and recommendations established by the IAEA safety standards or suggested in other IAEA publications. It should be noted that the IAEA safety standards are used as bases for national nuclear regulations in many developing countries, with this trend likely to be continued into the future.

The information presented in this report could be used in assessment studies of the innovative nuclear energy systems (INS) involving SMRs, as conducted by the IAEA's International Project on Innovative Reactors and Nuclear Fuel Cycles (INPRO) [14].

Part of this report is elaborated with participation of the research teams in member states involved in the development of methodologies for reliability assessment of passive safety systems in advanced reactors. This part (see Appendix I) provides a justification for the coordinated research project "Development of Methodologies for the Assessment of Passive Safety System Performance in Advanced Reactors", which is being implemented by IAEA in its programme and budget cycle of 2008–2009.

1.3. Scope

The report addresses 11 representative SMR concepts/ designs originating from 7 IAEA member states, which are Argentina, France, India, Italy, Japan, the Russian Federation, and the USA. The concepts have been selected to include:

- As many concepts as possible, for which noticeable progress toward advanced design stages or deployment is observed;
- Concepts representing different reactor lines; and
- To focus more on those concepts that could be deployed in the near term.

Presentation of certain SMR concepts in this report was also conditioned by the agreement of their developers to cooperate. In some cases, the designers considered the subject of this report too sensitive and withdrew from the cooperation.

1.4. Status of considered SMR designs and concepts

The SMR concepts included represent pressurized water reactors (5 inputs); pressurized light water cooled heavy water moderated reactors (1 input); high temperature gas cooled reactors (HTGRs, 1 input); liquid metal cooled fast reactors (1 input for sodium and 1 input for lead cooled reactors), and a single non-conventional design, which is a lead-bismuth cooled very high temperature reactor with pin-in-block HTGR type fuel.

Of the pressurized water reactors included, the KLT-40S (ANNEX I) has entered the deployment stage — construction began in 2007 in the Russian Federation of a pilot floating cogeneration plant of 400 MW(th)/70 MW(e) with two KLT-40S reactors. The deployment is scheduled for 2010.

Two reactors with integrated design of the primary circuit are in advanced design stages, and their commercialization could start around 2015. These are the 335 MW(e) IRIS design (ANNEX II) developed by the International consortium led by Westinghouse, USA; and the prototype 27 MW(e) CAREM (ANNEX III) developed in Argentina, for which construction is scheduled to be complete in 2011.

Two other PWR type designs, the SCOR (France) and the MARS (Italy) have a potential to be developed and deployed in a short term but show no substantial progress toward deployment. The SCOR of 630 MW(e) (ANNEX IV), which is at a conceptual design stage, is of interest as it represents a larger capacity integral-design PWR. The modular MARS of 150 MW(e) per module (ANNEX V), which is at the basic design stage, is of interest as it represents an alternative solution to other pressurized water SMRs, the solution based on the primary pressure boundary being enveloped by a protective shell with slowly moving low enthalpy water.

The advanced pressurized light water cooled heavy water moderated reactors are represented by one design – the AHWR of 300 MW(e) (ANNEX VI). The AHWR (India) is at a detailed design stage with the start-up of construction related actions expected before 2010.

The GT-MHR of 287.5 MW(e), a collaborative US – Russian concept of a HTGR with pin-in-block type fuel, is at the basic design stage (ANNEX VII). Its progress toward deployment may be not so noticeable as that of some other HTGRs (e.g., the PBMR of South Africa or the HTR-PM of China [2]); however, as passive safety design features of all HTGRs have much in common, the GT-MHR is quite representative of the passive safety design options implemented in other HTGRs.

Sodium and lead cooled fast SMRs are represented by the 4S-LMR concept of a sodium cooled small reactor without on-site refuelling developed by the Central Research Institute of Electric Power Industry (CRIEPI) and Toshiba in Japan (ANNEX VIII) and by the SSTAR and STAR-LM concepts of small lead cooled reactors without on-site refuelling developed by the Argonne National Laboratory in the USA (both described in ANNEX IX). Of the two designs, the 4S-LMR of 50 MW(e) and 10-year core lifetime is at a more advanced stage because for a similar design different essentially in the type of fuel and named the 4S, the conceptual design and major parts of the system design have been completed. A pre-application review by the US NRC has started in the fall of 2007. Construction of a demonstration reactor and safety tests are planned for early 2010s [3]. Different from it, both the SSTAR of 19.7 MW(e) and 30-year core lifetime and the STAR-LM of 181 MW(e) and 15-year core lifetime are at a pre-conceptual stage [3]. In 2008, because of a reduced funding, the activities for them in the USA were re-focussed toward a lead cooled fast reactor (LFR) Technology Pilot Plant (a demonstration plant) under a GNEP programme.

Finally, non-conventional designs are represented by the CHTR of 100 kW(th) and 15 -year core lifetime (ANNEX X). The CHTR (India) is a small reactor without on-site refuelling being designed as a semi-autonomous "power pack" for operation in remote areas and, specifically, for advanced non-electrical applications, such as hydrogen production. The CHTR is a non-conventional reactor merging the technologies of high-temperature gas cooled reactors and lead-bismuth cooled reactors. The core uses ^{233}U-Th based pin-in block fuel of the HTGR type with BeO moderator blocks, while the coolant is lead-bismuth. At the time of when this report was prepared, an extensive research and development programme including both analytical studies and testing was in progress for the CHTR at the Bhabha Atomic Research Centre (BARC) of India [3].

Detailed design descriptions of the abovementioned and other SMRs, as well as some results of the safety analyses performed for these reactors are provided in references [2 and 3]. Figure 2 illustrates a deployment potential of the innovative SMRs. Brown colour indicates the concepts that are manifesting noticeable progress toward advanced design stages and deployment.

1.5. Structure

The report includes an introduction, 6 chapters, 4 appendices and 10 annexes.

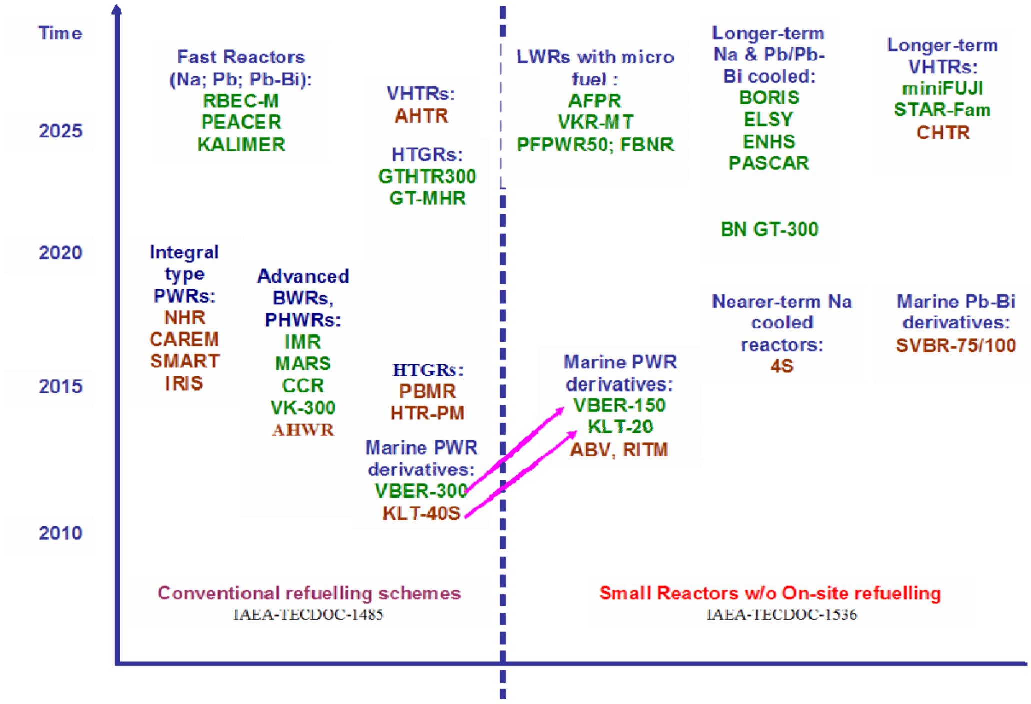

FIG. 2. Deployment potential map of innovative SMRs [2 and 3][2].

The introduction (Chapter 1) describes the background and identifies the objectives, the scope and the structure of this report, as well as the approach used in its preparation and the design status of the SMRs considered.

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[2] Brown colour is used to mark SMR concepts that show good progress towards advanced design stages, licensing, or deployment (as of 2008).

Chapter 2 provides an overview of the considerations for the incorporation of inherent and passive safety features into safety design concepts of SMRs. These considerations, presented in a generic way and, then, for each reactor line separately, were elaborated at the IAEA technical meetings in June 2005 and in October 2006.

Chapter 3 presents the design approaches applied by the designers to achieve defence in depth in SMRs. Both passive and active safety design features and systems are included in the consideration to highlight the role of the inherent and passive features and show how they may affect the design/function of the active safety systems. This chapter is based on the information and data provided by the designers of SMRs in member states and presented, in a structured form, in ANNEXES I-X to this report. The common format used to describe passive and active safety design features of SMRs is given in the Appendix IV.

Chapter 3 is structured as follows. First, a common general approach is described. After that, the description is provided for each reactor line addressed in the present report, including the pressurized water reactors, the pressurized light water cooled heavy water moderated reactors, the high temperature gas cooled reactors, the liquid metal cooled fast reactors, and the non-conventional designs. For each reactor line, a short summary of the design features of one or more of the corresponding SMRs presented in the annexes is included, followed by the summary tables and discussions of the safety design features contributing to each level of defence in depth. In this, dedicated passive and active safety systems are discussed in more detail in conjunction with defence in depth level 3. After that, summary tables and discussions follow on the lists of design basis and beyond design basis events, on the acceptance criteria, and on the features for plant protection against external event impacts. Each section winds-up with a summary table and a discussion of the measures planned in response to severe accidents.

Chapter 4 provides a review of the benefits and negative impacts in areas other than safety that in view of the SMR designers arise from the incorporation of the corresponding inherent and passive safety design features. The discussion is structured along the reactor lines considered, in the same way as Chapter 3.

Chapter 5 summarizes the approaches and considerations applied in the selection of combinations of passive and active safety systems in the considered SMRs.

Chapter 7 is a conclusion. It is elaborated as an executive summary of the report.

Appendix 1 addresses the issue of performance assessment of passive safety systems by providing a summary of the background and experience; a short description of the two methodologies for reliability assessment of passive safety systems; and a recommendation for further research and development based on the outputs of a dedicated IAEA technical meeting on June 2006. This appendix is referenced from Chapter 5.

Appendix 2 includes a paper on periodic confirmation of passive safety feature effectiveness, contributed by Dr. D.C. Wade of the Argonne National Laboratory (ANL) of the USA. This paper is referenced from Appendix 1.

Appendix 3 includes consensus and non-consensus definitions from the IAEA safety standards and other publications, relevant to the subject of this report, and also highlights some non-conventional terms used in member states.

Appendix 4 gives a common format for the description of the design features of SMRs as used in ANNEXES I–X.

ANNEXES I–X provide descriptions of the design features of the considered SMRs used to achieve defence in depth. The descriptions were contributed by member states; they are done according to the common outline given in Appendix 3. The order of the inputs corresponds to that used in Chapters 3 and 4, with pressurized water SMRs going first (ANNEXES I–V), followed by a pressurized light water cooled heavy water moderated reactor (ANNEX VI), a high temperature gas cooled reactor (ANNEX VII), the liquid metal cooled fast-spectrum SMRs (ANNEXES VIII, IX), and the non-conventional design (ANNEX X).

Contributors to drafting and review of this report are listed on the last page.

1.6. Approach

All structured descriptions of the SMR design features used to achieve defence in depth were prepared and reviewed firsthand by the designers of SMRs in member states, in communication with international experts and the IAEA secretariat.

Appendix 1 of this report was elaborated with participation of research teams involved in development of the methodologies for the reliability assessment of passive safety systems in advanced reactors.

The introductory and cross-cutting chapters were developed by international experts and the secretariat, and reviewed by SMR designers in member states. The conclusions were elaborated through the effort of the two IAEA technical meetings convened in June 2005 and in October 2006.

2. CONSIDERATIONS FOR THE INCORPORATION OF INHERENT AND PASSIVE SAFETY DESIGN FEATURES INTO SMRs

2.1. General considerations

General considerations for the incorporation of inherent and passive safety design features into SMRs are not different from those of advanced reactors of any capacity and type. Clearly, the implementation of the inherent and passive safety design features can facilitate improved defence in depth. It could also positively affect plant economy through:

- Reduced design complexity and reduced demand of human interventions resulting in fewer potentially unsafe actions;
- Reduced investment requirements, owing to the reduced qualification and operation and maintenance and, depending on specific design and regulations, reduced off-site emergency planning;
- Increased operability and capacity factors.

It is also noted that the use of inherent and passive safety features can facilitate advantages in areas other than economy, for example:

- Reduced adverse environmental impacts, e.g. through a reduced number of systems requiring maintenance and associated waste;

- Reduced vulnerability to sabotage, e.g. through semi-autonomous operation, better reactor self-control in accidents, and “passive shutdown”[3] capability;
- Deployment in developing countries, e.g., through simplified infrastructure requirements matching the limitations in human resource in these countries.

In view of the designers of SMRs, smaller capacity reactors have the following generic features, potentially contributing to a particular effectiveness of the implementation of inherent and passive safety features:

- Larger surface-to-volume ratio, which facilitates easier decay heat removal, specifically, with a single-phase coolant;
- An option to achieve compact primary coolant system design, e.g. the integral pool type primary coolant system, which could contribute to an effective suppression of certain initiating events;
- Reduced core power density, facilitating easy use of many passive features and systems, not limited to natural convection based systems;
- Lower potential hazard that generically results from lower source term owing to lower fuel inventory, lower non-nuclear energy stored in the reactor, and lower decay heat generation rate.

Section 2.2. below summarizes considerations of the SMR designers regarding those inherent and passive safety features that could be easier to achieve in a reactor of smaller capacity, for each reactor line considered in this report.

2.2. Reactor line-specific considerations

2.2.1. Pressurized water reactors

The designers of pressurized water SMRs mention cumulatively the following inherent and passive safety design features as facilitated by smaller reactor capacity and size:

- Integral design of the primary circuit with in-vessel location of the steam generators and control rod drives, to eliminate large diameter piping, minimize reactor vessel penetrations, and prevent large-break loss of coolant accidents (LOCA) and reactivity initiated accidents with control rod ejection, as well as to limit the scope of small and medium-break LOCA;
- Compact modular loop-type designs with a reduced length of piping, the integral reactor cooling system accommodating all main and auxiliary systems within a leak-tight pressure boundary, and leak restriction devices; altogether, to prevent LOCA or limit their scope and hazard ;
- A design with primary pressure boundary enclosed in a enveloping shell with low enthalpy slowly moving water, intended to prevent LOCA or limit their scope and hazard ;

[3] Throughout this report, ‘passive shutdown’ is used to denote bringing the reactor to a safe low-power state with balanced heat production and passive heat removal, with no failure to the barriers preventing radioactivity release to the environment; all relying on the inherent and passive safety features only, with no operator intervention, no active safety systems being involved, and no external power and water supplies being necessary, and with the grace period infinite for practical purpose.

- Increased thermal inertia at a reasonable reactor vessel size, contributing to long response time in transients and accidents;
- Enhanced level of natural convection, sufficient to remove decay heat passively from a shutdown reactor over an indefinite time;
- In-vessel retention of core melt, e.g., through passive external cooling of the reactor pressure vessel;
- Compact design of the primary circuit and the containment, to facilitate protection against missiles and aircraft crash.

2.2.2. Pressurized light water cooled heavy water moderated reactors

For a boiling light water cooled / heavy water moderated reactor considered in the present report (the AHWR, incorporating the pressure channels and the calandria; see ANNEX VI), smaller capacity, in view of the designers, facilitates:

- The use of natural convection for heat removal in normal operation, e.g., elimination of main circulation pumps;
- Achieving a slightly negative void coefficient of reactivity;
- Providing a relatively large coolant inventory in the main coolant system, to ensure its high thermal inertia and slow pace of the transients;
- Providing a relatively large inventory of water in a reasonably-sized gravity driven water pool (GDWP), located inside the containment and intended for passive emergency injection of the cooling water, passive containment cooling, and passive decay heat removal via the isolation condensers.

2.2.3. High temperature gas cooled reactors

For high temperatures gas cooled reactors (HTGRs) with pebble bed or pin-in-block tristructural-isotropic (TRISO) fuel and helium coolant, smaller reactor capacity facilitates:

- Long-term passive decay heat removal from the core to the outside of the reactor vessel based on natural processes of conduction, radiation and convection, with natural convection based heat removal from the outside of the reactor vessel to an ultimate heat sink;
- Achieving a large temperature margin between the operation limit and the safe operation limit, owing to inherent fission product confinement properties of the TRISO fuel at high temperatures and fuel burn-ups;
- De-rating of accident scenarios rated as potentially severe in reactors of other types, including loss of coolant (LOCA), loss of flow (LOFA), and reactivity initiated accidents; for example, helium release from the core in the GT-MHR can be a safety action and not an initiating event of a potentially severe accident;
- Achieving increased reactor self-control in anticipated transients without scram, without exceeding the safe operation limits for fuel;
- Relatively high heat capacity of the reactor core and the reactor internals and low core power density, resulting in slow progression of the transients.

It should be noted that, in view of the reactor vessel materials known currently, the HTGR unit capacity below ~600 MW(th) is a necessary condition to ensure long-term passive decay heat removal from the core as described in the first bullet of this sub-section. Therefore, all currently known concepts of HTGR with TRISO based fuel and gas coolant belong to the SMR range [2].

2.2.4. Sodium cooled and lead cooled fast reactors

For both, sodium cooled and lead cooled fast reactors, smaller unit capacity could facilitate:

- Effective use of auxiliary passive decay heat removal systems with the environmental air in natural draught acting as an ultimate heat sink;
- Achieving a relatively high heat capacity of the primary (or primary and adjacent intermediate) coolant system at its reasonable size, resulting in a slower progression of transients.

Specifically for sodium cooled fast reactors, smaller reactor capacity could facilitate achieving a negative whole-core void coefficient of reactivity to prevent the progression of design basis accidents into severe ones, otherwise possible at a start of sodium boiling.

Specifically for lead cooled fast reactors, smaller reactor capacity could facilitate simplified seismic protection and improved seismic response [2].

2.2.5. Non-conventional designs

The only non-conventional reactor concept considered in this report, the Compact High Temperature Reactor (CHTR) of BARC (India), is based on a synthesis of the technology of ^{233}U-Th HTGR type pin-in-block fuel and that of a lead-bismuth coolant; see ANNEX X. The CHTR is a very high temperature reactor concept. Smaller reactor capacity facilitates:

- Passive heat removal from the core in normal operation, with no main circulation pumps being employed; as well as passive and passively actuated heat removal from the core during and after the accidents, including those based on the use of heat pipe systems;
- Relatively high heat capacity of the ceramic core, resulting in slow temperature transients, at a reasonable reactor size;
- Prevention of the consequences of transient overpower events;
- Passive power regulation and increased reactor self-control in transients without scram.

3. DESIGN APPROACHES TO ACHIEVE DEFENCE IN DEPTH IN SMRs

3.1. General approach

In SMR designs, as in larger reactor designs, defence in depth strategy is used to protect the public and environment from accidental radiation releases. Nearly all SMR designs seek to strengthen the first and subsequent levels of defence by incorporating inherent and passive safety features. Certain common characteristics of smaller reactors lend themselves to inherent and passive safety features, such as relatively smaller core sizes enabling integral coolant system layouts and larger reactor surface-to-volume ratios or lower core power densities which facilitate passive decay heat removal. Using the benefit of such features, the first goal is to eliminate or prevent, by design, as many accident initiators and accident consequences as possible. Remaining plausible accident initiators and consequences are then addressed by appropriate combinations of active and passive safety systems. The intended outcome is greater plant simplicity with high safety levels that, in turn, might allow reduced emergency requirements off-site.

It should be noted that an approach to maximize the use of inherent safety features in order to minimize the number of accident initiators in a reactor concept, and then to deal with the remaining accidents using reasonable combinations of the active and passive safety systems is pursued by the Generation IV International Forum, in line with the Generation IV Technology Goals [15]. To a limited extent, such approach is also realized in several near-term designs of large-capacity water cooled reactors, such as the AP1000, the ESBWR, and the VVER1000, the goal being to achieve a high level of safety in a cost effective way [4].

3.2. Approaches for specific reactor lines

For each of the reactor lines considered (pressurized water reactors, pressurized light water cooled heavy water moderated reactors, high temperature gas cooled reactors, sodium cooled and lead cooled fast reactors, and non-conventional designs), the design features contributing to different levels of defence in depth are summarized and structured in the following way.

The first five tables for each reactor line give a summary of the design features contributing to Level 1 through Level 5 of the defence in depth with a short explanation of the nature of these contributions, in line with the definitions given in [7]. Passive and active safety systems are highlighted in more detail in conjunction with Level 3 of defence in depth.

It should be noted that original safety design concepts of the considered SMRs do not always follow the defence in depth concept recommended in the IAEA safety standard [7]. Although all designers were requested to follow the recommendations of [7] when providing the descriptions of SMR safety design features enclosed as ANNEXES I–X, the results turned out to be non-uniform, for example, some Level 4 features were in several cases attributed to Level 5 for PWRs, etc. To provide a uniform basis for the description, the attribution of safety design features to certain levels of defence in depth was harmonized for all SMRs considered, following the recommendations of [7], and in this way presented in all tables of this section. Therefore, the attribution indicated in the tables below may be in some cases different from that originally provided by designers in the corresponding annexes.

The sixth table for each reactor line summarizes the degree of detail in the definition of design basis and beyond design basis events, as observed in the corresponding annexes, and highlights the events that are specific to a particular SMR but not to the corresponding reactor line.

The seventh table gives a summary of deterministic and probabilistic acceptance criteria for design basis and beyond design basis events, as applied by the designers, and specifically highlights the cases when a risk-informed approach is being used or targeted.

The eighth table for each reactor line summarizes design features for plant protection against external event impacts, with a focus on aircraft crash and earthquakes, and gives a reference to the recent IAEA publication of relevance [6], when applicable.

Finally, the ninth table gives a summary of measures planned in response to severe accidents.

The final paragraph in each of the following subsections provides a summary of safety design approaches pursued by the designers of SMRs, using the above mentioned tables as reference, with a link to the IAEA safety standard [7] and other publications of relevance.

3.2.1. Pressurized water reactors

The pressurized water small and medium sized reactors are represented by three concepts using integral layout of the primary circuit with in-vessel location of the steam generators and control rod drives; one compact modular loop-type design with a reduced length of piping, the integral reactor cooling system accommodating all main and auxiliary systems within a leak-tight pressure boundary, and leak restriction devices; and one design originating from the mid 1980s, in which the primary pressure boundary enclosed in a enveloping shell with low-enthalpy slowly moving water.

The concepts with integral primary circuit layout include the CAREM-25 of 27 MW(e), a prototype for a series of larger capacity SMRs being developed by the CNEA (Argentina), the IRIS of 335 MW(e) being developed by the international consortium led by Westinghouse (USA), and the SCOR concept of 630 MW(e) being developed by the CEA (France). The CAREM-25 and the IRIS have reached detailed design stages with deployments being targeted for 2011 and 2015 respectively, while the SCOR is just a conceptual design. Detailed design descriptions of the CAREM-25, IRIS, and SCOR are presented in [2], and the corresponding structured descriptions of their passive safety design features are given in ANNEXES II, III, and IV, correspondingly. Figure 3 below provides an illustration of the primary coolant system layout for the indicated designs.

The compact modular loop-type concepts are represented by the KLT-40S, a 35 MW(e)/150 MW(th) reactor for a twin-unit floating heat and power plant, which was started in construction in the Russian Federation in April 2007. The power circuits of the two units are separate with each producing more heat power than is required to generate the rated electrical output; the remaining heat power will be used for district heating (as provided for in the “Lomonosov” first-of-a-kind floating nuclear power plant, under construction in Russia) or for seawater desalination (as it is foreseen for future units to be deployed outside of the Russian Federation). Detailed description of the KLT-40S design, developed by the OKBM and several other Russian organizations, is provided in [4]; a structured design description of the passive safety design features is given in ANNEX I. The IAEA publications [2 and 3] provide the descriptions of several other reactors for floating as well as land-based NPPs, employing the design concept similar to that of the KLT-40S. Layout of the KLT-40S reactor is shown in Fig. 4.

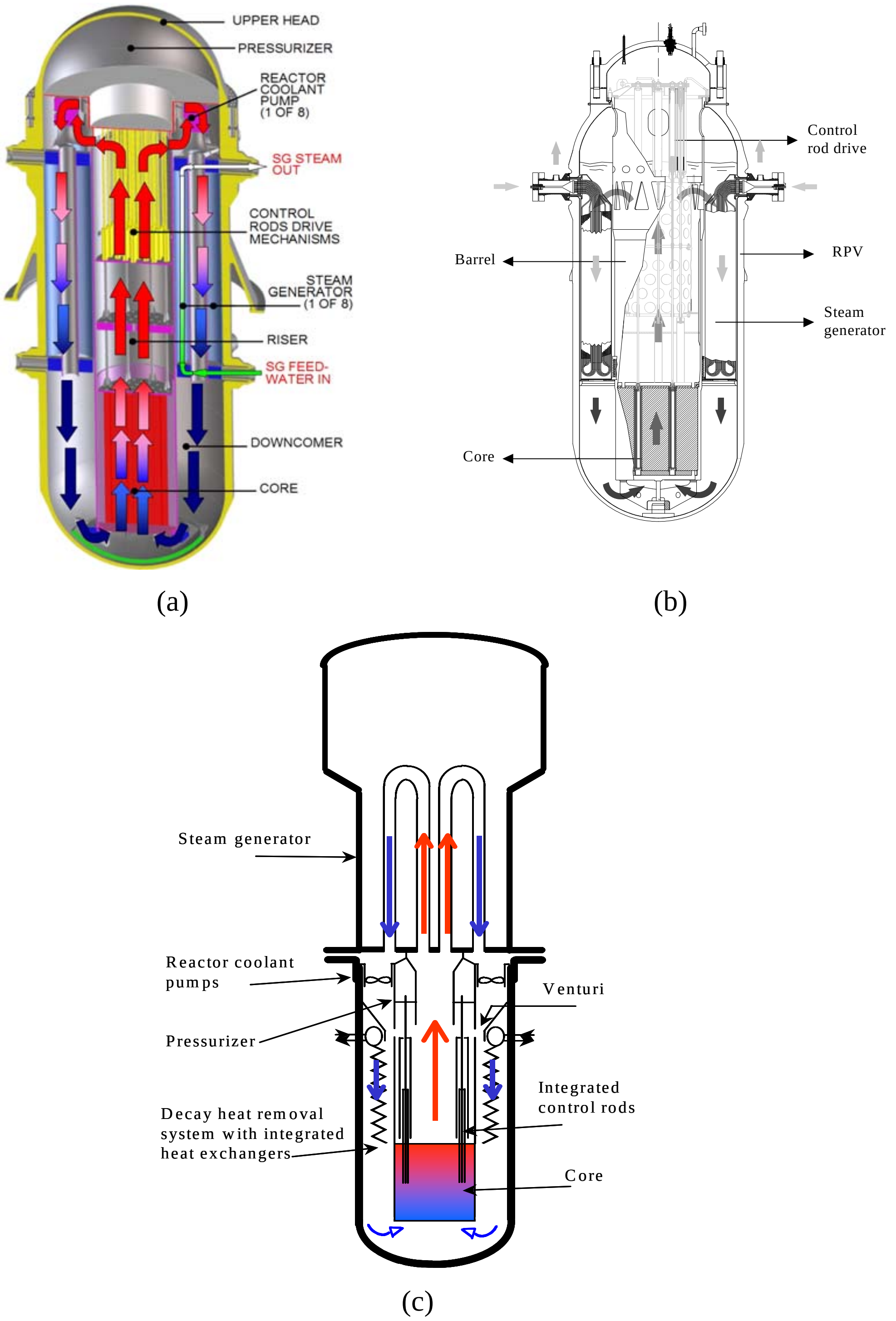
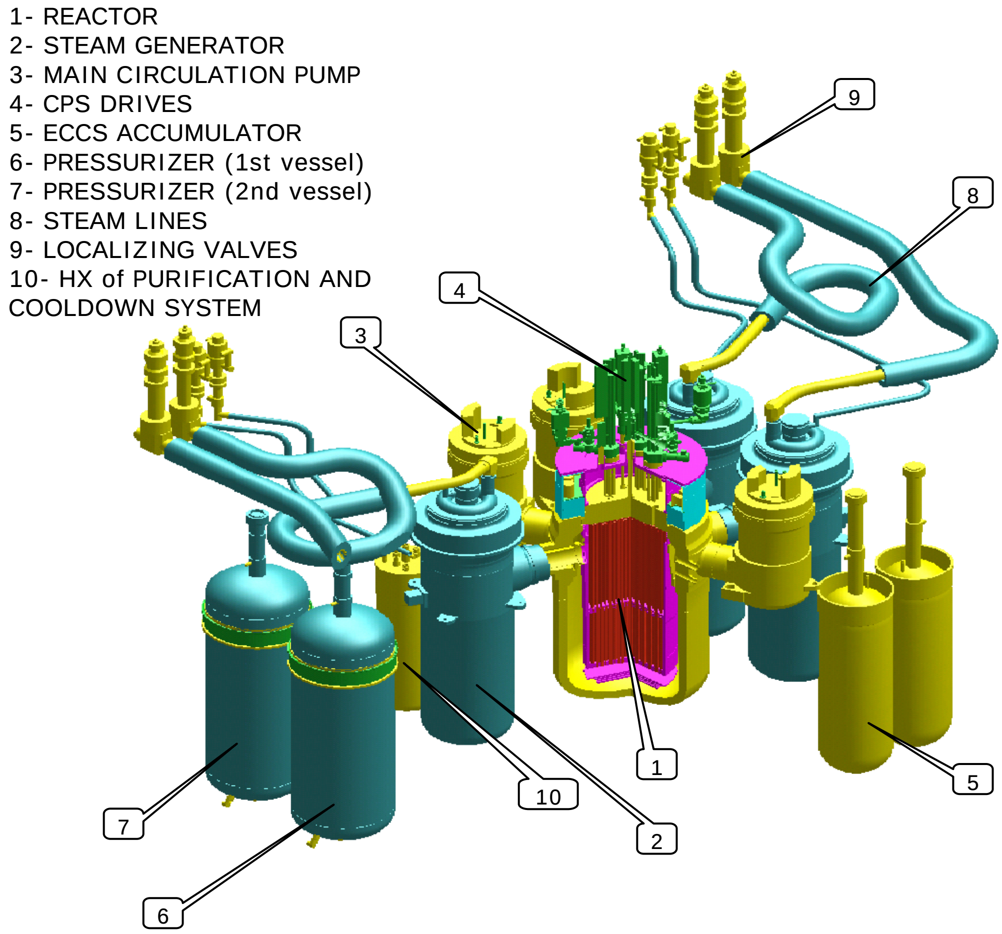


FIG. 3. Schematics of the primary coolant system for (a) IRIS; (b) CAREM-25; and (c) SCOR.

The MARS reactor of 150 MW(e) per module, in which the primary pressure boundary is enclosed in a pressurized low-enthalpy containment, was developed by a consortia of the academic, research and industrial organizations in Italy. The detailed design stage was reached, and several testing programmes were completed. Design description of the MARS is presented in [2]; passive safety design features of the MARS are described in ANNEX V. Layout of the MARS primary coolant system is shown in Fig. 5



CPS - control and protection system ECCS – emergency core cooling system HX – heat exchanger

FIG. 4. Layout of the KLT-40S reactor.

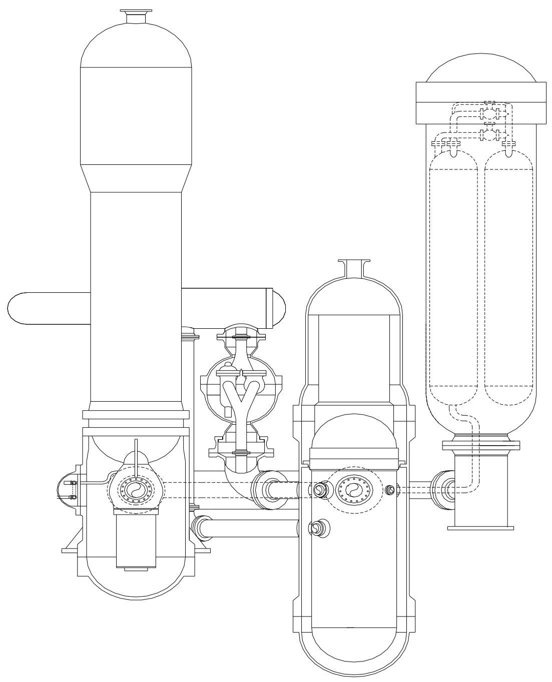


FIG. 5. Layout of the MARS reactor with pressurized containment for primary loop protection.

Design features of pressurized water SMRs contributing to the enhancement of Level 1 of defence in depth are summarized in Table 1; for the subsequent levels — in Tables 2, 3, 4 and 5, respectively.

TABLE 1. DESIGN FEATURES OF PRESSURIZED WATER SMR CONCEPTS CONTRIBUTING TO LEVEL 1 OF DEFENCE IN DEPTH

#	DESIGN FEATURES	WHAT IS TARGETED	SMR DESIGNS
1	Elimination of liquid boron reactivity control system	Exclusion of inadvertent reactivity insertion as a result of boron dilution	KLT-40S, CAREM-25, SCOR
2	Relatively low core power density	Larger thermal-hydraulic margins	MARS, IRIS, CAREM-25, SCOR
3	Integral design of primary circuit with in-vessel location of steam generators and (hydraulic) control rod drive mechanisms	Exclusion of large-break loss of coolant accidents (LOCA), exclusion of inadvertent control rod ejection; larger coolant inventory and thermal inertia	CAREM-25, IRIS, SCOR
4	Compact modular design of the reactor unit, eliminating long pipelines in the reactor coolant system	Decreased probability of LOCA	KLT-40S
5	Primary pressure boundary enclosed in a pressurized, low-enthalpy containment	Elimination of LOCA resulting from failure of the primary coolant pressure boundary, elimination of control rod ejection accidents	MARS
6	Leak-tight reactor coolant system (welded joints, packless canned pumps, and leak-tight bellows-sealed valves, etc.)	Decreased probability of LOCA	KLT-40S
7	Internal, fully immersed pumps	Elimination of pump seizure, rotor lock, and seal LOCA	MARS, IRIS, SCOR
8	Leak restriction devices in the primary pipelines	Limitation of the break flow in case of a pipeline guillotine rupture	KLT-40S
9	A single, small-diameter double connecting line between the primary coolant pressure boundary and the auxiliary systems	Prevention of LOCA caused by rupture of the connecting line	MARS
10	Natural circulation based heat removal from the core in normal operation, eliminating main circulation pumps	Elimination of loss of flow accidents (LOFA)	CAREM-25
11	Steam generator with lower pressure inside the tubes in a normal operation mode	Reduced probability of a steam tube rupture; prevention or downgrading of a steam-line break or a feed-line break	MARS, KLT-40S, IRIS
12	Steam generator designed for a full primary system pressure	Prevention or downgrading of a steam-line break or a feed-line break	IRIS, MARS

At Level 1 of defence in depth, “Prevention of abnormal operation and failure”, the dominant tendency is to exclude loss of coolant accidents (LOCA) or limit their scope and hazard by applying certain features in the reactor design, such as:

- In-vessel location of steam generators in PWRs with integral design of the primary circuit (CAREM-25, IRIS, SCOR), allowing to eliminate large diameter piping and, hence, large-break LOCA;
- In-vessel location of the control rod drive mechanism (CAREM-25, IRIS, SCOR), which allows to reduce the number and the diameters of necessary in-vessel penetrations;
- Compact modular design of the reactor unit, eliminating long pipelines in the reactor coolant system, leak restriction devices in the primary pipelines, and a so-called “leak-tight” reactor coolant system with packless canned pumps, welded joints, and leak-tight bellows-sealed valves (KLT-40S, based on the submarine and icebreaker reactor experience); internal, fully immersed pumps are also applied in the IRIS and the SCOR reactors with integral design of the primary circuit;
- Primary pressure boundary enclosed in a pressurized, low-enthalpy containment (a shell) with only a single, small-diameter pipeline between the primary coolant pressure boundary and the auxiliary systems (MARS).

As it was already mentioned, all PWRs with integral design of the primary circuit incorporate in-vessel control rod drives, which is not only a design feature to minimize reactor vessel penetration but is intended primarily to exclude reactivity initiated accidents with inadvertent control rod excursion (otherwise potentially facilitated by high primary pressure). Integral design of the primary circuit with in-vessel location of the steam generators and the control rod drives[4] apparently necessitates using a relatively low core power density, which in turn contributes to providing larger thermal-hydraulic margins.

Elimination of liquid boron reactivity control, which facilitates prevention of inadvertent reactivity excursion as the result of boron dilution, can not be attributed to a certain class of reactor concepts; it is applied in the KLT-40S and the CAREM-25 but is not applied in other concepts considered.

Finally, the use of natural convection for heat removal in normal operation, which allows to eliminate loss of flow accidents owing to pump failure, is not a preferable feature of PWR type small and medium sized reactors — it is applied only in the small-powered CAREM-25 design (of 27 MW(e)).

Four of the considered reactors mention design features applied to prevent steam generator tube rupture, see Table 1. The KLT-40S, the MARS and the IRIS apply steam generators with lower pressure inside the tubes in normal operation mode; again in the IRIS and in the MARS, steam generators are designed for a full primary system pressure.

All in all, PWRs with integral design of the primary circuit show a tangible and transparent approach to the elimination of several accident initiators by design. The question of whether this could be applied only to reactors within the small to medium

[4] Some PWRs use primary circuit with internal steam generators but have external control rod drives, i.e., SMART of the Republic of Korea [2].

power range is, however, open. For example, the French SCOR is as powerful as 630 MW(e), which is credited to the steam generator of an original design borrowing from the experience of the marine propulsion reactors [2]. A recent paper on SCOR [16] points to the option to develop a PWR of integral design as powerful as 1000 MW(e). In the latter case, however, the reactor vessel height exceeds 30 m (actually, two vertically adjusted half-vessels are used in the SCOR). It should also be noted that the SCOR design is at a conceptual design stage, while the IRIS and CAREM-25 have reached detailed design stages.

TABLE 2. DESIGN FEATURES OF PRESSURIZED WATER SMR CONCEPTS CONTRIBUTING TO LEVEL 2 OF DEFENCE IN DEPTH

#	Design feature	What is targeted	SMR designs
1	Active systems of instrumentation and control	Timely detection of the abnormal operation and failures	All designs
2	Negative reactivity coefficients over the whole cycle	Preventing transient over-criticality due to abnormal operation and failures	All designs
3	A relatively large coolant inventory in the primary circuit, resulting in large thermal inertia	Slow progression of transients due to abnormal operation and failures	CAREM-25, SCOR, IRIS, MARS
4	High heat capacity of nuclear installation as a whole	Slow progression of transients due to abnormal operation and failures	KLT-40S
5	Favourable condition for the implementation of leak before break concept, provided by the design of the primary circuit	Facilitate implementation of leak before break concept	KLT-40S
6	Small coolant flow in the low temperature pressurized water containment enclosing the primary pressure boundary	Facilitate implementation of leak before break concept	MARS
7	Redundant and diverse passive or active shutdown systems	Reactor shutdown	All designs

At Level 2 of defence in depth, “Control of abnormal operation and detection of failure”, active systems of instrumentation and control and negative reactivity coefficients over the whole burn-up cycle are common to all designs. These are the features typical of all state-of-the-art reactor designs, independent of their unit power range.

A relatively large coolant inventory in the primary circuit and high heat capacity of nuclear installation as a whole, resulting from an integral (IRIS, CAREM-25, SCOR) or compact modular (KLT-40S) design of the nuclear installation, are factors contributing to large thermal inertia and slow pace of the transients, altogether allowing to gain more time for failure detection or corrective actions. Larger coolant inventory and higher heat capacity of the primary circuit are related to relatively larger reactor vessel and internals or lower core power density as compared to a typical large-sized PWR.

Compact modular design of the reactor unit, eliminating long pipelines in the reactor coolant system, with leak restriction devices in the primary pipelines and a so-called “leak-tight” reactor coolant system with packless canned pumps, welded joints, and leak-tight bellows-sealed valves, implemented in the KLT-40S, are mentioned as factors contributing to effective realization of leak before break concept. In the MARS design, implementation of leak before break is facilitated by maintaining a small coolant flow in the low-temperature pressurized water shell (containment) enclosing the primary pressure boundary.

Finally, redundant and diverse passive or active shutdown systems are provided in all designs for the cases when abnormal operation tends to be out of control or the source of failure is not detected timely and adequately.

TABLE 3. DESIGN FEATURES OF PRESSURIZED WATER SMR CONCEPTS CONTRIBUTING TO LEVEL 3 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
1	Negative reactivity coefficients over the whole cycle	Preventing transient over-criticality and bringing the reactor to a sub-critical state in design basis accidents	All designs
2	Relatively low core power density	Larger thermal-hydraulic margins	MARS, IRIS, CAREM-25, SCOR
3	Relatively low primary coolant temperature	Larger thermal-hydraulic margins	MARS
4	A relatively large coolant inventory in the primary circuit (or primary circuit and the pressurized low-enthalpy containment, enclosing the primary pressure boundary; or primary circuit and the reactor building), resulting in a large thermal inertia	Slow progression of transients in design basis accidents	CAREM-25, SCOR, IRIS, MARS
5	High heat capacity of nuclear installation as a whole	Limitation of temperature increase in design basis accidents	KLT-40S
6	Restriction devices in pipelines of the primary circuit; with primary pipelines being connected to the hot part of the reactor	Limitation of the scope and slowing the progression of LOCA	KLT-40S
7	Use of once-through steam generators	Limitation of heat rate removal in a steam line break accident	KLT-40S
8	Steam generator designed for full primary pressure	Limitation of the scope of a steam generator tube rupture accident	IRIS, MARS
9	A dedicated steam dump pool located in the containment building	Prevention of steam release to the atmosphere in the case of a steam generator tube rupture	SCOR
10	The relief tank of a steam generator safety valve enclosed in a low temperature pressurized water containment enclosing the primary pressure boundary	Prevention of steam release to the atmosphere in the case of a steam generator tube rupture	MARS

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
11	'Soft' pressurizer system[5]	Damping pressure perturbations in design basis accidents	KLT-40S
12	Self-pressurization; large pressurizer volume; elimination of sprinklers, etc.	Damping pressure perturbations in design basis accidents	CAREM-25, IRIS, SCOR
13	Limitation of inadvertent control rod movement by an overrunning clutch and by the limiters	Limitation of the scope of reactivity insertion in an accident with the control rod drive bar beak	KLT-40S
14	Redundant and diverse reactor shutdown and heat removal systems	Increased reliability in carrying out the safety function	All designs
15	Insertion of control rods to the core, driven by gravity	Reactor shutdown	KLT-40S, CAREM-25
16	Insertion of control rods to the core, driven by force of springs	Reactor shutdown	KLT-40S
17	Non safety-grade control rod system with internal control rod drives	Reactor shutdown	IRIS
18	One of the shutdown systems based on gravity driven insertion of control rods to the core	Reactor shutdown	SCOR
19	Safety-grade active mechanical control rod scram system	Reactor shutdown	MARS
20	Additional (optional) passive scram system actuated by a bimetallic core temperature sensor and operated by gravity	Reactor shutdown	MARS
21	Gravity-driven high-pressure borated water injection device (as a second shutdown system)	Reactor shutdown	CAREM-25
22	Injection of some borated water from the emergency boron tank at high pressure(as an auxiliary shutdown measure)	Reactor shutdown	IRIS
23	Active safety injection system based on devices with a small flow rate	Reactor shutdown	SCOR
24	Emergency injection system (with borated water), actuated by rupture disks	Reactor shutdown plus prevention of core uncovery in LOCA	CAREM-25
25	Natural convection core cooling in all modes	Passive heat removal	CAREM-25

[5] "Soft" pressurizer system is characterized by small changes of the primary pressure under a primary coolant temperature increase. This quality, due to a large volume of gas in the pressurizing system, results in an increased period of pressure increase up to the limit value under the total loss of heat removal from the primary circuit.

#	Design feature	What is targeted	SMR designs
26	Natural convection level in the primary circuit with operating passive residual heat removal systems sufficient to remove decay heat under a station blackout	Passive heat removal	IRIS, SCOR
27	Level of natural circulation sufficient for adequate core cooling in a condition with all main circulation pumps switched off	Passive heat removal	KLT-40S
28	Passive emergency (or residual) core heat removal system with natural convection of the coolant in all circuits, with water evaporation in the water (e.g., storage) tanks	Passive decay heat removal	KLT-40S, IRIS, CAREM-25
29	Residual heat removal through the steam generator. The steam is discharged to the atmosphere, and the steam generator is fed by the start-up shutdown system (SSS). This system is not safety grade.	Passive decay heat removal	SCOR
30	Redundant passive residual heat removal systems on the primary circuit with two diverse heat sinks; infinite autonomy achieved with the air-cooling tower heat sink	Passive decay heat removal	SCOR
31	Passive emergency core cooling system with the infinite grace period, using natural draught of air as an ultimate heat sink; actuated upon flow rate decrease	Passive decay heat removal	MARS
32	Decay heat removal through a steam line of the steam generator, requiring no action to be initiated	Passive decay heat removal	SCOR
33	A small automatic depressurization system from the pressurizer steam space	Depressurization of the reactor vessel when in-vessel coolant inventory drops below a specified level	IRIS
34	Safety (relief) valves	Protection of reactor vessel from over-pressurization	IRIS, CAREM-25
35	Long-term gravity make-up system	Assures that the core remains covered indefinitely following a LOCA	IRIS
36	Emergency injection system (with borated water), actuated by rupture disks	Prevention of core uncovery in LOCA	CAREM-25

As it has been discussed above, certain design features provided at Level 1 of defence in depth in PWR type SMRs contribute to prevention or de-rating of certain design basis accidents, such as large-break or medium break LOCA, core uncovery in LOCA, steam generator tube rupture, reactivity accidents with inadvertent ejection of a control rod, or loss of flow, narrowing the scope of events to be dealt with at Level 3 of defence in depth, “Control of accidents within design basis”. For the remaining events, a variety of design features is specified at Level 3; altogether, these features fit into the following main groups:

(1) Inherent safety features provided by design and contributing to larger thermal margins, lower parameter variation, better reactor self-control, slower pace of the transients, and damping of perturbations in design basis events. These features are highlighted in positions 1–13 of Table 3;

(2) All designs incorporate at least two redundant and diverse shutdown systems; see positions 14–24 of Table 3. These systems may be passive, such as mechanical control rods inserted into the core driven by gravity or by the force of springs, or active, such as standard mechanical control rods. Some passive systems are passively actuated, e.g., by system de-energization, or by core temperature sensor, etc. The role of safety injection systems with borated water is essentially reduced in some cases, e.g., in the IRIS and SCOR, or the function of a safety injection is coupled with core uncovery prevention, e.g., in the CAREM-25. Safety injection may be passive (IRIS) or active (SCOR); it may also be actuated passively, by disk rupture under an over-pressure (CAREM-25). For some designs (KLT-40S), safety injection of borated water is not indicated at all;

(3) All pressurized water SMRs incorporate passive residual heat removal systems of various design, often redundant, based on natural convection of the coolant; see positions 25–32 of Table 3. Such features of PWR type SMRs as reduced core power density or relatively large coolant inventory in the primary circuit, or taller reactor vessel, discussed in more details above, in conjunction with levels 1 and 2 of defence in depth, contribute to passive residual heat removal that is effective under a total power station blackout, with a increased or practically infinite grace period. It could be emphasized that all decay heat removal systems in all PWR type SMRs are passive, and most of them require no operator action to get actuated;

(4) Finally, positions 33–36 of Table 3 indicate design features/ systems dedicated to prevention of core uncovery in design basis accidents. These may include automatic depressurization systems, safety relief valves, long-term gravity make-up systems and emergency boron injection systems also acting as make-up systems. All of the indicated systems are passive and passively actuated.

The approaches for using safety grade/ non safety grade systems vary between different SMR concepts.

In the IRIS (ANNEX II), all passive safety systems are safety grade; all safety grade systems are passive. For example, refuelling water storage tank is safety grade. All active systems are non safety grade.

In the CAREM-25 (ANNEX III), all safety systems are passive and safety grade; auxiliary active systems are safety grade also.

In the SCOR (ANNEX IV), redundant residual heat removal systems on the primary coolant system with pool as a heat sink (RRPp) are safety grade; similar-designation systems with air as a heat sink (RRPa) are safety grade, except for the chilled water pool and pumps. The start-up shutdown system is non-safety grade. Safety injection system is the only active safety system that is safety grade. In the case of a steam generator line rupture, there is no need in safety grade auxiliary feedwater system, because normal operation systems are used in this case.

In the MARS (ANNEX V), all nuclear components of the reactor core are safety grade. CPP – the enveloping primary circuit boundary – is non-safety grade. The hydraulic connections to the primary coolant boundary are safety grade. The steam generator tubes are safety grade. The containment building is safety grade. SCCS — the passive core cooling system — is safety grade. The optional passive scram system is safety grade, as well as the active scram system.

No information on the grade of safety systems was provided for the KLT-40S.

TABLE 4. DESIGN FEATURES OF PRESSURIZED WATER SMR CONCEPTS CONTRIBUTING TO LEVEL 4 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
1	Relatively low core power density	Limitation or postponing of core melting	IRIS, CAREM-25, SCOR, MARS
2	Relatively low temperature of reactor coolant	Limitation or postponing of core melting	MARS
3	Low heat-up rates of fuel elements predicted in a hypothetical event of core uncovery, owing to the design features	Prevention of core melting due to core uncovery	CAREM-25
4	Low-enthalpy pressurized water containment embedding the primary pressure boundary	Additional barrier on the way of possible radioactivity release to the environment	MARS
5	Passive emergency core cooling, often with increased redundancy and grace period (up to the infinite in time)	Provision of a sufficient time for accident management, e.g., in the case of failure of active emergency core cooling systems	KLT-40S, IRIS, CAREM,-25 SCOR, MARS
6	Passive system of reactor vessel bottom cooling	In-vessel retention of core melt	KLT-40S
7	Natural convection of water in the flooded reactor cavity	In-vessel retention of core melt	SCOR
8	Passive flooding of the reactor cavity following a small LOCA	Prevention of core melting due to core uncovery; in vessel retention	IRIS
9	Flooding of the reactor cavity, dedicated pool for steam condensation under a steam generator tube rupture	Reduction of radioactivity release to the environment due to increased retention of fission products	SCOR

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
10	Containment and protective enclosure (shell) or double containment	Prevention of radioactive release in severe accidents; protection against external event impacts (aircraft crash, missiles)	KLT-40S, IRIS, CAREM,-25 MARS
11	Containment building	Prevention of radioactive release in severe accidents; protection against external event impacts (aircraft crash, missiles)	All designs
12	Very low leakage containment; elimination /reduction of containment vessel penetrations	Prevention of radioactivity release to the environment	IRIS
13	Reasonably oversized reactor building, in addition to the primary coolant pressure boundary and the additional water filled pressurized containment	Prevention of radioactivity release to the environment in unforeseen LOCA and severe accidents (LOCAs are prevented by design by the CPP	MARS
14	Indirect core cooling via containment cooling	Prevention of core melting; in-vessel retention	IRIS
15	Passive containment cooling system	Reduction of containment pressure and limitation of radioactivity release	KLT-40S
16	Relatively small, inerted, pressure suppression containment	Prevention of hydrogen combustion	SCOR
17	Inerted containment	Prevention of hydrogen combustion	IRIS
18	Reduction of hydrogen concentration in the containment by catalytic recombiners and selectively located igniters	Prevention of hydrogen combustion	CAREM-25
19	Sufficient floor space for cooling of the molten debris; extra layers of concrete to avoid containment basement exposure directly to the debris	Prevention of radioactivity release to the environment	CAREM-25

The design features of PWR type SMRs contributing to Level 4 of defence in depth, “Control of severe plant conditions, including prevention of accident progression and mitigation of consequences of severe accidents”, could be categorized as follows:

(1) Inherent or passive safety features, provided by design, contributing to the limitation or postponing of core melting, or the prevention of core melting due to core uncovery, or providing additional barriers on the way of possible radioactivity release to the environment. These are highlighted in positions 1–4 of Table 4;

(2) Passive emergency core cooling systems, often redundant and offering an increased grace period up to the infinite autonomy. These are intended to provide a sufficient time for accident management. Passive emergency core cooling systems and passive decay heat removal systems are highlighted in more detail in Table 3;

(3) Passive systems of reactor vessel cooling based on natural convection of water in a flooded reactor cavity, intended to secure in-vessel retention of the corium; see positions 6–9 of Table 4. It should be noted that such features of smaller reactors as reduced core power density or relatively larger or taller reactor vessel, discussed above in conjunction with Level 1 of defence in depth, facilitate effective in-vessel retention of corium and allow to exclude core catchers from the reactor design;

(4) Containment buildings, in most cases a containment and a protective shell or a double containment, typical of all PWR type SMRs, are highlighted in positions 10–13 of Table 4. Like in reactors of other types and capacity, these are intended to prevent radioactivity release to the environment in severe accidents, and are also designed to provide protection against the impacts of external events (discussed later in this section). The containments for PWR type SMRs are more compact than for large-sized PWRs, providing a smaller target for external missile of an aircraft. However, they could be made reasonably oversized to confine hydrogen and other gaseous products in the case of a severe accident;

(5) Design features to prevent hydrogen combustion of limit hydrogen concentration inside the containment; see positions 16–18 of Table 4;

(6) In one design, the CAREM-25, sufficient floor space for cooling of the molten debris and extra layers of concrete to avoid containment basement exposure directly to the debris provides a kind of a substitute for the core catcher.

For Level 5 of defence in depth, “Mitigation of radiological consequences of significant release of radioactive materials”, the designers of several PWR type SMRs considered in the present report mention smaller source terms, possibly resulting from relatively smaller fuel inventory, smaller non-nuclear energy stored in the reactor, and smaller integral decay heat rates as compared to a typical large-sized PWR, see Table 5. The designers also suggest that design features of Levels 1–4 of defence in depth could be sufficient to achieve the goal of the defence in depth Level 5. However, such suggestion needs to be proven and accepted by the regulators, which was not the case at the time when this report has been prepared. Certain activities of PWR type SMR designers targeted at proving the option of a reduced emergency planning zone were, however, in progress. One such activity, generic for many innovative SMRs, is being carried out within the IAEA Coordinated Research Project “Small Reactors without On-site Refuelling”, on the example of the IRIS reactor.

Table 6 summarizes the information on design basis and beyond design basis events provided by the designers of PWR type SMRs in ANNEXES I–V, and highlights the events that are specific for a given SMR but not for a generic PWR reactor line, where applicable. De facto, such events are mentioned only for the KLT-40S, for which two groups of specific events are specified, the first two related to the ‘soft’ pressurizer system operated by gas from a gas balloon, and the latter five specific of a floating (barge-mounted) NPP. For the IRIS design version considered for future licensing without off-site emergency planning, a consideration of such rare hypothetical events as rupture of the reactor vessel and failure of all safety systems is made. It should be noted that this will not be the case for the first-of-a-kind plant licensing. In several cases, a qualitative comparison of the progression of transients in a given SMR and in a typical PWR is provided; see ANNEXES I–V for details.

TABLE 5. DESIGN FEATURES AND MEASURES OF PRESSURIZED WATER SMR CONCEPTS CONTRIBUTING TO LEVEL 5 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
1	Mainly administrative measures	Mitigation of radiological consequences of significant release of radioactive materials	KLT-40S
2	Relatively small fuel inventory, smaller non-nuclear energy stored in the reactor, and smaller integral decay heat rate	Smaller source term	Several designs
3	Design features of levels 1–4 could be sufficient to achieve the goal of defence-in-depth Level 5[6]	Exclusion of a significant release of radioactive materials beyond the plant boundary or essential reduction of the zone of off-site emergency planning	KLT-40S, IRIS, CAREM,-25 MARS, SCOR

TABLE 6. SUMMARY OF DESIGN BASIS AND BEYOND DESIGN BASIS EVENTS, INCLUDING THOSE SPECIFIC FOR A PARTICULAR SMR

#	SMR DESIGN	LISTS OF INITIATING EVENTS	EVENTS SPECIFIC TO A PARTICULAR SMR
1	KLT-40S	Detailed lists of initiating events for abnormal operation occurrences (AOO), design basis accidents (DBA), and beyond design basis accidents (BDBA) are presented (ANNEX I)	(1) Disconnection of the gas balloons from the pressurizer during power operation (2) Rupture of a pipeline connecting the gas balloon to the pressurizer (3) Explosion of the gas balloons (4) Collision with another ship (5) Sinking of the floating power unit (6) Grounding of the floating power unit, including that on a rocky ground (7) Helicopter crash-landing

[6] Some features mentioned by the contributors in ANNEXES II, III, IV as contributing to defence-in-depth level 5 generically belong to the defence in depth level 4.

#	SMR DESIGN	LISTS OF INITIATING EVENTS	EVENTS SPECIFIC TO A PARTICULAR SMR
2	IRIS	List of design basis events corresponds to that considered by the US NRC for a typical PWR (ANNEX II) Beyond design basis events are defined on a preliminary basis: • Hypothetical reactor pressure vessel break; • Transient with failure of all safety systems	No design-specific events identified
3	CAREM-25	List of DBA defined; list of BDBA is said to be defined with no details presented (ANNEX III); Argentine's risk-informed regulatory approach to BDBA outlined (ANNEX III)	No design-specific events identified
4	SCOR	DBA and BDBA lists defined and presented; for DBA, the progression of transients in comparison with a typical PWR is qualitatively analyzed (ANNEX IV)	No design-specific events identified
5	MARS	A complete safety analysis is performed, based on a preliminary HAZOP; the general approach used and some selected points are highlighted (ANNEX V)	No design-specific events identified

Table 7 summarizes the information on acceptance criteria for design basis and beyond design basis events, provided by the designers of PWR type SMRs in ANNEXES I–V. Deterministic acceptance criteria for design basis accidents (DBA) are in most cases similar to those used for typical PWRs. Probabilistic acceptance criteria for beyond design basis accidents (BDBA) are specified as numbers for core damage frequency and large (early) release frequency in all cases except for the CAREM-25, where the requirement is to meet nationally established risk-informed criteria set by the annual probability-effective dose curve shown in Fig. 6. For one design, the MARS of Italy, notwithstanding the fact that the probabilistic safety assessment gave a much lower value, core damage frequency is still accepted at 10^{-7} 1/year level, in view of a possible common cause failure resulting from ultra-catastrophic, natural events (meteorite impact).

TABLE 7. SUMMARY OF ACCEPTANCE CRITERIA

#	SMR DESIGN	DETERMINISTIC ACCEPTANCE CRITERIA	PROBABILISTIC ACCEPTANCE CRITERIA (OR TARGETS)
1	KLT-40S	Detailed lists of acceptance criteria for pre-accident situations, DBA and BDBA defined and presented (ANNEX I)	Probabilistic acceptance criteria defined in compliance with the Russian regulatory document OPB-87/97 (see ANNEX I): • Core damage frequency (CDF) 10^{-5} 1/year; • Probability of large radioactivity release 10^{-6}1/year The probabilistic risk assessment (PRA) has demonstrated CDF to be one order of magnitude less than the prescribed limit, taking into account the uncertainties
2	IRIS	Deterministic acceptance criteria for DBA are assumed to be the same as for conventional PWRs. Deterministic acceptance criteria for BDBA, defined on a preliminary basis, include in-vessel retention of core melt by passive means (ANNEX II).	The probabilistic acceptance criteria are: • Core damage frequency $< 10^{-7}$1/year; • Large early release frequency $<10^{-9}$1/year
3	CAREM-25	Deterministic acceptance criteria for DBA are assumed to be the same as for conventional PWRs.	Risk-informed criteria set by the annual probability – effective dose curve are applied to BDBA (ANNEX III)
4	SCOR	The qualitative and quantitative objectives of radiological protection of the population and the environment developed for Generation III reactors, e.g., the EPR, are applied.	No details have been provided
5	MARS	Deterministic acceptance criteria for DBA are assumed to be the same as for conventional PWRs.	Core damage frequency accepted to be 10^{-7} 1/year, taking into account possible common cause failure depending on ultra-catastrophic events.

Table 8 summarizes the information on design features for protection against external event impacts provided by the designers of PWR type SMRs in ANNEXES I–V, with a focus on protection against aircraft crash and seismic events. Regarding other natural and human induced external events, more detained information for the IRIS and the CAREM-25 designs is provided in a dedicated IAEA report *Advanced Nuclear Plant Design Options to Cope with External Events*, IAEA-TECDOC-1487 [6]. The requirements to plant protection against external hazards excluding seismic hazard are in the IAEA safety standard [9].

Protection against aircraft crash is generally provided by the containment or a double containment (or the containment and a protective shell), with relatively small containment size rated as a factor that reduces the probability of an external missile impact on the plant. In the IRIS, the reactor building is half-embedded underground; therefore, the reactor additionally appears as low-profile, minimum sized target to an aircraft.

Structures, systems, and components of the KLT-40S are designed taking into account possible impacts of natural and human induced external events typical of the floating NPP installation site and transportation routes; see the details in Table 6. Crash-landing of a helicopter is mentioned as an event considered in the design. For the CAREM-25, protection against aircraft crash is assumed to be provided by appropriate site selection; the MARS containment is designed against the worst aircraft impact.

Seismic design corresponds to (0.4–0.5) g peak ground acceleration (PGA); for the KLT-40S, the equipment, machinery, and systems important to safety, and their mounting, are designed to withstand 3g PGA. Where indicated, the approach to seismic design is in line with the IAEA safety standard [8].

TABLE 8. SUMMARY OF DESIGN FEATURES FOR PROTECTION AGAINST EXTERNAL EVENT IMPACTS

#	SMR DESIGN	AIRCRAFT CRASH / EARTHQUAKES	OTHER EXTERNAL EVENTS
1	KLT-40S	No details provided regarding aircraft crash; crash-landing of a helicopter is considered in the design / The equipment, machinery, and systems important to safety and their mounting are designed to withstand 3g peak ground acceleration (PGA); seismic design: 7 on the MSK scale at 10^{-2} 1/year frequency for design earthquakes; 8 on the MSK scale at 10^{-4} 1/year frequency for maximum design earthquakes	Structures, systems, and components designed taking into account possible impacts of natural and human induced external events typical of the floating NPP installation site and transportation routes. Specific external events for a floating NPP are summarized in Table 6
2	IRIS	The reactor, the containment, the passive safety systems, the fuel storage, the control room, and the back-up control room located in the reinforced concrete auxiliary building, half-embedded underground. The reactor appears as low-profile, minimum-sized target to an aircraft / 0.5g PGA	Design features for protection against the impacts of natural and human induced external events are described in more detail in [6]
3	CAREM-25	Aircraft crash is not considered in CAREM-25 design – the protection is assumed to be provided by site selection and administrative measures; there are two shells (containment, confinement), and the nuclear module is compact and small, which reduces the probability of an external missile impact on the containment / 0.4 g PGA; “probable earthquake” is similar to operating basis earthquake (US NRC) or L-S1 (IAEA classification); “severe earthquake” is similar to safe shutdown earthquake (US NRC) or L-S2 (IAEA classification)	Design features for protection against the impacts of natural and human induced external events are described in more detail in [6]
4	SCOR	No information was provided	No information was provided
5	MARS	Designed against aircraft crash/ Seismic loads under reference site conditions	No further information was provided

The designers of all SMR type PWRs foresee that, eventually, their designs could be licensed with reduced or even eliminated off-site emergency planning measures or, at least, without evacuation measures beyond the plant boundary; see Table 9.

TABLE 9. SUMMARY OF MEASURES PLANNED IN RESPONSE TO SEVERE ACCIDENTS

#	SMR DESIGN	MEASURES
1	KLT-40S	- Exclusion of staff presence in the compartments adjacent to the containment and in other compartments with high radiation level; - To limit radiation dose to the population living within a 1 km radius from the floating NPP, it may be required (depending on the actual radiation situation) that some protective measures, such as iodine prophylaxis or sheltering, are implemented; - As a protection measure, temporary limitation could be established on the consumption of separate agricultural products grown in the area of up to 5 km radius from the floating NPP, and contaminated by the radioactive products; - Evacuation of the population is not required at any distance from the floating NPP.
2	IRIS	Measures essentially not needed; an option to license IRIS with reduced or eliminated off-site emergency planning is under consideration; else, the plant could be licensed with measures typical of a conventional PWR
3	CAREM-25	Measures essentially not needed; an option to license CAREM with simplified or abandoned off-site emergency planning requirements is considered, with a link to the risk-informed regulatory criteria for BDBA (see Fig. 6 and ANNEX III).
4	SCOR	No information was provided except for that on passive safety design features eliminating or preventing radioactivity releases beyond the plant boundary
5	MARS	Deterministic and probabilistic safety analyses performed conclude that licensing of MARS may not require any off-site emergency planning.

As a desired or possible feature, reduced off-site emergency planning is mentioned in the Technology Goals of the Generation IV International Forum [15], in the User Requirements of the IAEA's International Project on Innovative Reactors and Nuclear Fuel Cycles (INPRO) [14], and in the recommendations of the International Nuclear Safety Advisory Group (INSAG-12) [11], with a caution that full elimination of off-site emergency planning may be difficult to achieve or with a recommendation that Level 5 of defence in depth still needs to be kept, notwithstanding its possibly decreased role [11].

Achieving the goal of a reduced off-site emergency planning would require both, development of a methodology to prove that such reduction is possible in the specific case of a plant design and siting, and adjustment of the existing regulations. Risk-informed approach to reactor qualification and licensing could be of value here, once it gets established Within the deterministic safety approach it might be very difficult to justify reduced emergency planning in view of a prescribed consideration of a postulated severe accident with radioactivity release to the environment, e.g., owing to a common cause failure, such a catastrophic natural disaster. Probabilistic safety assessment (PSA), as a supplement to the deterministic approach, might help justify very low core damage frequency (CDF) or large early release frequency (LERF), but it does not address the consequences and, therefore, does not provide for assessment of the source terms. Risk informed approach that introduces quantitative safety goals, based on the probability-consequences curve, could help solve the dilemma by providing for a quantitative measure for the consequences of severe accidents and by applying a rational technical and non-prescriptive basis to define a severe accident.

It is worth mentioning that nuclear regulations in some countries, e.g., Argentina, already incorporate provisions for applying a risk-informed approach in the analysis of severe accidents, see Fig. 6 and ANNEX III.

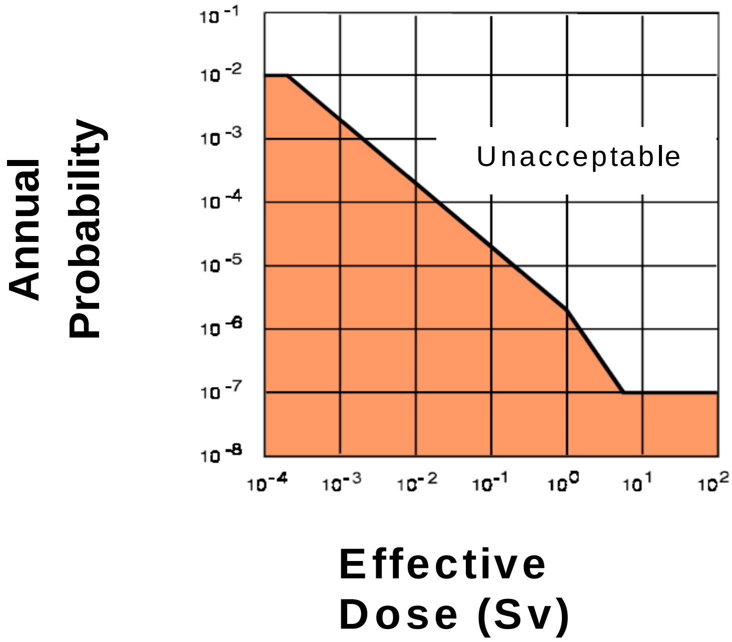


FIG. 6. Acceptance criteria for beyond design basis accidents as provided for by the regulations in Argentina (see ANNEX III).

The IAEA has recently published a report titled *Proposal for a Technology-Neutral Safety Approach for New Reactor Designs*, IAEA-TECDOC-1570 [13]. Based on a critical review of the IAEA safety standard NS-R-1 *Safety of the Nuclear Power Plants: Design Requirements* [7], the IAEA-TECDOC-1570 outlines a methodology/process to develop a new framework for development of the safety approach based on quantitative safety goals (a probability- consequences curve correlated with each level of defence-in-depth), fundamental safety functions, and generalized defence-in-depth, which includes probabilistic considerations. Different from it, the current safety approach [7] is based on qualitative safety goals, fundamental safety functions, application of the defence in depth, and application of probabilistic safety assessment complementing the deterministic methods.

Future IAEA publications and, specifically, a report of the abovementioned coordinated research project would provide more details on the progress in justification of the possibility to limit measures of Level 5 of defence in depth to the plant site.

In the meantime, the designers of PWR type SMRs accept that licensing of their plants in the near-term could be accomplished in line with the existing regulations, prescribing standard measures for the mitigation of radiological consequences of significant release of radioactive materials. These measures are mostly of administrative character. In particular, the KLT-40S designers mention that administrative measures are foreseen for plant personnel and the population within 1 km radius from the plant, but indicate that evacuation is not required at any distance from the floating NPP; for more details see ANNEX I.

Design approaches used to achieve defence in depth in the pressurized water SMRs considered in this report are generally in line with the recommendations of the IAEA safety standard NS-R-1 *Safety of the Nuclear Power Plants: Design Requirements* [7]. Specifically, the designers often refer to [7] when discussing safety objective; safety functions; defence in depth concept; accident prevention; radiation protection and acceptance criteria; safety classification; safety assessment and single failure criterion; common cause failure and redundancy, diversity and independence; conservatism in the design; and human factors. It should be noted that, because of the limited information obtained from member states, this report is not intended to provide a review of the safety design approaches applied by SMR designers against the IAEA safety standards.

The designers anticipate that future revisions of safety standards with more focus on risk-informed approach to design qualification, for example, such as suggested in the IAEA-TECDOC-1570 [13] could facilitate the goal of achieving plant qualification and licensing with reduced off-site emergency planning requirements.

3.2.2. Pressurized light water cooled heavy water moderated reactors

This reactor line is represented by only one design considered in the present report, which is the Advanced Heavy Water Reactor (AHWR) developed by the Bhabha Atomic Research Centre (BARC) of India; see Annex VI for details. The AHWR use boiling light water as a coolant in pressure channels and heavy water as a moderator in the calandria. On-line refuelling is applied, and the fuel is Pu-Th based. Figure 7 gives a schematic of the main heat transport system and passive decay heat removal system of the AHWR.

The design features contributing to different levels of defence in depth are summarized in Tables 10–14.

TABLE 10. DESIGN FEATURES OF AHWR CONTRIBUTING TO LEVEL 1 OF DEFENCE IN DEPTH

<table>
<tr><th>#</th><th>DESIGN FEATURES</th><th>WHAT IS TARGETED</th></tr>
<tr><td>1</td><td>Heat removal from the core under both normal operation and shutdown conditions performed by natural convection of the coolant</td><td>Elimination of loss of flow hazard</td></tr>
<tr><td>2</td><td>Slightly negative void coefficient of reactivity</td><td rowspan="4">Reduction of the extent of transient overpower accidents</td></tr>
<tr><td>3</td><td>Relatively low core power density</td></tr>
<tr><td>4</td><td>Negative fuel temperature coefficient of reactivity</td></tr>
<tr><td>5</td><td>Low excess reactivity due to the use of Pu-Th based fuel and on-line refuelling</td></tr>
</table>

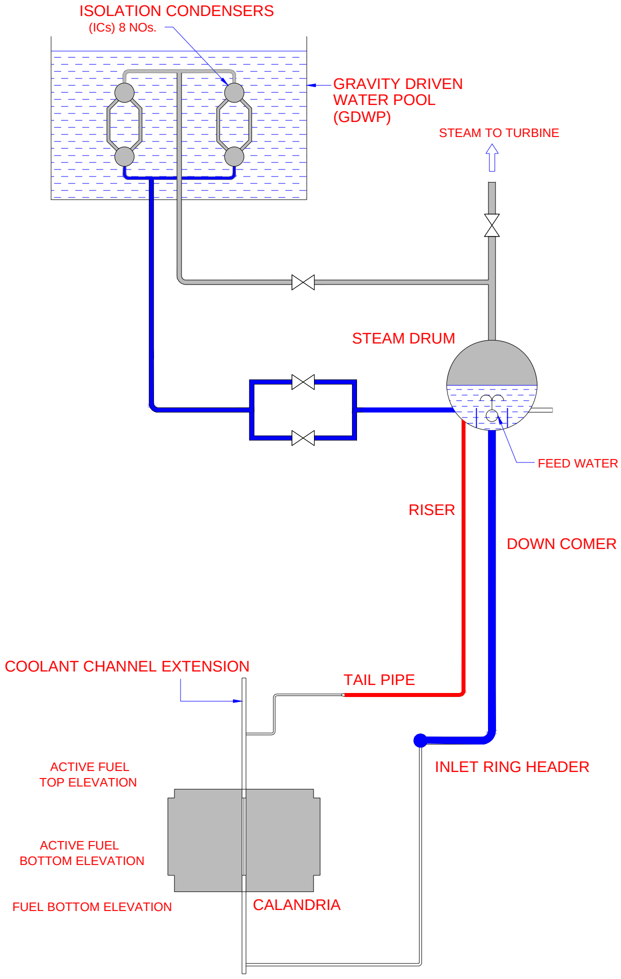


FIG. 7. Heat transport systems of AHWR.

TABLE 11. DESIGN FEATURES OF AHWR CONTRIBUTING TO LEVEL 2 OF DEFENCE IN DEPTH

#	Design feature	What is targeted
1	Large coolant inventory in the main coolant system	Increased thermal inertia; slower progression of transients
2	Digital control systems using advanced information technology	Increased reliability of the control system
3	Advanced displays and diagnostics, using artificial intelligence and expert systems	Increased operator reliability
4	Two independent and diverse shutdown systems, one based on mechanical control rods, and another employing injection of liquid poison into low pressure moderator; of 100% shutdown capacity each	Reactor shutdown

TABLE 12. DESIGN FEATURES OF AHWR CONTRIBUTING TO LEVEL 3 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED
1	Large inventory of water inside the containment (about 6000 m^3 of water in the gravity driven water pool (GDWP))	Prolonged core cooling with increased grace period
2	Passive injection of cooling water, first from the accumulator and later from the overhead GDWP, directly into the fuel cluster through four independent parallel trains	Increased reliability of emergency core cooling systems
3	Passive decay heat removal system, which transfers decay heat to GDWP using natural convection	Increased reliability of decay heat removal
4	Two independent and diverse shutdown systems, one based on mechanical control rods, and another employing injection of liquid poison into low pressure moderator; of 100% shutdown capacity each	Increased reliability of reactor shutdown
5	Additional passive shutdown device for the injection of a poison using steam pressure	Increased reliability of reactor shutdown

A distinct feature of the AHWR contributing to all levels of defence in depth is the absence of dedicated active safety systems. Heat is removed by natural convection in all modes, including normal operation mode. In the case when the main condenser and the passive Isolation Condensers (ICs) become unavailable to remove decay heat, decay heat could be removed using purification coolers of the main heat transport system in an active mode. Two independent fast acting shutdown systems are Category D [12] passive systems. All passive systems are safety grade.

Natural convection in normal operation mode contributes to the elimination of loss of flow hazard; see Table 10. Negative void reactivity coefficient, relatively low core power density, negative fuel temperature coefficient of reactivity, and low excess reactivity owing to the use of Pu-Th fuel with on-line refuelling contribute to a reduction of the extent of transient overpower accidents. Relatively large coolant inventory in the main coolant system contributes to increased thermal inertia and slower progression of transients (see Table 11), while large inventory of water inside the containment contributes to a prolonged reactor cooling with increased grace period (see Table 12).

TABLE 13. DESIGN FEATURES OF AHWR CONTRIBUTING TO LEVEL 4 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED
1	Use of the moderator as a heat sink	Establishing additional path for heat removal
2	Flooding of the reactor cavity following a LOCA	Prevention of core melt
3	Double containment	Prevention of radioactivity release to the environment; protection against external events
4	Passive containment isolation system	Prevention of core melt
5	Passive containment cooling	Prevention of core melt
6	Vapour suppression in GDWP	Prevention of failure of the primary coolant system and the containment under severe plant conditions

TABLE 14. DESIGN FEATURES OF AHWR CONTRIBUTING TO LEVEL 5 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED
1	Design features of levels 1–4 could be sufficient to achieve the goal of defence-in-depth Level 5[7]	Elimination of the need for any intervention in the public domain beyond the plant boundaries as a consequence of any accident condition within the plant

Flooding of the reactor cavity following a LOCA, use of the passive containment isolation system and passive containment cooling contribute to the prevention of core melting; see Table 13. Vapour suppression in the Gravity Driven Water Pool (GDWP), located inside the containment, contributes to the prevention of a failure of the primary coolant system and the containment under severe plant conditions.

Altogether, design features of levels 1–4 of the defence in depth are indicated by the designers as sufficient to achieve the goal of defence in depth level 5, see Table 14.

Small power rating of the AHWR obviously contributes to the extended use of natural convection based passive systems for normal and emergency reactor cooling. Other inherent and passive features of the AHWR are generically independent of the reactor capacity.

Tables 15 and 16 summarize design basis events and acceptance criteria for the AHWR. An event specific to the AHWR is the instability during a start-up, owing to the natural convection cooling mode; see Table 15.

[7] Some features mentioned in ANNEXES II, III, IV as contributing to defence-in-depth Level 5 generically belong to the defence in depth Level 4.

TABLE 15. SUMMARY OF DESIGN BASIS AND BEYOND DESIGN BASIS EVENTS, INCLUDING THOSE SPECIFIC FOR A PARTICULAR SMR

SMR DESIGN	LISTS OF INITIATING EVENTS	EVENTS SPECIFIC TO A PARTICULAR SMR
AHWR	Forty-three postulated initiating events have been identified for the AHWR; short summary of the design basis and beyond design basis event groups is given in ANNEX VI.	Instability during a start-up

TABLE 16. SUMMARY OF ACCEPTANCE CRITERIA

SMR DESIGN	DETERMINISTIC ACCEPTANCE CRITERIA	PROBABILISTIC ACCEPTANCE CRITERIA (OR TARGETS)
AHWR	Deterministic acceptance criteria are defined. It is noted that a large number of accident scenarios that would conventionally fall within the category of beyond design basis accidents have been demonstrated, via safety analysis, not to violate the acceptance criteria established for the design basis accidents.	The probability of unacceptable radioactivity release beyond the plant boundaries is expected not to exceed $1{\times}10^{-7}$ 1/year.

Table 17 gives a summary of the design features for protection against external event impacts; for more details see [6]. Double containment is used for the protection against aircraft crash. Seismic design is in line with the IAEA safety standard [8].

TABLE 17. SUMMARY OF DESIGN FEATURES FOR PROTECTION AGAINST EXTERNAL EVENT IMPACTS

SMR DESIGN	AIRCRAFT CRASH / EARTHQUAKES	OTHER EXTERNAL EVENTS
AHWR	Double containment is used for protection against aircraft crash / The AHWR structures, systems and components are being designed for high level and low probability seismic events such as operating basis earthquake (OBE) and safe shutdown earthquake (SSE); seismic instrumentation is planned in accordance with the national and international standards	The safety design features of the AHWR intended to cope with external events are described in more detail in [6]. Specifically, the AHWR is being designed to cope with floods (high grade elevation); trajectory missiles (adequate protection of all safety related buildings); ingress of toxic gases; etc. Combinations of internal and external events are considered. Important nuclear auxiliary systems are located inside the reactor building and in the basement, to the extent possible. The plant incorporates many passive safety features ensuring a grace period of 3 days.

According to the information provided in Table 18, the design target for the AHWR is to eliminate the need for any intervention in the public domain beyond the plant boundary as a consequence of any accident condition within the plant; see also Table 14.

TABLE 18. SUMMARY OF MEASURES PLANNED IN RESPONSE TO SEVERE ACCIDENTS

SMR DESIGN	MEASURES
AHWR	Measures essentially not needed; one of the design objectives of the AHWR is to eliminate the need for any intervention in the public domain beyond the plant boundaries as a consequence of any postulated accident condition within the plant

Issues of achieving plant licensing with reduced off-site emergency planning requirements are discussed in more detail in Section 3.2.1., in conjunction with measures planned in response to severe accidents for pressurized water type SMRs. This discussion is also relevant to pressurized light water moderated heavy water cooled reactors considered in this section.

The IAEA safety standard NS-R-1 “Safety of the Nuclear Power Plants: Design Requirements” [7] provides a basis for the national nuclear regulations in India.

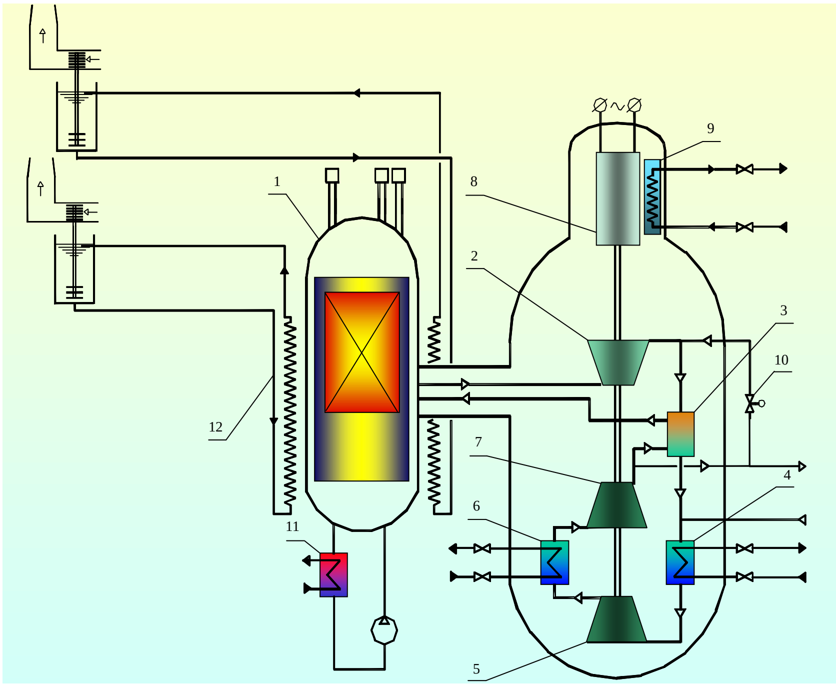
Because the AHWR uses only passive natural convection based systems for both, heat removal in normal operation (boiling light water coolant in channels) and heat removal in emergency conditions, including long-term decay heat removal; performance qualification and, specifically, justification of reliability of passive safety systems would be required to justify low targeted values of the CDF and LERF. Assessment methodologies that could facilitate achieving such a justification are discussed in Appendix 1 of the present report.

As in the case with the PWR type SMRs, future revisions of the IAEA safety standards with more focus on risk-informed safety approach, e.g., such as suggested in the IAEA-TECDOC-1570 *Proposal for a Technology-Neutral Safety Approach for New Reactor Designs* [13] could be helpful to facilitate achieving the goal of plant licensing with reduced off-site emergency planning requirements.

3.2.3. High temperature gas cooled reactors

All high temperature gas cooled reactor (HTGRs) use tristructural-isotropic (TRISO) coated fuel particles. Each of the particles consists of a fuel kernel coated with, among other layers, a ceramic layer of SiC that retains fission products at high temperatures and high fuel burn-ups. Some HTGR designs, e.g., the PBMR [2] use graphite spheres (pebbles) in which thousands of TRISO fuel particles are embedded, but other HTGR designs use pin-in-block type fuel with graphite TRISO particles incorporated in graphite pins. An example of such designs is the GT-MHR addressed in the present report; see ANNEX VII. The ability of TRISO fuel particles to contain fission products at high temperatures creates additional opportunities, relative to established practices in light water reactors, in designing safety systems and mitigation measures and essentially makes it possible to eliminate adverse consequences of many severe accidents by design. Passive decay heat removal in HTGRs can be accomplished by heat conduction through the graphite holding the TRISO particles, followed by convection and radiation in the structures and other media in the absence of the primary coolant. Also, due to large heat capacity of the graphite in the HTGR core, HTGRs have a slow and stable response to transients caused by initiating events, facilitating better reactor self-control at all levels of defence in depth. A requirement of passive decay heat removal through the reactor vessel wall and the properties of known materials for use in the reactor pressure vessel limit the unit power of HTGRs by approximately 600 MW(th); therefore, all HTGR designs fall within the SMR unit size range [2].

Figure 8 shows the flow diagram of the GT-MHR (see ANNEX VII for more details).



1–Reactor; 2–Turbine; 3–Recuperator; 4, 6–Precooler and intercooler;
5, 7–Low and high pressure compressors; 8–Generator; 9–Cooler; 10–Bypass valve;
11–Reactor shutdown cooling system; 12–Reactor cavity cooling system

FIG. 8. Flow diagram of the GT-MHR cooling system.

Tables 19-23 summarize the design features of the GT-MHR contributing to levels 1–5 of defence in depth.

TABLE 19. DESIGN FEATURES OF GT-MHR CONTRIBUTING TO LEVEL 1 OF DEFENCE IN DEPTH

#	DESIGN FEATURES	WHAT IS TARGETED
1	Use of TRISO fuel	Reliable operation and high temperatures and high fuel burn-ups
2	Use of helium coolant	• Good heat transfer properties; no dissociation and phase changes; low activation; chemical inertness; • Eliminates an option of transient overpower at coolant density variation.
3	Use of direct closed gas turbine cycle	• Design simplification, with minimization of the necessary plant equipment and systems; • Exclusion of the steam-turbine power circuit and associated impacts of its possible failures

#	DESIGN FEATURES	WHAT IS TARGETED
4	Relatively low power density of the core + large volume of graphite inside the reactor vessel + high temperature TRISO fuel + neutronic properties of helium + negative reactivity feedbacks on reactor temperature and power increase	Large temperature margin between the operation limit and the safe operation limit, and large thermal inertia of the reactor core, and self-control properties of the reactor, cumulatively resulting in an essential de-rating of accident scenarios rated as potentially severe in reactors of other types and facilitating better reactor self-control. For example, helium release from the core in the GT-MHR is a safety action, with long-term passive decay heat removal from the core possible via convection, conduction and radiation in all structures and media of the voided reactor
5	No large diameter pipelines in the primary circuit	Limitation of the scope of LOCA

TABLE 20. DESIGN FEATURES OF GT-MHR CONTRIBUTING TO LEVEL 2 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED
1	Relatively low power density of the core + large volume of graphite inside the reactor vessel + high temperature TRISO fuel + neutronic properties of helium + negative reactivity feedbacks on reactor temperature and power increase	Increased self-control properties of the reactor under a large temperature margin between the operation limit and the safe operation limit
2	Use of reliable automated control systems with a self-diagnostics capability	Increased reliability of the control of abnormal operation and prevention of failure
3	Use of the state-of-the-art operator information support system	Increased reliability of the control of abnormal operation and prevention of failure
4	Two diverse and independent passive shutdown systems; One active system of normal operation, capable of reactor shutdown.	Reactor shutdown

For Level 1 of defence in depth, “Prevention of abnormal operation and failure”, design features of the GT-MHR cumulatively result in an essential de-rating of accident scenarios rated as potentially severe in reactor of other types, including LOCA, LOFA, and reactivity initiated accidents. For example, helium release from the core in the GT-MHR can be a safety action and not an initiating event of a potentially severe accident, with indefinite passive decay heat removal from the core possible via convection, conduction and radiation in all structures and media[8]. Also, use of a direct gas turbine cycle eliminates accident initiators otherwise associated with a steam-water power

[8] Long-term passive decay heat removal may cause degradation of core structures, e.g., via graphite oxidation, etc.; therefore, early restart of normal operation systems is targeted in management of design basis accidents to facilitate continuation of a normal operation of the plant after the accident.

circuit, such as steam generator tube rupture in PWRs or water ingress in the core in indirect cycle HTGRs. Absence of large diameter piping in the primary circuit reduces the scope of possible loss of coolant accidents. Helium properties exclude transient overpower events owing to coolant density variation.

TABLE 21. DESIGN FEATURES OF GT-MHR CONTRIBUTING TO LEVEL 3 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED
1	Increased role of the inherent safety features, such as negative reactivity feedbacks on reactor power and temperature and natural processes of conduction, radiation and convection, provided by design	Slow progression of transients resulting from large thermal inertia of the core; large temperature margin between the operation limit and the safe operation limit; and slow temperature variation at power variation
2	Preferential use of passive safety systems (see Section 7.2. in ANNEX VII)	Increased reliability of carrying out the safety functions
3	Mechanical control rod system providing gravity-driven insertion of control rods to the core and the reflector, operated in the case of de-energization actuated by the control system	Reactor scram
4	Reactor emergency shutdown system based on gravity-driven insertion of spherical absorbing elements to the dedicated channels located within the core stack, initiated by supplying power from diesel generators to the drive motors	Effective shutdown of the reactor and keeping it sub-critical in a cold un-poisoned state
5	Active electromechanical reactivity control system, which is a normal operation system that shoulders the functions of a safety system	Reactivity control and hot reactor shutdown
6	Passive residual heat removal from the core based on natural processes of conduction, radiation and convection, requiring no external power sources, control signals, or human intervention, ending up with heat removal from outside of the reactor vessel to the environmental air by the always effective passive reactor cavity cooling system	• Increased reliability of control of accidents within the design basis; • Securing that fuel safe operation limits are met at passive shutdown and cooling of the reactor
7	• Low core power density; • Annular reactor core with a high surface-to-volume ratio; • Central reflector; • High heat capacity of the reactor core and internals; • Heat resistant steel used for the reactor internals and vessel.	Facilitate effective operation of the reactor cavity cooling system

For Level 2 of defence in depth, “Control of abnormal operation and detection of failure”, the contribution comes from advanced instrumentation and control and operator support systems, but also from the inherent safety features owing to the reactor design. The latter secure increased self-control properties of the reactor under a large temperature margin between the operation limit and the safe operation limit. Finally, two independent and diverse passive reactor shutdown systems and one active system of normal operation, capable to perform reactor shutdown, are available to contribute to this level.

TABLE 22. DESIGN FEATURES OF GT-MHR CONTRIBUTING TO LEVEL 4 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED
1	Additional physical barriers provided by the design of fuel with multi-layer TRISO coatings	Mitigation of consequences of severe accidents
2	Inherent and passive safety features and passive safety systems incorporated in plant design, cumulatively (see ANNEX VII)	Securing that the final stable and safe conditions are reached when the chain reaction of fission is suppressed and when continuous cooling of nuclear fuel and retention of radioactive substances within the established boundaries are provided
3	Stop of reactor core cooling by helium, as a safety action	Passive residual heat removal from the core based on natural processes of conduction, radiation and convection, requiring no external power sources, control signals, or human intervention, ending up with heat removal from outside of the reactor vessel to the environmental air by the always effective passive reactor cavity cooling system
4	An option of beyond design basis accident management by personnel in the case of failures of safety components and systems, secured by: • Safety design features of the reactor that limit the progression of accidents; • Characteristics of the passive systems; • Capabilities of the normal operation systems; • Large time margins for implementation of accident management measures.	Increased confidence that the objectives of the defence-in-depth Level 4 will be fulfilled
5	Containment designed to retain the helium-air fluid and to withstand external loads	Increased confidence that the objectives of the defence-in-depth level 4 will be fulfilled under impacts of the internal and external events and combinations thereof

As it was already mentioned, increased role of the inherent safety features, such as negative reactivity feedbacks on reactor power and temperature; high thermal inertia of the reactor core; and natural processes of conduction, radiation and convection, provided by design of the HTGRs, facilitate high degree of reactor self-control at all levels of defence in depth and secure that safe operation limits of the fuel are met, and that passive shutdown and cooling of the reactor are provided for a variety of postulated initiating events; see ANNEX VII. These features are also effective at Level 3 of defence in depth, “Control of accidents within design basis”. Specifically, long-term passive decay heat removal accomplished via natural processes of conduction, convection and radiation and through operation of the reactor cavity cooling system, even in the absence of the helium coolant in the reactor coolant system, is facilitated by the GT-MHR design features listed explicitly in position 7 of Table 21. For Level 3 of defence in depth, the reactor incorporates two independent and diverse reactor shutdown systems, which operate on passive principles and are passively actuated. In addition to them, an active electromechanical reactivity control system, which is a normal operation system, is capable of accomplishing the function of hot reactor shutdown.

The GT-MHR design provides for no dedicated active safety systems. Active systems of normal operation, such as the power conversion unit (PCU), the shutdown cooling system (SCS), and the electromechanical reactivity control system can be used for safety purposes; see ANNEX VII. These systems remove heat under abnormal operation conditions, and in design basis and beyond design basis accidents. All main passive safety systems are safety grade. The electromechanical reactivity control system (an active system of normal operation) is safety grade too.

The GT-MHR features contributing to increased confidence that the objective of Level 4 of defence in depth, “Mitigation of radiological consequences of significant release of radioactive materials”, would be fulfilled are (see Table 22):

- Additional physical barriers provided by the design of fuel with TRISO multi-layer coatings, securing short-term fission product confinement capability at temperatures as high as 2100°C and long-term fission product confinement capability at 1600°C – essentially, each micro fuel element in the HTGR has its own containment;
- Safety design features of the reactor that limit the progression of accidents (see more detailed discussion for levels 2 and 3 of defence in depth);
- Characteristics of the passive systems (long-term passive decay heat removal capability via conduction, convection and radiation even in the absence of helium in the core, but with the operation of a passive cavity cooling system);
- Capabilities of the normal operation systems – although passive decay heat removal can be practically infinite, retaining the capability to restart the reactor for normal operation after an emergency may be facilitated by on-time restart of some normal operation systems during the emergency process, e.g., to prevent graphite oxidation, see ANNEX VII for details;
- Large time margins for implementation of accident management measures (see more detailed discussion for Level 3 of defence in depth);
- Use of the containment designed to retain the helium-air fluid.

TABLE 23. DESIGN FEATURES OF GT-MHR CONTRIBUTING TO LEVEL 5 OF THE DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED
1	Design features of levels 1–4 could be sufficient to achieve the goal of defence-in-depth Level 5	No accident mitigation measures required both within and beyond the NPP site

The designers of the GT-MHR foresee that the design features of levels 1–4 of the defence in depth could be sufficient to achieve the goal of defence-in-depth Level 5, "Mitigation of radiological consequences of significant release of radioactive materials".

Tables 24 and 25 summarize the information on design basis and beyond design basis accidents and acceptance criteria provided by the designers of the GT-MHR in ANNEX VII. The event (abnormal operation occurrence) specific to the GT-MHR but not necessarily to other HTGRs is inadvertent insertion of absorbing elements from the reserve shutdown system hoppers into the reactor core.

TABLE 24. SUMMARY OF DESIGN BASIS AND BEYOND DESIGN BASIS EVENTS, INCLUDING THOSE SPECIFIC FOR A PARTICULAR SMR

SMR DESIGN	LISTS OF INITIATING EVENTS	EVENTS SPECIFIC TO A PARTICULAR SMR
GT-MHR	Detailed lists of initiating events for abnormal operation occurrences, DBA, and BDBA is presented (ANNEX VII)	Abnormal operation occurrence: Inadvertent insertion of absorbing elements from the reserve shutdown system (RSS) hoppers into the reactor core

TABLE 25. SUMMARY OF ACCEPTANCE CRITERIA

SMR DESIGN	DETERMINISTIC ACCEPTANCE CRITERIA	PROBABILISTIC ACCEPTANCE CRITERIA (OR TARGETS)
GT-MHR	The acceptance criteria are radiation safety criteria (deterministic criteria related to the limit doses of irradiation to the personnel and population). In addition to this, the operation limits and the safe operation limits are defined for process parameters; the operation limits are defined for the equipment; the design limits are specified for the analysis of design basis accidents; and the acceptance criteria are introduced for different operation modes, see ANNEX VII.	Probabilistic acceptance criteria are defined as follows: • The overall probability of severe beyond design basis accidents less than 10^{-5} per reactor per year; Probability of large radioactivity release less than 10^{-7} per reactor per year.

Table 26 summarizes the design features of the GT-MHR contributing to plant protection against external event impacts.

TABLE 26. SUMMARY OF DESIGN FEATURES FOR PROTECTION AGAINST EXTERNAL EVENT IMPACTS

SMR DESIGN	AIRCRAFT CRASH / EARTHQUAKES	OTHER EXTERNAL EVENTS
GT-MHR	The design of the GT-MHR ensures protection against aircraft crash of a 20 t aircraft falling with a 200 m/s speed and producing a 7 m^2 impact area / Maximum design basis earthquake corresponds to 8 on the MSK scale (horizontal PGA component is 0.2 g; vertical component is 2/3 of the horizontal one); design basis earthquake corresponds to 7 on the MSK scale (PGA components are two times lower than for the maximum design basis earthquake).	Other external events considered in the GT-MHR design are winds, low and high environmental temperatures, shock wave impacts, etc. The reactor plant is arranged in a monolithic ferroconcrete underground containment that provides protection against the external event impacts. Apart from external events, the containment provides a protection against internal impacts, such as caused by jets and missiles.

TABLE 27. SUMMARY OF MEASURES PLANNED IN RESPONSE TO SEVERE ACCIDENTS

SMR DESIGN	MEASURES
GT-MHR	Design features and inherent properties of the GT-MHR ensure that the temperature of the coated particle fuel is kept below 1600 ^{o}C in any accidents with a heat removal failure, including complete failure of all active means of the reactor emergency protection and shutdown. At this temperature, the integrity of the fuel element coatings is maintained; therefore, no protective measures would be required for the population beyond the buffer area.

Table 27 gives a summary of the measures planned in response to severe accidents. Like in the case of several other SMRs in this report, the designers foresee no need in measures for the population protection beyond a certain buffer area around the plant, in any accidents with a heat removal failure accompanied by the failure of all active means of the reactor emergency protection and shutdown.

Issues of achieving plant licensing with reduced off-site emergency planning requirements are discussed in more detail in Section 3.2.1., in conjunction with measures planned in response to severe accidents for pressurized water type SMRs. This discussion is also relevant to high temperature gas cooled reactors considered in this section.

Although the ultimate goal is to prove that no accident mitigation measures would be required both within and beyond the NPP site, licensing of a first-of-a-kind plant is likely to be carried out in compliance with the existing regulatory rules and practices.

It is expected that technology-neutral approach may facilitate assessment of the design features of HTGRs, including the GT-MHR. Specifically, the IAEA-TECDOC-1570 *Proposal for a Technology-Neutral Safety Approach for New Reactor Designs* [13] suggests that *"*in the design of innovative reactors it may be possible, by following the risk-informed approach, to provide a justification that a confinement system designed to the same standards that have been established for LWR technology would not be needed. This may be because, for example, there are mitigating features of the design of the fuel which limit the quantity of radioactive materials released, and allow the reactor to return to a stable state without impairing the ability of the fuel to be maintained within its design matrix with little or no release of fission products. Another consideration may be that of the timescale before the plant state escalates to a condition where corrective action, e.g., initiation of cooling systems, is necessary.*"*

Certain passive decay heat removal mechanisms of the GT-MHR (and HTGRs), such as natural convection, conduction and radiation, are rated reliable and independent of possible disruptions of core configuration. Their reliability, as well as passive response of the reactor to unprotected accidents, such as LOCA or control rod ejection, could be proven via a 'license-by-test' approach, e.g., as demonstrated in the tests performed at the HTR-10 reactor in China [17].

3.2.4. Liquid metal cooled fast reactors

All fast reactor designs in the SMR family offer design flexibility in setting desired combinations of reactivity coefficients and effects. This flexibility, coupled with the inherent properties of the advanced types of fuel, creates a potential to prevent transient overpower accidents; to ensure increased reactor self-control in a variety of other anticipated transients without scram and combinations thereof; and to enable “passive shutdown” (see the definition in the end of Appendix 2) and passive load following capabilities for a plant[9]. Smaller specific core power or relatively tall reactor vessels facilitate the use of natural convection of a single-phase liquid metal coolant to remove decay heat or even the heat produced in normal operation (for heavy liquid metal cooled SMRs). For sodium cooled reactors, smaller reactor size facilitates achieving negative whole-core sodium void reactivity effect. For lead cooled reactors, there could be a certain size limit to ensure a reliable seismic design [2].

Figure 9 and Figure 10 show general layouts of the 4S-LMR and the SSTAR, respectively.

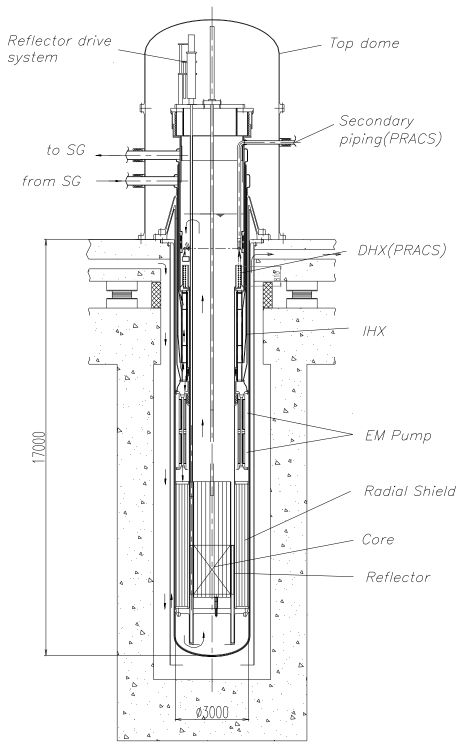


FIG. 9. Vertical view of the 4S-LMR layout.

Fast-spectrum liquid metal cooled SMR designs are represented by the 4S-LMR concept of a sodium cooled small reactor without on-site refuelling developed by the Central Research Institute of Electric Power Industry (CRIEPI) and Toshiba in Japan (see ANNEX VIII) and by the SSTAR and STAR-LM concepts of small lead cooled reactors without on-site refuelling developed by the Argonne National Laboratory (ANL) in the USA (see ANNEX IX). The lead cooled SMR concepts use CO_2 as working media in the Brayton cycle power circuit, and incorporate no intermediate heat transport system. Although essentially different in several important features, both sodium cooled and lead cooled SMR concepts belong to a family of pool-type integral design liquid metal cooled fast reactors, and close cooperation between their designers has been established long ago

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[9] It should be noted that such features of liquid metal cooled reactors as passive load following and “passive shutdown” have been more analyzed in the past for smaller-sized reactors, such as EBR-II of 65 MW(th) or PRISM of 850 MW(th). However, for sodium and lead cooled fast reactors, there are no reasons that such features couldn’t be realized in larger reactors with nitride or metallic fuel. Certain analytical studies carried out in the past provide preliminary proofs of this [26, 27, and 28].

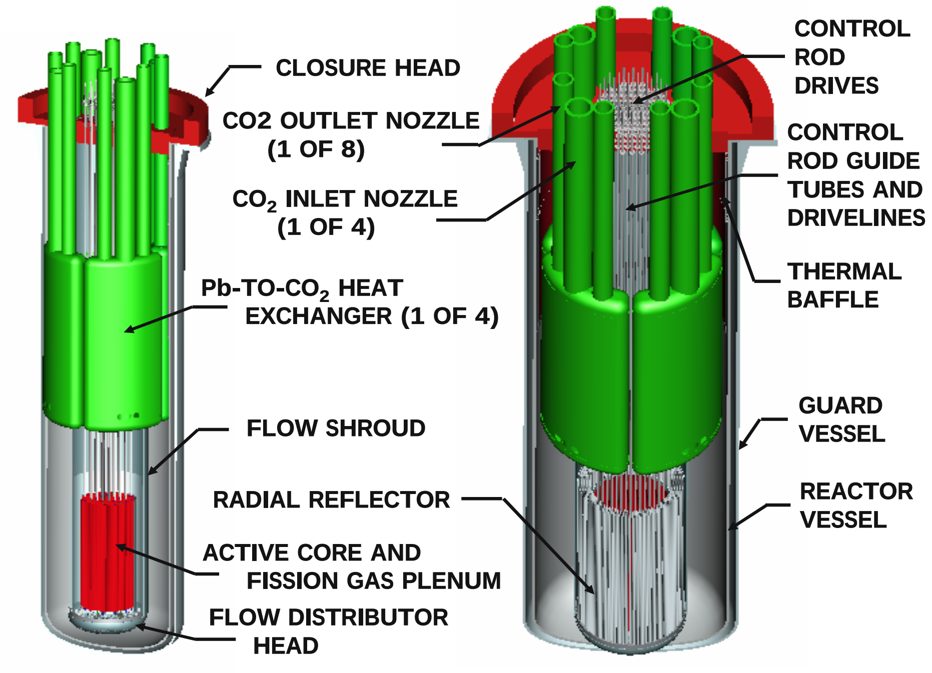

FIG. 10. General view of the SSTAR layout.

[3]. Of the two designs, the 4S-LMR is in a more advanced stage, because for a similar design, different essentially in the type of fuel and named the 4S, the conceptual design and major parts of the system design have been completed [3]. A pre-application review by the US NRC has been initiated in the fall of 2007. Construction of a demonstration reactor and safety tests are planned for early 2010s [3]. Different from it, both the SSTAR and STAR-LM are at a pre-conceptual stage. It should be noted that small size and capacity of the fast reactors considered in this section are, first-of-all, conditioned by the requirement of operation without on-site refuelling (see [3] for more detail) and not by the a priori considerations of achieving a somewhat higher degree of passive response in accidents.

Tables 28–32 summarize the design features of the 4S-LMR and the SSTAR and the STAR-LM contributing to defence in depth levels 1–5.

TABLE 28. DESIGN FEATURES OF SODIUM COOLED AND LEAD COOLED FAST SMRS CONTRIBUTING TO LEVEL 1 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
1	Low-pressure primary coolant system	Low non-nuclear energy stored in the primary coolant system – elimination of a potential of release of this energy	4S-LMR, SSTAR, STAR-LM
2	Use of metallic fuel with high thermal conductivity (relatively low temperature)	High margin to fuel failure	4S-LMR
3	Use of nitride fuel with high thermal conductivity (relatively low temperature)	High margin to fuel failure	SSTAR, STAR-LM
4	Relatively low linear heat rate of fuel	Higher margin to fuel failure	4S-LMR
5	Power control via pump flow rate in the power circuit, with no control rods in the core	Elimination of an accident with control rod ejection	4S-LMR

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
6	Large negative feedbacks from fast-spectrum core plus natural convection of the coolant in all modes, enabling passive load following and "passive shutdown"[10]	Essential prevention or de-rating of the initiating events resulting from malfunctioning of the systems or components, or operator actions that would otherwise need to be considered as sources of failure	SSTAR, STAR-LM
7	Low burn-up reactivity swing over long core lifetime/ refuelling interval	Elimination of transient overpower accident due to control rod ejection	SSTAR, STAR-LM
8	Elimination of feedback control of moveable reflectors (that compensate for reactivity changes due to fuel burn-up); a pre-programmed reflector drive system is used	Prevention of transient overpower	4S-LMR
9	Electromagnetic impulsive force used in the reflector driving system	Intrinsic limitation of the speed of positive reactivity insertion	4S-LMR
10	Intermediate heat transport system	Prevention of sodium-water reaction	4S-LMR
11	Pb coolant not reacting chemically with CO_2 working fluid; no intermediate heat transport system	Elimination of a chemical interaction between the primary coolant and the working fluid of a power circuit	SSTAR, STAR-LM
12	Natural convection of the coolant plus open fuel element lattice (large fuel element pitch-to-diameter ratio)	- Elimination of loss of flow accidents; - Prevention of flow blockage accidents	SSTAR, STAR-LM
13	Primary electromagnetic (EM) pumps arranged in two units connected in series, with each unit capable of taking on one half of the pump head	Prevention of loss of flow	4S-LMR
14	The reactor vessel enclosed in a guard vessel to prevent loss of the primary coolant; pool type design with intermediate heat exchangers located inside the main reactor vessel	Prevention of loss of coolant (LOCA)	4S-LMR
15	Use of double piping, double tubes and double vessels for the secondary sodium, including heat transfer tubes of the steam generator	- Prevention of LOCA - Prevention of sodium-water reaction	4S-LMR
16	- The reactor vessel enclosed in a guard vessel such that even in the case of primary vessel boundary rupture, the faulted level of lead will always exceed the Pb entrances to the PB-to-CO_2 heat exchangers; - High boiling point of the Pb coolant (1740°C), exceeding the point at which stainless steel core structures melt; - Pool type design configuration; - High density of Pb coolant limits void growth and downward penetration following a postulated in-vessel heat exchanger tube rupture.	Prevention of loss of coolant (LOCA) and its possible consequences	SSTAR, STAR-LM

[10] "Passive shutdown" is used to denote bringing the reactor to a safe low-power state with balanced heat production and passive heat removal, with no failure to the barriers preventing radioactivity release to the environment; all relying on the inherent and passive safety features only, with no operator intervention, no active safety systems being involved, and no external power and water supplies being necessary; and with practically infinite grace period.

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
17	High-reliability system of control of dissolved oxygen potential in the Pb coolant	- Maintaining the integrity of stainless steel cladding in all modes of operation by preventing corrosion[11]; - Prevention of the formation of corrosion debris with a potential to block coolant area.	SSTAR, STAR-LM

Design features contributing to Level 1 of defence in depth, "Prevention of abnormal operation and failure", are summarized in Table 28.

Low-pressure primary coolant system, securing low non-nuclear energy stored in the primary coolant system is a common feature of all liquid metal cooled reactors, irrespective of their size and capacity. In addition to it, like many innovative liquid metal cooled reactors of a variety of capacities and sizes, all SMRs considered in this section rely on advanced fuel designs with high thermal conductivity, ensuring increased margins to fuel failure.

The lead cooled SSTAR and STAR-LM reactors incorporate optimum sets of reactivity feedbacks, provided by design and contributing to the elimination of transient overpower, as well as to the prevention or de-rating of the initiating events resulting from malfunctioning of systems or operator actions. Specifically, the designers of the SSTAR and STAR-LM mention the so-called "passive shutdown" capability of their reactors as provided by design.

The sodium cooled 4S-LMR provides for power control via pump flow rate in the power circuit, with no control rods in the core, and for pre-programmable movement of axial reflectors with no feedback control, contributing to burn-up reactivity compensation. Both of these features contribute to the prevention of transient overpower accidents.

To prevent sodium-water reaction, the 4S-LMR incorporates intermediate heat transport system, like most of the sodium cooled fast reactors. As the CO_2 is used as a working medium in the power circuits of the SSTAR and STAR-LM, which does not react chemically with Pb, these reactors do not incorporate intermediate transport system.

Natural convection is used in the SSTAR and STAR-LM to remove heat under normal operation, eliminating loss of flow accidents. De-rating of loss of flow in the 4S-LMR is achieved by a scheme with two electromagnetic pumps connected in series.

Both sodium and lead cooled SMRs incorporate guard vessel to prevent LOCA; the 4S-LMR also incorporates double piping and double vessels for secondary sodium, including heat transfer tubes of the steam generator.

Finally, reliable system of corrosion control is assumed to be provided for the SSTAR and STAR-LM to maintain the integrity of stainless steel claddings and to prevent the formation of the corrosion debris with a potential of coolant area blockage. For these reactors it is important to maintain the oxygen potential in the correct regime to prevent the formation of PbO, which needs to be avoided. There could also be corrosion debris such as Fe that migrates into the coolant where it forms iron oxide that should be filtered out.

[11] Corrosion/erosion is generally a slow and easily detectable process.

TABLE 29. DESIGN FEATURES OF SODIUM COOLED AND LEAD COOLED FAST SMRS CONTRIBUTING TO LEVEL 2 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
1	All-negative temperature reactivity coefficients	Increased self-control of abnormal operation	4S-LMR
2	Large negative feedbacks in fast spectrum core; natural convection of the coolant in all modes; physical properties of Pb coolant and nitride fuel with high heat conductivity	Increased self-control of abnormal operation, including passive load following and “passive shutdown”	SSTAR, STAR-LM
3	Large thermal inertia of the coolant and the shielding structure	Slow pace of the transients due to abnormal operation	4S-LMR, SSTAR, STAR-LM
4	Sodium leak detection system in the heat transfer tubes of the steam generator, capable of detecting both inner and outer tube failures	Enhanced detection of failure of the secondary sodium boundary	4S-LMR
5	Two redundant power monitoring systems; balance of plant temperature monitoring system; electromagnetic pump performance monitoring system; cover gas radioactivity monitoring system, etc.	Enhanced control of abnormal operation and detection of failure	4S-LMR
6	System of monitoring of the dissolved oxygen potential in the Pb coolant	Control of the corrosion/erosion processes of stainless steel claddings in Pb flow and detection of failures	SSTAR, STAR-LM
7	Independent and redundant shutdown systems (see Table 30 for details)	Reactor shutdown	All designs

For Level 2 of defence in depth, “Control of abnormal operation and prevention of failure”, the contributions come from large thermal inertia of the primary coolant system and reactor internals, resulting in a slow progress of transients, and from optimum negative feedbacks, provided by design and ensuring high-degree of reactor self-control. Specifically, passive load following and “passive shutdown” capability are mentioned for the SSTAR and STAR-LM. Monitoring and detection systems are other important contributors. Finally, independent and redundant active or passive shutdown systems are available for the cases when all other measures of control and prevention turn out to be ineffective.

TABLE 30. DESIGN FEATURES OF SODIUM COOLED AND LEAD COOLED FAST SMRS CONTRIBUTING TO LEVEL 3 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
1	Use of metallic fuel with high thermal conductivity (relatively low temperature)	High margin to fuel failure; larger grace period	4S-LMR
2	Use of nitride fuel with high thermal conductivity (relatively low temperature)	High margin to fuel failure; larger grace period	SSTAR, STAR-LM
3	Relatively low linear heat rate of fuel	Higher margin to fuel failure; larger grace period	4S-LMR
4	All-negative temperature reactivity coefficients	Increased reactor self-control in design basis accidents	4S-LMR
5	Large negative feedbacks from fast spectrum core, natural convection of the coolant in all modes, physical properties of Pb coolant and nitride fuel with high heat conductivity	Increased self-control of the reactor in design basis accidents, including passive load following and “passive shutdown” (in the case of a failure of both scram systems)	SSTAR, STAR-LM
6	Negative whole-core void worth	Prevention of design basis accidents propagation into beyond design basis conditions (due to coolant boiling or loss)	4S-LMR
7	- Very high boiling point of Pb coolant (1740°C); - Escape path for gas/void to reach free surface provided by design; - The reactor vessel is enclosed in a guard vessel such that even in the case of primary vessel boundary rupture, the faulted level of lead will always exceed the Pb entrances to the PB-to-CO_2 heat exchangers.	Prevention of core void as the extension of design basis accidents; securing of normal heat removal path through Pb/CO_2 heat exchangers in DBA	SSTAR, STAR-LM
8	Large specific (per unit of power) inventory of the primary coolant	Increased grace period	4S-LMR, SSTAR, STAR-LM
9	Effective radial expansion of the core (negative feedback), provided by design	Increased reactor self-control in design basis accidents; prevention of DBA propagation into beyond design basis conditions	4S-LMR, SSTAR, STAR-LM
10	Low pressure loss in the core region, provided by design	Increased level of natural circulation to remove decay heat from the core	4S-LMR
11	A combined system of electromagnetic pumps and synchronous motors (SM), ensuring favourable flow coast-down characteristics	Increased grace period in the case of pump failure	4S-LMR

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
12	Natural convection of the coolant in all modes of operation plus open fuel element lattice (large fuel element pitch-to-diameter ratio)	Increased reliability of heat removal by natural convection of the coolant via Pb-CO_2 heat exchangers and, in the case of their failure, by natural convection based decay heat removal systems RVACS and DRACS	SSTAR, STAR-LM
13	Two independent systems of reactor shutdown, each capable of shutting down the reactor by: - A drop of several sectors of the reflector; or - Gravity-driven insertion of the ultimate shutdown rod.	Reactor shutdown	4S-LMR
14	Two independent and redundant active safety grade shutdown systems	Reactor shutdown[12]	SSTAR, STAR-LM
15	Redundant and diverse passive auxiliary cooling systems (RVACS and IRACS or PRACS), both using draught of environmental air as an ultimate heat sink	Increased reliability of decay heat removal from the core	4S-LMR
16	Two or more safety grade independent Direct Reactor Auxiliary Cooling System (DRACS) providing independent paths for decay heat removal. The reactor vessel auxiliary cooling system (RVACS), if present, will be a single safety grade decay heat removal system. If RVACS and DRACS are both present, this provides even a greater diversity. However, if DRACS are effective, the role of RVACS would be reduced. All systems will use natural draught of air as an ultimate heat sink.	Increased reliability of decay heat removal from the core (especially, when the normal path via Pb-CO_2 heat exchangers becomes unavailable)	SSTAR, STAR-LM
17	Use of double piping, double tubes and double vessels for the secondary sodium, including heat transfer tubes of the steam generator	Prevention of steam generator tube rupture, sodium-water reaction, and pressure increase in the intermediate heat transport system	4S-LMR
18	Passive pressure relief from primary coolant system	Protection of the reactor vessel and enclosure from over-pressurization in the case when one or more of the in-vessel Pb-to-CO_2 heat exchanger tubes fail	SSTAR, STAR-LM

[12] It is noted that the operation of these systems may actually be unnecessary because the inherent and passive features are in any case capable to ensure a “passive shutdown”, i.e., bringing the reactor to a safe low-power state with balanced heat production and passive heat removal, with no failure to the barriers prevention radioactivity release to the environment, and with practically indefinite grace period

For Level 3 of defence in depth, “Control of accidents within design basis”, the contribution comes for the following main groups of design features:

(1) Inherent safety features, highlighted in positions 1–8 of Table 30. In addition to the features already discussed in conjunction with defence in depth levels 1 and 2, it is important to note negative whole-core void worth provided by design in the 4S-LMR and inherent features of the lead cooled SSTAR and STAR-LM, practically eliminating the option of coolant boiling or gas bubbles arriving to the core (preventing the propagation of a design basis accident into a severe accident with transient overpower);

(2) By-design provisions for certain passive mechanisms such as radial expansion or enhanced level of natural convection in the primary coolant system, highlighted in positions 9–12 of Table 30;

(3) Two independent systems of reactor shutdown, provided in each design; see positions 13–14 of Table 30. Those operate based on gravity in the 4S-LMR, while in the SSTAR and the STAR-LM both systems are active and safety grade. For the SSTAR and STAR-LM, it is mentioned that the operation of these systems may actually be unnecessary because the inherent and passive features are in any case capable to ensure a “passive shutdown” of the reactor;

(4) Not less than two redundant and diverse passive decay heat removal systems in each design, with some of them, possibly, providing several passive decay heat removal paths, and all using natural draught of air as an ultimate heat sink, positions 15–16 of Table 30;

(5) Special design features provided to prevent or mitigate the effects of pressurized medium from the power circuit getting into the primary circuit; positions 17–18 of Table 30.

The 4S-LMR incorporates no active safety systems. However, there are several active systems providing normal operation of the reactor at rated or de-rated power, e.g., electromagnetic pumps providing forced convention of sodium coolant to remove core heat, or burn-up reactivity compensation system based on slow upward movement of the reflector, using advanced pre-programmed drive mechanism. These systems can contribute to performing safety functions in certain accident scenarios. No information was provided on which systems of the 4S-LMR are safety grade.

All passive and active safety systems in the SSTAR and the STAR-LM are assumed to be safety grade.

TABLE 31. DESIGN FEATURES OF SODIUM COOLED AND LEAD COOLED FAST SMRS CONTRIBUTING TO LEVEL 4 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
1	Inherent safety features of a metal or nitride fuelled core, such as high thermal conductivity and low accumulated enthalpy	Prevention of core melting	4S-LMR, SSTAR, STAR-LM
2	Large negative feedbacks from fast spectrum core, natural convection of the coolant in all modes, physical properties of Pb coolant and nitride fuel with high heat conductivity	Prevention of core melting	SSTAR, STAR-LM

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
3	Relatively low linear heat rate of fuel	Prevention of core melting	4S-LMR
4	Large specific (per unit of power) inventory of the primary coolant, contributing to high heat capacity of the primary coolant system	Increased capability of the coolant system to absorb heat; prevention of core melting	4S-LMR, SSTAR, STAR-LM
5	Negative whole-core void worth	Prevention of transient overpower in the case of coolant boiling or void penetration to the core	4S-LMR, SSTAR[13]
6	Redundant and diverse passive auxiliary cooling systems (RVACS and IRACS or PRACS), both using draught of environmental air as an ultimate heat sink	Increased reliability of decay heat removal from the core	4S-LMR
7	Two redundant and diverse passive decay heat removal systems, reactor vessel auxiliary cooling system (RVACS) and, perhaps, direct reactor auxiliary cooling system (DRACS), both using draught of environmental air as an ultimate heat sink	Increased reliability of decay heat removal from the core	SSTAR, STAR-LM
8	Effective mechanism of fuel carry-over from the core in the case of fuel element cladding failure	Prevention of re-criticality	4S-LMR
9	High effective density of the Pb coolant (~11 g/cm^3) plus pool type design	In the case of melting, fuel is moved to an upper free level of lead, preventing re-criticality	SSTAR, STAR-LM
10	Fast-acting system of sodium drain from the steam generator to the dump tank	Mitigation of sodium-water reaction	4S-LMR
11	The reactor vessel enclosed in a guard vessel to prevent loss of primary sodium; pool type design with intermediate heat exchangers located inside the main reactor vessel	Prevention of radioactivity release to the environment	4S-LMR
12	Use of double piping, double tubes and double vessels for the secondary sodium, including heat transfer tubes of the steam generator	Prevention of radioactivity release to the environment	4S-LMR

[13] In both the SSTAR and STAR-LM, generation of void in the core is practically excluded by design; in addition to this, Pb boiling temperature (1740°C) exceeds the melting temperature of core structures made of stainless steel.

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
13	The guard vessel surrounds the reactor vessel, and an upper enclosure head covers both, the reactor vessel and the guard vessel. A hermetic seal is established between the upper closure head and the guard vessel. In the event of a rupture of one or more Pb-to-CO_2 heat exchangers, the CO_2 would vent trough an upper closure head into the volume of the containment structure	Prevention of radioactivity release to the environment; securing the integrity of the reactor vessel and a heat removal path contributing to core melt prevention	SSTAR, STAR-LM
14	The containment	Prevention of radioactivity release to the environment	4S-LMR, SSTAR, STAR-LM
15	The reactor located in a concrete silo below the ground level	Prevention of radioactivity release to the environment	4S-LMR, SSTAR, STAR-LM

The design feature contributing to Level 4 of defence in depth, "Control of severe plant conditions, including prevention of accident progression and mitigation of consequences of severe accidents" fit in the following main groups; see Table 31:

(1) The inherent safety features contributing to prevention of core melting, positions 1–5 of Table 31;

(2) Redundant and diverse passive decay heat removal systems with natural draught of air used as an ultimate heat sink, discussed in more detail in conjunction with Level 3 of defence in depth;

(3) Inherent and passive design features for the prevention of recriticality, positions 8–9 of Table 31. Those include an effective mechanism of fuel carry-over from the core in the case of fuel element cladding failure (4S-LMR) and high density of the Pb coolant securing that molten fuel is moved to the upper free level of lead (SSTAR and STAR-LM);

(4) Guard vessels in addition to the main vessels, for all designs, and double piping for the 4S-LMR, positions 11–13 of Table 31;

(5) The containment and reactor location in a concrete silo below the ground level, for all designs considered.

For Level 5 of defence in depth, "Mitigation of radiological consequences of significant release of radioactive materials", the designers of the 4S-LMR foresee no measures needed beyond the plant boundary in response to any severe accidents and combinations thereof, even in the case when there is no operator intervention, no emergency team actions, and no external power and water supply. The designers of the SSTAR and STAR-LM take a more conservative approach, suggesting that standard measures may still be applicable but within the exclusion zone reduced against that of the present day reactors; see Table 32 and Table 35.

TABLE 32. DESIGN FEATURES OF SODIUM COOLED AND LEAD COOLED FAST SMRS CONTRIBUTING TO LEVEL 5 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED	SMR DESIGNS
1	The inherent and passive safety features ensure the plant to survive all postulated design basis and beyond design basis accidents, including anticipated transients without scram and combinations thereof, without operator intervention, emergency team actions, and external power and water supply	Eliminate the need for any intervention in the public domain beyond the plant boundaries as a consequence of any accident condition within the plant	4S-LMR
2	The inherent and passive safety features ensure lower probability of radioactivity material release to the environment (as compared to present day light water reactors).	At least, to reduce the exclusion zone against that provided for the currently operated reactors	SSTAR, STAR-LM

Issues of achieving plant licensing with reduced off-site emergency planning requirements are discussed in more detail in Section 3.2.1., in conjunction with measures planned in response to severe accidents for pressurized water type SMRs. This discussion is also relevant to sodium cooled and lead cooled fast reactors considered in this section.

Tables 33 and 34 summarize the information on design basis and beyond design basis accidents and the acceptance criteria.

TABLE 33. SUMMARY OF DESIGN BASIS AND BEYOND DESIGN BASIS EVENTS, INCLUDING THOSE SPECIFIC FOR A PARTICULAR SMR

SMR DESIGN	LISTS OF INITIATING EVENTS	EVENTS SPECIFIC TO A PARTICULAR SMR
4S-LMR	Lists of initiating events for DBA and BDBA have been defined and are presented as a summary or examples (ANNEX VIII). The events were identified systematically based on consideration of the 4S operation cycle and the events postulated for the MONJU and DFBR sodium cooled fast reactors (Japan). The lists of events typical of LWRs were also taken into account. On a broad scale, the BDBA are divided into two big groups that are anticipated transients without scram (ATWS) and accidents without scram (AWS). The ATWS comprise the sequences in which one of the reactor shutdown systems does not operate for any reason. The AWS groups the sequences more severe than ATWS, which include failures of more than one redundant system, e.g. failures of both pumps, both shutdown systems, and one or both of decay heat removal systems.	- Failure in insertion of the ultimate shutdown rod; - Failure in the operation of pre-programmed moveable reflector.

SMR DESIGN	LISTS OF INITIATING EVENTS	EVENTS SPECIFIC TO A PARTICULAR SMR
SSTAR, STAR-LM	With the new 10 CFR Part 53 regulation being considered currently (see ANNEX IX), a limited set of traditional design basis accidents have been identified, including loss of heat sink, in-vessel heat exchanger tube rupture, transient overcooling, transient overpower/reactivity insertion, and loss of load. The list of beyond design basis accidents has also been identified that includes failure to scram due to the assumed failure of both safety grade active shutdown systems.	- Cessation of heat removal from in-vessel heat exchangers by CO_2 working fluid with or without scram; - Transient overcooling due to initiating event on supercritical CO_2 Brayton cycle secondary side.

Table 33 also lists the features that are specific for the considered SMRs but not for a reactor line as a whole. For the sodium cooled 4S-LMR, these are failure in insertion of the ultimate shutdown rod and failure in the operation of pre-programmed moveable reflector, in view of the fact that these design features are unique to the 4S-LMR. As both SSTAR and STAR-LM are being designed with a non-conventional CO_2 based Brayton cycle power circuit, the specific events are indicated as those related to disruption in the operation of this power circuit.

TABLE 34. SUMMARY OF ACCEPTANCE CRITERIA

SMR DESIGN	DETERMINISTIC ACCEPTANCE CRITERIA	PROBABILISTIC ACCEPTANCE CRITERIA (OR TARGETS)
4S-LMR	Acceptance criteria for DBA are based on the experience with conventional light water reactors and previous design experience with sodium cooled fast reactors; specifically, the criteria that have been applied in the Clinch River Breeder reactor project are used (see ANNEX VIII); Acceptance criteria for ATWS and AWS are presented explicitly; see ANNEX VIII.	The acceptance criteria for DBA are risk-informed, as indicated by Table VIII-4 in ANNEX VIII, and envelope both normal operation and anticipated and the unlikely and very unlikely events (frequency down to 10^{-6}/year), which in the 4S are treated as design basis events. The acceptance criteria for ATWS and AWS are specified in deterministic way, with no frequency being indicated.
SSTAR, STAR-LM	It is expected that development of the SSTAR (and even more so the STAR-LM) would take place on a timescale consistent with application of the new risk-informed and technology-neutral 10 CFR 53 regulations, which would provide a basis for the definition of the acceptance criteria. No further details were provided.	

The 4S-LMR appears to be the only SMR concept in this report for which the acceptance criteria for design basis accidents are specified in a risk-informed way; see ANNEX VIII. Addressed within the design basis are events with the frequency as low as $10^{-6}\times1$/year. Different from it, the acceptance criteria for severe accidents, which in the case of the 4S-LMR include extremely rare failures of more than one redundant system, are specified in a deterministic way, with no frequency indicated.

For the SSTAR and STAR-LM, an expectation of new technology-neutral and risk informed regulations to arrive in time with the design completion is mentioned, but no details are provided regarding the acceptance criteria themselves.

Table 35 summarizes the design features for protection against external event impacts, while Table 36 lists measures foreseen in response to severe accidents.

TABLE 35. SUMMARY OF DESIGN FEATURES FOR PROTECTION AGAINST EXTERNAL EVENT IMPACTS

SMR DESIGN	AIRCRAFT CRASH / EARTHQUAKES	OTHER EXTERNAL EVENTS
4S-LMR	The reactor vessel is located in a shaft below the ground level, which, together with the containment and a relatively small footprint of the plant, contributes to an increased protection against aircraft crash/ The reactor building is isolated horizontally by seismic isolators; the reactor 'tiny' shape results in a higher characteristic frequency; therefore, the design is expected to be rigid against a vertical shock.	The capability of the plant to survive all postulated accidents relying only on the inherent and passive safety features without the need of the operator intervention, emergency team actions, and the external power and water supply, is rated as an important feature contributing to protection of the plant against external event impacts. No further details were provided.
SSTAR, STAR-LM	The reactor vessel is located in a shaft below the ground level, which, together with the containment and a relatively small footprint of the plant, contributes to an increased protection against aircraft crash/ No information was provided regarding seismic design	The capability of passive load following and "passive shutdown" provided by the inherent and passive safety features could be viewed as an important feature contributing to protection of the plant against external event impacts. No further details were provided.

For both, the 4S-LMR and the SSTAR and STAR-LM, strong reliance on the inherent and passive safety features expected to make unnecessary the operator intervention, the emergency team actions and the external power and water supplies, via ensuring a 'passive shutdown' capability of the reactor, are mentioned as factors important for the protection against both internal and external event impacts and combinations thereof.

TABLE 36. SUMMARY OF MEASURES PLANNED IN RESPONSE TO SEVERE ACCIDENTS

SMR DESIGN	MEASURES
4S-LMR	The safety analyses have shown that the 4S-LMR fuel never melts under any hypothetically postulated conditions, such as ATWS or AWS. Some fuel pins with maximum cladding temperature might fail in more severe AWS events. The analyses performed for hypothetical conditions when all fuel element claddings fail, show the dose equivalent to be 0.01 Sv at a distance of 20 m from the reactor. No measures beyond this boundary are required.
SSTAR, STAR-LM	It is envisioned that the exclusion zone for SSTAR and STAR-LM may at least be reduced in size as a result of the inherent safety features and the expected low probability of radioactive material release relative to light water reactor designs having similar power level. No further details were provided.

The design features of sodium cooled and lead cooled fast SMRs addressed in this report fit in within the fundamental requirements suggested in the IAEA safety standard “Safety of the Nuclear Power Plants: Design Requirements” [7].

However, all considered fast-spectrum SMR designs are being developed to offer several unique qualities, such as:

(1) A “passive shutdown” capability, i.e., the capability to bring the reactor to a safe low power state with balanced heat production and passive heat removal, and with no failure to barriers preventing radioactivity release to the environment; all relying on the inherent and passive safety features only, and with practically indefinite grace period;

(2) Very low pressure in the primary coolant system, challenging the notion of a primary pressure boundary used all throughout the safety standard [7];

(3) Design basis events encompassing events with occurrence frequencies as low as 10^{-6} 1/year and including combinations of the unprotected transients [2, 3], each of which is rated severe for the current generation of light water reactors.

The designers of fast spectrum SMRs target licensing within the currently established national regulatory framework but mention that further elaboration of national regulatory norms toward technology-neutral and risk-informed approach could facilitate licensing consideration and further design improvements.

As an example, the recently published IAEA report *Proposal for a Technology-Neutral Safety Approach for New Reactor Designs* [13] suggests that “the means for shutting down the reactor shall consist of a minimum of two lines of protection (shutdown mechanisms – whether they be control rods or *inherent feedback features* of the core design) required to achieve the mission within the reliability requirements for safety”.

3.2.5. Non-conventional designs

Non-conventional designs are represented in this report by the Compact High Temperature Reactor (CHTR) concept of a small very high temperature reactor developed by the Bhabha Atomic Research Centre (BARC) of India. Description of the passive safety design features of the CHTR is provided in ANNEX X; more detailed design description of the CHTR is given in the report [3].

The CHTR of 100 kW(th) is being designed as a semi-autonomous “power pack” for operation in remote areas and, specifically, for advanced non-electrical applications, such as hydrogen production. The CHTR could also be viewed as a prototype of somewhat larger, but still fitting into a SMR range, future reactors. It is a non-conventional reactor merging the technologies of high-temperature reactors with pin-in-block type TRISO fuel and lead-bismuth cooled reactors. The core uses ^{233}U-Th based fuel of HTGR type with BeO moderator blocks, while the coolant is lead-bismuth eutectic. The reactor has an essentially thermal spectrum of neutrons and uses heat pipe systems to deliver heat to process heat applications, as well as to remove heat from the core during postulated accident conditions.

Figure 11 shows a schematic of the CHTR primary circuit loop.

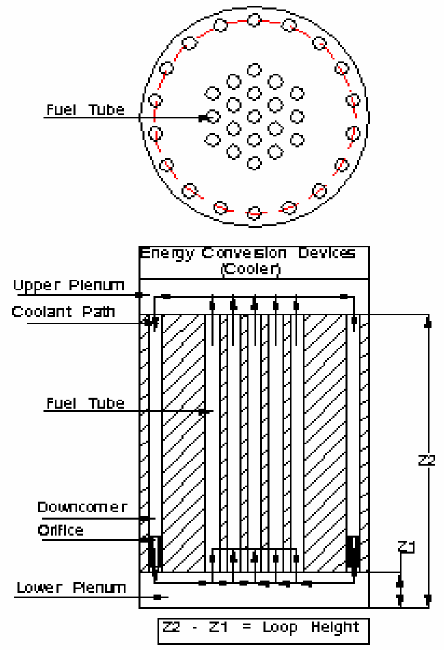

FIG. 11. Schematic view of CHTR primary circuit loop.

Tables 37 to 41 summarize the design features of the CHTR contributing to different defence in depth levels.

TABLE 37. DESIGN FEATURES OF CHTR CONTRIBUTING TO LEVEL 1 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED
1	Low core power density	Prevention of failure through increased temperature margin; lower non-nuclear (thermal) energy stored in the core
2	Heat removal from the core by natural circulation under normal operating conditions	Elimination of loss of flow accidents
3	Low overall reactivity margin of the reactor core, provided by design and, specifically, by the use of burnable poison to compensate for reactivity change due to fuel burn-up	Limitation of the scope of transient overpower accidents due to inadvertent control rod withdrawal by reducing the worth of control rods
4	Use of all ceramic core with high heat capacity and high temperature margins	Prevention of failure through increased temperature margin

The design features contributing to Level 1 of defence in depth, “Prevention of abnormal operation and failure”, listed in Table 37, are intended to provide high margins to fuel failure, low overall reactivity margin in the reactor core, and to exclude loss of flow accidents by relying on heat removal by natural circulation in all operation modes.

TABLE 38. DESIGN FEATURES OF CHTR CONTRIBUTING TO LEVEL 2 OF THE DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED
1	Negative reactivity effects (void, power, temperature, etc.) achieved with the use of the lead-bismuth coolant; specifically, high negative Doppler coefficient, achieved through the selection of an appropriate fuel composition	Higher degree of reactor self-control in abnormal operation
2	Use of all ceramic core with high heat capacity and high temperature margins	Slow progression of transients due to abnormal operation, simplifying the control
3	Increased reliability of the control system achieved through the use of passive power regulation system; this system passively inserts negative reactivity to the core when temperature increases beyond the allowable limits	Passive control of power and temperature
4	The use of two independent passively operating shutdown systems	Prevention of abnormal operation progression into a design basis accident

For Level 2 of defence in depth, “Control of abnormal operation and detection of failure”, these are negative reactivity effects, high thermal inertia of the core structures, the passive power regulation system (based on a gas expansion device), and the two independent passive shutdown systems that make a major contribution; see Table 38. It is remarkable that the objectives of Level 2 of the defence in depth are expected to be fully met by passive means, independently of the operator intervention.

TABLE 39. DESIGN FEATURES OF CHTR CONTRIBUTING TO LEVEL 3 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED
1	Very high boiling point of the Pb-Bi coolant (1670°C)	Prevention of coolant boiling in design basis accidents that otherwise might result in accident propagation to beyond design basis conditions
2	The use of a high heat capacity ceramic core	Increased grace period
3	The use of two independent passive systems to transfer reactor core heat to the outside environment, one comprising a gas gap filling system, and another – a heat pipe based system	Increased reliability of heat removal from the reactor core in design basis accidents
4	The use of an independent system based on carbon-carbon composite heat pipes to transfer heat from the reactor core to the atmosphere in the case of a loss of coolant (additional to the two systems mentioned in item 2[14])	Increased reliability of heat removal from the reactor core in design basis accidents
5	The use of two independent shutdown systems, one comprising passively activated gravity driven drop of mechanical shut-off rods, and the other employing a temperature feedback gas-expansion	Reactor shutdown

[14] Each of the indicated passive decay heat removal systems is capable of dissipating 200% of the rated reactor power

For Level 3 of defence in depth, “Control of accidents within design basis”, there are three groups of features that make a major contribution:

(1) Inherent safety features, provided by design and intended to prevent accident propagation into BDBA conditions, with an increased grace period, positions 1–2 of Table 39;

(2) Three independent passive systems for heat removal in postulated accident conditions, two — based on heat pipes, and one providing for the filling of the gas gap around the reactor by lead-bismuth to increase heat conductivity and facilitate heat removal to the environment. Of these, one heat pipe based system and the system based on gas gap filling, are dedicated passive safety systems; another heat-pipe based system is a normal operation heat removal system capable of carrying out a safety function in accidents;

(3) Two independent passive shutdown systems, one based on passively activated gravity-driven drop of mechanical rods, and another — using the effect of a temperature feedback gas-expansion to increase neutron leakage from the core and insert a negative reactivity.

All passive safety systems of the CHTR are safety grade. The CHTR active safety systems, which are the reset systems of passive shutdown and passive gas gap heat removal, as well as the system of liquid metal draining from the gas gaps to a reservoir, and a de-fuelling and refuelling system, are all non safety grade.

TABLE 40. DESIGN FEATURES OF CHTR CONTRIBUTING TO LEVEL 4 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED
1	Low pressure of the Pb-Bi coolant – the coolant flows out very slowly in the case of a break of the primary boundary and eventually solidifies	Prevention of radioactivity release to the environment via relying on relatively low non-nuclear energy stored in the primary coolant system
2	Proven fission product confinement capability of the TRISO coated particle fuel at high temperatures (1600°C in the long-term and up to 2100°C in the short term)	- Prevention of core melting; and - Limitation of fission product release in severe accidents.
3	Large heat capacity of the ceramic core	Slow fuel temperature rise with more than 50 minutes being available even when all heat sinks are lost
4	The use of heat sink located outside of the outer steel shell of the reactor	Increased reliability of heat removal in severe accidents
5	Reactor located in an underground pit and covered by a reinforced concrete barrier (the confinement structure); additionally, the steel vessel is foreseen	Prevention of radioactivity release to the environment; protection from the impacts of severe external events
6	High density of Pb-Bi coolant, comparable to the density of the fuel	Prevention of re-criticality — in the case of a severe accident with fuel failure, the fuel would be carried over to the upper part of the reactor, preventing the re-criticality

For Level 4 of defence in depth, “Control of severe plant conditions, including prevention of accident progression and mitigation of consequences of severe accidents”, the contribution comes from the intrinsic features such as low pressure of Pb-Pi coolant; from the proven fission product confinement capability of the high-temperature HTGR type fuel; and from large heat capacity of the ceramic core ensuring slow fuel temperature rise even in the case when all heat sinks are lost; see positions 1–3 of Table 39. In addition to this, heat sink is located outside of the outer steel shell of the reactor, and the reactor itself is located in an underground pit covered by the reinforced confinement structure. Steel shell (vessel) of the reactor is expected to act as a second containment. Finally, like in the case of lead cooled fast reactors, re-criticality is prevented by high density of the Pb-Bi coolant, via passive carry over of molten fuel or fuel debris to the upper part of the reactor.

TABLE 41. DESIGN FEATURES OF CHTR CONTRIBUTING TO LEVEL 5 OF DEFENCE IN DEPTH

#	DESIGN FEATURE	WHAT IS TARGETED
1	Passive safety design features contributing to levels 1–4 of the defence in depth are expected to prevent any significant release of radioactive materials in any design basis and beyond design basis accident	No evacuation or relocation measure needed outside of the plant boundary

The designers of the CHTR rate passive safety design features contributing to levels 1–4 of the defence in depth as sufficient to meet the objective of the defence in depth Level 5; see Table 41.

Issues of achieving plant licensing with reduced off-site emergency planning requirements are discussed in more detail in Section 3.2.1., in conjunction with measures planned in response to severe accidents for pressurized water type SMRs. This discussion is also relevant to the reactors of non-conventional design considered in this section.

Tables 42 and 43 summarize the information provided by the designers of the CHTR on design basis and beyond design basis events considered and on the corresponding acceptance criteria.

TABLE 42. SUMMARY OF DESIGN BASIS AND BEYOND DESIGN BASIS EVENTS FOR CHTR, INCLUDING THOSE SPECIFIC FOR A PARTICULAR SMR

SMR DESIGN	LISTS OF INITIATING EVENTS	EVENTS SPECIFIC TO A PARTICULAR SMR
CHTR	A preliminary list of design basis and beyond design basis events has been compiled, with a short summary being provided in ANNEX X.	Nothing in particular has been specified (apparently, because the CHTR concept is unique and has no analogues)

TABLE 43. SUMMARY OF ACCEPTANCE CRITERIA

SMR DESIGN	DETERMINISTIC ACCEPTANCE CRITERIA	PROBABILISTIC ACCEPTANCE CRITERIA (OR TARGETS)
CHTR	Top-level acceptance criteria for DBA and BDBA have been formulated, see ANNEX X.	The probability of unacceptable radioactivity release beyond the plant boundary is targeted to be less than 1×10^{-7}/year

Table 44 summarizes the design features of the CHTR contributing to plant protection against external event impacts, while Table 45 summarizes measures planned in response to severe accidents.

TABLE 44. SUMMARY OF DESIGN FEATURES FOR PROTECTION AGAINST EXTERNAL EVENT IMPACTS

SMR DESIGN	AIRCRAFT CRASH / EARTHQUAKES	OTHER EXTERNAL EVENTS
CHTR	For protection against aircraft crash and missiles, the CHTR would be installed in an underground pit with low exterior profile of the reactor building; additionally, the reactor would be first provided with a low-leakage thick steel vessel to absorb energy in the case of a postulated aircraft impact/ The CHTR structures, systems and components are being designed for high level and low probability seismic events such as the operating basis earthquake (OBE) and safe shutdown earthquake (SSE); seismic isolators and dampers are planned also	Design features for protection against the impacts of natural and human induced external events are described in detail in [6]. The external events considered in plant design include earthquakes, aircraft crash, cyclones, and flooding. For protection against flooding, the reactor would be provided with a low-leakage thick steel vessel having a reduced number and size of the penetrations; additional water tight barriers and a duct would be provided for systems communicating to the control room.

Seismic design of the CHTR corresponds to the recommendations of the IAEA safety standard [8]. The protection against aircraft crash is provided by locating the reactor in an underground pit with low exterior profile of the reactor building; additionally, the reactor would be provided with a low-leakage thick steel vessel to absorb energy in the case of a postulated aircraft impact. This leak-tight vessel with minimum number of penetrations would also provide the protection against flooding. The reactor, including the steel vessel, is located in the reinforced confinement structure.

TABLE 45. SUMMARY OF MEASURES PLANNED IN RESPONSE TO SEVERE ACCIDENTS

SMR DESIGN	MEASURES
CHTR	The safety analyses performed indicate the inherent and passive features of the CHTR might be able to prevent the TRISO coated particle fuel from exceeding the limiting temperatures in postulated accidents. The design objective is to prove that no emergency evacuation or relocation measures in the public domain would be required.

The design objective for the CHTR is to prove that no emergency evacuation or relocation measures in the public domain would be required in any accidents without the operator intervention and the emergency team actions, and without the external water and power supplies.

According to its designers, the CHTR is being developed in line with the recommendations of the IAEA safety standard NS-R-1 *Safety of the Nuclear Power Plants: Design Requirements* [7], which is the basis for the currently adopted national nuclear regulations in India. In view of the designers, further design development could be facilitated by technology-neutral revisions of the indicated standard, e.g., such as suggested in the recently published IAEA-TECDOC-1570 *Proposal of a Technology-Neutral Safety Approach for New Reactor Designs* [13]. In addition to this, the risk-informed approach suggested in the abovementioned document that includes the quantitative safety goals linked to defence in depth levels could facilitate assessment of such claimed qualities of the CHTR as absence of the need in off-site emergency planning and reactor self-control in accidents relying on the inherent and passive safety features only, as provided by design.

4. BENEFITS AND NEGATIVE IMPACTS ARISING FROM THE INCORPORATION OF INHERENT AND PASSIVE SAFETY DESIGN FEATURES INTO SMRs

Discussed below are the specific positive/ negative effects of incorporation of inherent and passive safety design features that, in view of the SMR designers, affect plant characteristics in areas other than safety.

4.1. Water cooled SMRs

Table 46 summarizes the positive/ negative effects of the inherent and passive safety design features of the pressurized water type SMRs in areas other than safety, based on the inputs provided by the SMR designers in ANNEXES I–V to this report.

As it can be seen from Table 46, relying more on inherent and passive safety features and passive safety systems as compared to traditional solutions based on the active safety systems is in all cases a trade-off regarding the plant economy.

TABLE 46. SUMMARY OF POSITIVE/ NEGATIVE EFFECTS FROM INCORPORATION OF INHERENT AND PASSIVE SAFETY DESIGN FEATURES INTO PRESSURIZED WATER TYPE SMRS - AREAS OTHER THAN SAFETY

#	DESIGN FEATURE	POSITIVE EFFECTS	NEGATIVE EFFECTS	SMR DESIGNS
1	Elimination of liquid boron reactivity control system	- Decrease of capital and operation costs; plant simplification; - Relaxed concerns with relation to human actions of malevolent character	Certain deterioration of fuel cycle characteristics	KLT-40S, CAREM-25, SCOR

#	DESIGN FEATURE	POSITIVE EFFECTS	NEGATIVE EFFECTS	SMR DESIGNS
2	Integral primary circuit with internal steam generators and control rod drives	- Core damage frequency (CDF) and large early release frequency (LERF) are reduced, allowing the economy of twin-unit and multi-unit plants and, potentially, the economy from reduced or eliminated emergency planning;	- Increased cost owing to the limited power of a single module[15]	IRIS, CAREM-25, SCOR
		- Decreased plant cost, resulting from compact primary circuit, the use of a compact steel containment, and reduced siting area;	Increased cost of a larger reactor pressure vessel[16]	IRIS, SCOR
		- Reduced operation and maintenance costs resulting from simplified operation and maintenance;		IRIS
		- Higher capacity factor;		IRIS
		- Possibly, reduced security costs resulting from the "inherent security"		IRIS
		- Certain economy achieved via longer reactor pressure vessel lifetime owing to a reduced fast neutron fluence		CAREM-25, IRIS
		- Reduced plant costs resulting from simplification of certain safety systems		CAREM-25, IRIS
3	Modular design of the reactor unit	Decrease of the plant costs resulting from compactness of the reactor unit and smaller dimensions of the containment	Certain deterioration of maintainability as compared to loop type plants	KLT-40S
4	Totally leak-tight reactor coolant system	Decrease in the operation costs resulting from a decrease in the amount of radioactive waste		KLT-40S
5	Primary coolant pressure boundary enclosed in a pressurized low-enthalpy water containment	- Could facilitate cost reduction via plant licensing without off-site emergency planning; - Complicates unauthorized access to fuel.	Negatively affects plant costs via the incorporation of: - Additional pressure vessel; - Control rod drive mechanisms able to operate in cold water; Complicates plant maintainability through lower accessibility of the primary pressure boundary.	MARS

[15] With a potential of being counteracted by modular construction of multiple units at a site
[16] Counteracted by reduced containment size and reduced plant footprint

#	DESIGN FEATURE	POSITIVE EFFECTS	NEGATIVE EFFECTS	SMR DESIGNS
6	Reduced number of safety grade systems and reduced number of components requiring maintenance	Improved plant economy owing to simplified operation and maintenance and reduced operation waste		MARS
7	Incorporation of passive safety systems		Increase of plant construction and maintenance costs	KLT-40S
8	Use of self-actuated devices in passive systems		Increase of plant construction and maintenance costs	KLT-40S
9	All safety grade safety systems are passive	- Reduced operation and maintenance costs resulting from a reduced complexity and improved reliability of the plant; - Added resilience to sabotage and other malevolent actions.		IRIS
10	Natural convection of the coolant	Reduced operation and maintenance costs owing to design simplification and elimination of main coolant pumps	Increased specific cost of reactor pressure vessel; potentially increased complexity of reactor operation (start-up, etc.)	CAREM-25
11	Increased reliance on natural convection of the coolant	Decrease of costs owing to simplified operation and maintenance	Increased specific cost of reactor pressure vessel; potentially increased complexity of reactor operation (start-up, etc.)	SCOR
12	Relatively low core power density and coolant temperature facilitating the use of a passive emergency core cooling system with an infinite grace period, actuated upon flow rate decrease	Essential simplification of the design, with cost savings	Increased plant costs owing to limited reactor power and energy conversion efficiency	MARS

Regarding the solutions intended to eliminate certain types of accidents or prevent their consequences by design, see positions 1–6 of Table 46, the commonly mentioned expected benefits are:

- Plant capital costs decreased due to compact primary circuit and compact containment (except for the MARS);
- Plant operation costs decreased due to simplicity of operation and maintenance, specifically, due to a reduction of the number of systems requiring maintenance;
- Plant costs decreased due to elimination/ reduction of the off-site emergency planning;

- Plant costs decreased via an enhanced options to build several plants at a site or to use twin- or multiple-unit plants, owing to the decreased core damage frequency and large early release frequency;
- Less concerns regarding human actions of malevolent character and, potentially, cost reduction resulting from such 'inherent security' of the plant.

At the same time, the same solutions are expected to result in the following negative implications:

- Increased plant capital cost owing to the limited power of a single module (potentially counteracted by modular construction of multiple units at a site);
- Increased cost of a larger reactor pressure vessel (or additional pressure vessel for the MARS design);
- Certain deterioration of burn-up cycle characteristics (in the case when liquid boron system is abandoned) or maintainability (for a compact modular design of the KLT-40S and for the MARS design with the additional pressure vessel).

In nearly all cases, the abovementioned benefits and disadvantages have a potential to counteract each other; for example, increased specific capital costs for a single unit plant could possibly be counteracted by modular construction of multiple units at a site; increased vessel cost could be counteracted by a reduced containment cost; and certain deterioration of maintainability could be counteracted by a reduced number of systems needing maintenance.

Regarding positive/negative impacts resulting from the application of passive safety systems, the opinions of SMR designers may vary. For example, the designers of the KLT-40S see only negative cost implication of passive safety systems, such as increased construction and maintenance costs; positions 7–8 of Table 46. The designers of the IRIS see only positive cost implications of passive safety systems, such as the reduced operation and maintenance costs and the enhanced resilience to sabotage; position 9 of Table 46. Other designers mention both, positive and negative features. The opinion of the designers may also be conditioned by a specific passive safety system type, i.e., the expectations might be different for, say, gravity driven passively actuated shutdown system and natural convection based decay heat removal system.

4.2. Pressurized light water cooled heavy water moderated reactors

TABLE 47. SUMMARY OF POSITIVE/ NEGATIVE EFFECTS FROM INCORPORATION OF INHERENT AND PASSIVE SAFETY DESIGN FEATURES INTO THE AHWR – AREAS OTHER THAN SAFETY

#	DESIGN FEATURE	POSITIVE EFFECTS	NEGATIVE EFFECTS
1	Core cooling by natural convection	- Simplifies design and maintenance, eliminates nuclear grade main circulating pumps, their drives and a control system, contributing to reduced plant cost; - Reduces the power requirement for plant operation, resulting in higher net plant efficiency and lower specific capital cost.	Increased diameter and length of the piping; with associated increase in plant cost.

Table 47 summarizes the positive/negative implications of the inherent and passive safety design features of the AHWR in areas other that safety, based on the inputs provided by the AHWR designers in ANNEX VI to this report.

As it can be seen from Table 47, the designers of the AHWR foresee both positive and negative impacts on plant economy resulting from the core cooling by natural convection in all modes. Simplified design and maintenance and elimination of pumps, and the resulting smaller power requirements for plant own needs, are on the positive side regarding the plant economy, while increased diameter and length of the piping are on the negative side.

4.3. High temperature gas cooled reactors

Table 48 summarizes the positive/ negative effects of the inherent and passive safety design features of the GT-MHR in areas other that safety, based on the inputs provided by the GT-MHR designers in ANNEX VII to this report.

TABLE 48. SUMMARY OF POSITIVE/ NEGATIVE EFFECTS FROM INCORPORATION OF INHERENT AND PASSIVE SAFETY DESIGN FEATURES INTO THE GT-MHR – AREAS OTHER THAN SAFETY

<table>
<tr><th>#</th><th>DESIGN FEATURE</th><th>POSITIVE EFFECTS</th><th>NEGATIVE EFFECTS</th></tr>
<tr><td>1</td><td>Helium coolant properties</td><td></td><td>Primary circuit and coolant costs are increased, taking into account helium volatility</td></tr>
<tr><td>2</td><td>Graphite as a structural material for the reactor core</td><td></td><td>- Facilities should be constructed to produce graphite of specified properties;
- Increase of reactor core cost;
- Need to dispose of large volumes of graphite.</td></tr>
<tr><td>3</td><td>Low core power density</td><td></td><td>Decrease of specific economic indices; increase of the reactor cost.</td></tr>
<tr><td>4</td><td>Annular reactor core with a high surface-to-volume ratio to facilitate core cooling</td><td rowspan="2"></td><td rowspan="2">Increase of the reactor vessel dimensions and cost</td></tr>
<tr><td>5</td><td>Central reflector</td></tr>
<tr><td>6</td><td>Heat resistant steel used for the reactor internals and the reactor vessel</td><td></td><td>Increase of reactor cost</td></tr>
<tr><td>7</td><td>TRISO coated particle fuel capable of reliable operation at high temperatures and fuel burn-ups</td><td></td><td>- Increase of fuel cost;
- Fuel production facilities need to be constructed.</td></tr>
</table>

#	Design feature	Positive effects	Negative effects
8	Design to limit fuel temperature in accidents by passively removing heat through the vessel wall, limiting total core power		Limited option to benefit from economy of scale, owing to limited unit capacity
9	Containment designed to retain the helium-air fluid and to withstand external loads		Increase of NPP cost
10	No large diameter pipelines in the primary circuit and no steam generators	Decrease of reactor cost	

Although this was not requested by the suggested format, the designers of the GT-MHR apparently base their judgement on a comparison between typical light water reactors and the GT-MHR. The reason is apparently that, all HTGR designs incorporate somewhat similar inherent and passive safety design features, with no active system based HTGR alternative being available for comparison [2]. Within the comparison as it was done, all major departures of a HTGR from a light water reactor (LWR) result in a plant cost increment. Such departures include the use of helium and graphite, annular design of the reactor core, the use of the TRISO coated particle fuel, and the use of the containment designed to retain the helium-air fluid, as well as to withstand external loads; see Table 48. The positive impacts in plant costs are expected from the elimination of large diameter piping in the primary circuit and from the absence of steam generators.

In addition to what is suggested by the designers in ANNEX VII and in Table 1, it could be recalled that the abovementioned departures of the GT-MHR (and the HTGRs in general) from LWRs altogether could enable an increase plant efficiency up to ~50% against ~32% in LWRs, which is probably capable of counteracting the cost penalties specified in Table 48. Also, it could be recalled that nearly all HTGRs provide for multi-module plant configurations [2], which is yet another factor capable of counteracting the abovementioned cost increase for a single module plant.

4.4. Sodium cooled and lead cooled fast reactors

Table 49 summarizes the positive/ negative effects of the inherent and passive safety design features of the sodium cooled and lead cooled fast SMRs in areas other that safety, based on the inputs provided by the designers in ANNEX VIII and ANNEX IX to this report.

As it can be seen from Table 49, the designers of the 4S-LMR have provided no information regarding positive or negative effects of the inherent and passive safety design features in areas other than safety, so that the inputs to this table are limited to those for the lead cooled SSTAR and STAR-LM – the concepts that are still at a feasibility study stage. Like in the case of HTGRs, the table incorporates an implicit comparison with present day LWRs, except for the cases when the SSTAR and STAR-LM incorporate features not typical for other lead or lead-bismuth cooled reactors, such as CO2 based Brayton cycle for electricity production.

TABLE 49. SUMMARY OF POSITIVE/ NEGATIVE EFFECTS FROM INCORPORATION OF INHERENT AND PASSIVE SAFETY DESIGN FEATURES INTO SODIUM COOLED AND LEAD COOLED FAST SMRS - AREAS OTHER THAN SAFETY

#	DESIGN FEATURE	POSITIVE EFFECTS	NEGATIVE EFFECTS	SMR DESIGNS
1	Positive / negative effects of passive safety design features on economics, physical protection, etc. have not been investigated yet.			4S-LMR
2	Use of lead (Pb) as a coolant	Lack of chemical interaction with working fluid enables elimination of intermediate heat transport circuit, reducing capital and operating costs	- Weight resulting from high Pb density may require greater vessel thicknesses, increasing capital costs; - Coolant chemistry control/ filtering systems needed to prevent erosion/corrosion effects contribute to increased cost	SSTAR, STAR-LM
3	Use of transuranic nitride fuel	- Transuranics are self-protective in safeguards sense; - Transuranic nitride fuel together with fast spectrum core and closed fuel cycle has a potential to reduce fuel costs		
4	Natural circulation heat transport	Natural circulation cooling enabled by Pb coolant properties eliminates main coolant pumps, contributing to reduced plant cost	Need for height separation of thermal centres between heat exchangers and core may require taller reactor and guard vessels, increasing capital costs	
5	Large reactivity feedbacks from fast spectrum core enabling passive load following and passive shutdown	Enhances reliability and reduces operator requirements potentially reducing operating costs		
6	Low burn-up reactivity swing over long core lifetime/ refuelling interval, reducing reactivity investment in each control rod	Core is fissile self-sufficient with conversion ratio near unity such that the spent core can be reprocessed to further utilize its energy content, influencing positively upon fuel economics		
7	Escape path for gas/ void to reach free surface in the primary coolant system, provided by design		Requires slightly greater reactor and guard vessel diameters, increasing capital costs	
9	Supercritical carbon dioxide Brayton cycle energy conversion with CO_2 working fluid that does not react chemically with Pb primary coolant	- Lack of chemical reaction between primary Pb and CO_2 working fluids enables elimination of intermediate coolant circuit, reducing capital and operating costs; - Use of supercritical carbon dioxide Brayton cycle with smaller turbo-machinery components than Rankine saturated steam cycle reduces plant capital and operating costs.	- Research and development costs will be required for supercritical CO_2 Brayton cycle; - Need to contain CO_2 with potential activity entrained from Pb coolant released from the reactor system following in-vessel heat exchanger tube rupture impacts upon containment requirements, potentially increasing the containment building costs; - Need to preclude radiolytic decomposition of CO_2 may require additional shielding of in-vessel Pb-to-CO_2 heat exchangers, potentially increasing the reactor system costs.	

Specifically, the CO_2 based Brayton cycle is viewed as a factor contributing to higher prototype plant costs via a high cost of the R&D still needed to prove the viability of such an option for the power circuit; see position 9 of Table 49. Being emplaced, such a cycle could, however, contribute to reduced costs via compact sizes of the turbo-machinery and via elimination of the intermediate circuit[17].

For all other passive features of the SSTAR and STAR-LM, the expected effects in areas other than safety are specified either positive or positive and negative; see positions 1–7 of Table 49. Toward the positive side are:

- Lack of chemical interactions and elimination of intermediate heat transport system;
- Increased reactor self-control and simplicity of operation, owing to natural convection cooling and optimum reactivity feedbacks, with a potential for lower operating costs;
- Self-sufficiency on fissile transuranic materials (a closed cycle is required to benefit from this) and intrinsic proliferation resistance features of the transuranic fuel (high content of ^{238}Pu — an isotope which, due to α-decay, produces significant amount of residual heat that would complicate or even make practically impossible to create a nuclear weapon — and trans-plutonium isotopes, spoiling the effectiveness of fissile material for a weapon purpose).

Toward the negative side are:

- Higher required thickness of the reactor vessel, owing to large weight resulting from lead being used as a coolant, and resulting in higher vessel costs;
- Cost increase owing to the required systems of coolant chemistry control and filtering, needed to prevent corrosion/ erosion;
- Higher vessel costs owing to increased vessel height and diameter needed to assure natural convection core cooling and the escape path for gas/ void to reach free surface in the primary coolant system.

Altogether, the above discussed indications of positive/ negative effects of the inherent and passive safety features of the SSTAR and STAR-LM in areas other that safety should be viewed as very preliminary, because of the pre-conceptual design stage of these reactor concepts.

4.5. Non-conventional designs

Table 50 summarizes the positive/ negative implications of the inherent and passive safety design features of CHTR — the only non-conventional SMR concept considered in this report — in areas other that safety, based on the inputs provided by the designers of the CHTR in ANNEX X.

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[17] It should be noted that all known designs and concepts of lead cooled reactors foresee no intermediate heat transport system, even if steam turbine cycle is used for power conversion, which is most common [18].

TABLE 50. SUMMARY OF POSITIVE/ NEGATIVE EFFECTS FROM INCORPORATION OF INHERENT AND PASSIVE SAFETY DESIGN FEATURES INTO THE CHTR – AREAS OTHER THAN SAFETY

#	DESIGN FEATURE	POSITIVE EFFECTS	NEGATIVE EFFECTS
1	Natural convection of heavy metal coolant	Saving in cost due to the absence of pumps and associated components, and due to simplified design and maintenance	
2	Thorium fuel cycle with TRISO coating particle based fuel configuration; low core power density selected for the demonstration prototype	Increased proliferation resistance	Higher specific cost of reactor due to lower core power density and because TRISO particles occupy larger volume as compared to conventional fuel
3	Heat-pipe based heat transfer to secondary system	Simplified design and maintenance, saving in cost due to the absence of heat exchanger and associated components	
4	Passive power regulation system	Simplified design and maintenance, saving in cost with respect to a conventional complex mechanism based system	
5	Passive heat removal based on gas gap filling with molten metal in accident conditions	Simplified design and maintenance, the associated reduction in cost	

For the CHTR, feasibility study has been completed in 2006. Like in the case of the SSTAR and STAR-LM concepts, the design stage may be too early to assess positive/ negative implications of the inherent and passive safety design features in areas other than safety. However, different from the SSTAR and STAR-LM, conceptual design development for the CHTR with an extensive testing programme showed a noticeable progress in the Bhabha Atomic Research Centre (BARC) of India at the time when this report has been prepared.

According to Table 50 and ANNEX X, the positive implications of the inherent and passive safety design features of the CHTR could be:

- Cost savings due to the absence of pumps and steam generators, with heat pipes being used for heat transfer to the secondary circuit;
- Cost savings due to simplified design and maintenance, owing to the passive power regulation system and passively actuated passive decay heat removal system based on gas gap filling with molten metal;
- Increased proliferation resistance owing to the use of TRISO fuel within thorium fuel cycle, possibly resulting from the absence of a commercial technology of TRISO fuel reprocessing and from radiation barriers provided by the daughters of ^{232}U in the thorium cycle (for more details see [2 and 3]).

The anticipated negative implication is higher specific plant cost owing to low core power density, resulting from the use of the TRISO fuel (similar to HTGRs).

5. APPROACHES TO SAFETY SYSTEM SELECTION: ACTIVE VERSUS PASSIVE SAFETY SYSTEMS

The enveloping design approach for SMR designs considered in the present report is to eliminate as many accident initiators and/ or prevent as many accident consequences as possible by design, and then to deal with the remaining accidents/ consequences using reasonable combinations of active and passive safety systems and consequence prevention measures.

To prevent accidents, inherent safety features are used in the design, making direct contributions to defence in depth Level 1. These features may be very different for different reactor lines, e.g., eliminated piping or internal location of control rod drives in pressurized water reactors; eliminated steam generators and steam power circuit in direct cycle HTGRs; optimum combinations of reactivity effects and negative void worth in sodium cooled and lead cooled fast reactors; they are summarized in more detail below.

When available, contributions of inherent safety features to the subsequent levels of defence in depth may help reduce the hazard associate with accidents by ensuring increased reactor self-control, by slowing down accident progression, or by limiting the accident scope. Relatively high heat capacity of the primary circuit is typical here, for many reactor lines.

Certain inherent safety features, such as high temperature fission product confinement properties of fuel and high temperature margin to fuel failure, contribute directly to defence in depth levels 3 and 4.

In addition to the inherent safety features, some reliable passive features, such as additional passive structures (containment, guard vessel, or additional pressure boundary around the primary circuit, or coaxial double pipes — categorized as Category A passive systems in [12] but often referred to as inherent or by-design safety features [2, 3]), or reliable mechanisms of heat transfer, such as heat transfer by conduction and radiation via reactor core and reactor internals, or ultimate heat sink based on natural draught of air outside of the reactor vessel, could contribute to various levels of defence in depth in a way similar to inherent safety features, i.e., help to prevent certain accidents or accident consequences or reduce their scope.

With maximum possible use of the inherent and passive safety features provided by design, the remaining accident sequences are then dealt with dedicated active or passive safety systems.

There is no single approach in selecting an optimum combination of the active and passive safety systems, even for a single reactor line. A balanced view is that passive safety systems that use natural mechanisms such as gravity or buoyancy, or spring force for their operation, require no operator action to get actuated, and rely on no external power or working media supply, have a potential to make plant design, maintenance and operation more simple, to enhance plant safety under a variety of internal and external events and combinations thereof, to improve plant resilience to human actions of malevolent character (add “intrinsic security”), and to improve plant economy. At the same time, it is recognized that the incorporation of passive safety systems in the reactor designs need to be validated and tested adequately due to several issues highlighted the Appendix 1.

For a passive safety system, functional failure (i.e., a failure of the system to perform its function) may happen if the boundary conditions deviate from a specified range of the values on which the performance of the system depends. Mainly because the driving forces in passive systems are most often small, the overall balance of forces defining the functional operation of a system may easily get changed even with a small disturbance or change in the operating parameters [22, 23, 24, and 25]. The difficulties in evaluation of a functional failure of passive safety systems may be related to:

- Lack of plant data and operating experience;
- The experimental data obtained from integral facilities or even from separate effect tests being insufficient to understand system performance characteristics in normal operation and in transients and accidents;
- Lack of a clear definition of a failure mode for the passive safety systems;
- Difficulties in modelling of the physical performance of such systems; for example, for natural convection based systems, such difficulties may be related to:
  - Low flow rate of natural convection, under which the flow can be not fully developed and multi-dimensional in its nature;
  - Flow instabilities, which include flashing, geysering, density-waving, flow pattern transition instabilities, etc.;
  - Critical heat flux changes under oscillatory conditions;
  - Flow stratification with kettle type boiling, particularly, in large diameter vessels;
  - Thermal stratification in large water pools;
  - Effects of non-condensable gases on condensation, etc.
- Unknown capability of the so-called “best estimate codes” to simulate performance of passive safety systems, owing to the fact that such codes were mainly developed to model active safety systems.

Therefore, before incorporating passive safety systems into plant design, their capacity or reliability needs to be validated and tested over a broad range of states, ranging from normal power operation to transients and accidental conditions [22, 23].

In addition to what was mentioned above:

- Economics of the advanced reactors with passive safety systems should be assessed taking into account all related aspects of construction and decommissioning;
- Ageing of passive safety systems should be considered, especially, for longer plant lifetimes; for example, corrosion and deposits on heat exchanger surfaces could impair the functional performance of passive safety systems;
- Passive safety systems should be designed with a provision of easy in-service inspection, testing and maintenance, and to ensure that the dose rate to the workers is within the limits prescribed by the regulations.

With all these aspects in mind, selection of an optimum combination of the active and passive safety systems depends on a previous experience in their validation and testing and on the availability of a system prototype, on a function that the system is expected to perform, and on the considerations of redundancy, diversity and independence as measures to cope with common cause failure [7], as well as on considerations of the plant economy, operating complexity, applications, security, and other factors.

It should be noted that passive safety systems in the SMRs considered in this report are not limited to natural convection based systems for passive decay heat removal, such as emergency core cooling systems, or to passive safety injection systems, but also include passive shutdown systems, such as based on gravity or spring-force driven insertion of control rods, actuated upon flow disruption or system de-energization; passive systems of gas gap filling with (liquid metal) coolant to boost conduction for heat removal to the outside of the reactor vessel; passive mechanisms of fuel carry over from the core in the case of a fuel element failure, to avoid recriticality in fast reactors; and others.

A useful categorization of passive systems is provided in IAEA-TECDOC-626 [12]; for convenience, some definitions from this reference are reproduced in Appendix 1 to this report.

Particular approaches to the application of passive versus active safety systems applied by the designers of the SMRs considered in the present report are highlighted in Section 3.2., in conjunction with Level 3 of defence in depth. A common feature of all SMRs considered in the present report is that all of them use passive decay heat removal systems. In all cases these systems are redundant and safety grade. Regarding shutdown systems, they could be active or passive, safety grade or non safety grade, based on different principles and using different components — control rods, absorber balls, or safety injections. Where applicable, depressurization systems are provided, which in most cases are actuated passively, by safety relief valves (check valves).

All solutions with active and passive safety systems described in the present report follow the principles of redundancy, diversity and independence [7].

In the case of light water reactors, there are certain advantages regarding passive safety systems, because more experience in the validation, testing, certification or operation of such systems has been accumulated [19]. Certain, although more limited, experience is available for HTGR type reactors [17]. For SMRs of other types, extensive R&D programmes would be required and, in some cases, such programmes were already in progress at the time when this report was prepared [2, 3].

Performance assessment issues for passive safety systems are highlighted in more detail the Appendices 1 and 2.

6. SUMMARY AND CONCLUSIONS

The report presents a description of the design features used to achieve defence in depth in the eleven representative concepts of small and medium sized reactors (SMRs), representing different reactor lines The descriptions are structured to follow the definitions and recommendations of the IAEA safety standard *Safety of the Nuclear Power Plants: Design Requirements* [7], with some references made to other IAEA safety standards and publications, such as [8], [13], and [12].

The selected SMRs represent different reactor lines, intended for different applications, and targeting different deployment timeframes. The reactor lines considered are pressurized water reactors — the KLT-40S, the IRIS, the CAREM-25, the SCOR, and the MARS, targeted for co-generation or electricity production; pressurized boiling light water cooled heavy water moderated reactors — the AHWR, targeted for electricity generation with potable water production; high temperature gas cooled reactors — the GT-MHR, targeted for electricity generation and advanced non-electrical applications, including complex cogeneration with bottoming cycles; sodium cooled and lead cooled fast reactors — the 4S-LMR and the SSTAR and the STAR-LM, targeted for electricity production or cogeneration; and non conventional very high temperature designs — the CHTR, targeted for hydrogen production and other advanced non-electrical applications. Design descriptions, design status, targeted deployment dates, and applications of the SMRs considered in this report are presented in more detail in [4, 2, and 3].

One of the reactors, the KLT-40S for a floating NPP, is under construction with deployment of the plant scheduled for 2010. The IRIS, the CAREM-25, and the AHWR are likely to be commercialized by 2012–2015. The SCOR, the MARS, and the 4S-LMR have a potential to be deployed as first of a kind or prototype plants by 2015. The GT-MHR, the SSTAR, the STAR-LM, and the CHTR are targeted for deployment by 2020—2025; they are still at pre-conceptual design stages.

An enveloping design approach for the SMR designs considered in this report is to eliminate as many accident initiators and/or to prevent as many accident consequences as possible, by design, and then to deal with the remaining accidents/consequences using plausible combinations of the active and passive safety systems and consequence prevention measures. This approach is also targeted for Generation IV energy systems and, to a certain extent it is implemented in some near-term light water reactor designs of larger capacity, such as the VVER-1000, the AP1000, and the ESBWR [4].

General features of SMRs that, in view of their designers, contribute to a particular effectiveness of the implementation of inherent and passive safety design features in smaller reactors are:

- Larger surface-to-volume ratio, which facilitates easier decay heat removal, especially with a single-phase coolant;
- An option to achieve compact primary coolant system design, e.g. integral pool type primary coolant system, which could contribute to an effective suppression of certain initiating events;
- Reduced core power density, facilitating easy use of many passive features and systems;
- Lower potential hazard that generically results from lower source term owing to lower fuel inventory, lower non-nuclear energy stored in the reactor, and lower integral decay heat rate.

For the pressurized water reactors, there are three distinct design approaches, which are designs with integral primary circuit, with the reactor vessel accommodating steam generators and internal control rod drives and eliminating large diameter piping, and minimizing reactor vessel penetrations; compact modular loop-type designs with a reduced length of piping, the integral reactor cooling system accommodating all main and auxiliary systems within a leak-tight pressure boundary, and the leak restriction devices; and a design with the primary pressure boundary enclosed in a enveloping shell with low-enthalpy slowly moving water.

All pressurized water small and medium sized reactors incorporate design features to prevent loss of coolant (LOCA) accidents or reduce their scope. In addition to this, the pressurized water SMRs also incorporate features for the prevention of certain reactivity initiated accidents (integral designs of the primary circuit with in-vessel location of the control rod drives), for the smooth and slow character of transients owing to the internal or 'soft'[18] pressurization and a relatively large water inventory, and for the de-rating of events with steam generator tube rupture. Whether or not these features are unique to SMRs is an open question. For example, conceptual design studies performed for PWRs with the integral design of the primary circuit accommodating both, steam generators and control rod drives, point to an option to realize such features in the reactors of up to 1000 MW(e) capacity. However, such proposals are still at an early conceptual design stage [16]. As comes to the compact modular loop-type designs, based on the experience of the marine propulsion reactors, their maximum possible unit size (known from the design studies accomplished) is around 400 MW(e) [2]. There are no known large-capacity reactor proposals for a design with the primary pressure boundary enclosed in an enveloping shell with slowly moving water of low-enthalpy.

The advanced pressurized boiling light-water cooled heavy water moderated reactors are represented by one design (the AHWR), with its principal feature being heat removal by natural circulation in all modes. Main circulation pumps and, therefore, loss of flow accidents, are prevented by design. Maximum unit size within which such technical solution can be maintained has not been examined.

For high temperature gas cooled reactors (HTGRs), the concept considered (GT-MHR) corresponds to one of the two known fuel design options - that with the pin-in-block TRISO based fuel. HTGR concepts incorporating alternative fuel design – the pebble bed TRISO fuel – were not considered in the present report. Independent of the fuel design, all HTGRs incorporate design provisions to reduce hazard in accident scenarios that are potentially severe in reactors of other types, including loss of coolant (LOCA), loss of flow (LOFA), and reactivity initiated accidents. These provisions are based on a proven fission product confinement capability of the TRISO fuel at high temperatures and high fuel burn-ups, which also enables long-term passive decay heat removal, even from a voided reactor core, via natural processes of conduction, radiation, and convection. For the known materials of reactor vessel and known HTGR core designs, passive decay heat removal is possible only when the reactor unit power is below ~600 MW(th). Direct gas-turbine cycle HTGRs also eliminate steam generators and steam-turbine power circuit, complete with the otherwise possible initiating events.

For fast reactor line, the sodium-cooled 4S-LMR and the lead cooled SSTAR and STAR-LM concepts have been considered. Both designs incorporate optimum sets of reactivity feedbacks and other inherent safety features, provided by design, to effectively reduce the scope and hazard of certain accidents and combinations of accidents that are potentially severe in reactors of other types. Specifically, this is the case for transient overpower events.

In the 4S-LMR, the corresponding features include negative whole-core void reactivity effect, contributing to defence in depth level 3, and the absence of control rods in the core, with power being controlled via a feedwater flow rate in the power circuit. The burn-up reactivity compensation is then performed with an active system based on a very slow upward movement of the pre-programmed radial reflectors, with no feedback control. Should a reflector get stuck, the reactor would operate safely for a certain time and then get "passively shut down"[19] by the increasing negative reactivity. At the same

[18] "Soft" pressurizer system is characterized by small changes of the primary pressure under a primary coolant temperature increase.
[19] 'Passive shutdown' is used by the designers to denote bringing the reactor to a safe low-power state with

time, the drop of axial reflectors is a standard reactor shutdown feature. Altogether, the features mentioned above are unique to a small reactor size.

For the lead cooled SSTAR and STAR-LM, the inherent safety features contributing to the prevention of possible accidents or to a reduction of their scope are generally typical of the lead cooled reactor line. They include very high boiling point of lead; pool type design with a free surface of lead to allow removal of the gas bubbles from primary coolant before they enter the core; the guard vessel and reactor location in the concrete shaft; optimum sets of reactivity effects; and high heat capacity and small overall reactivity margin in the reactor core. Although some designers see it as capacity independent, the "passive shutdown" option for larger-sized lead cooled reactors needs to be further examined and proven. It should be noted that some designers mention the unit size of the lead and lead-bismuth cooled reactors to be limited from seismic considerations. According to the studies performed in Japan, this size cannot exceed ~750 MW(e), which is slightly above the SMR range boundary of 700 MW(e); see Annex XV in reference [2].

Finally, the CHTR — a non-conventional design of a lead-bismuth cooled very high temperature reactor, designed to operate with ^{233}U-Th based TRISO fuel —merges the technologies and inherent safety features of the lead cooled and HTGR type reactors, and also incorporates other features altogether intended to prevent failures through increased temperature margins, to eliminate loss of flow accidents via natural circulation, to incorporate reliable heat pipe based systems for heat removal, and to reduce the scope and hazard of transient overpower accidents by limiting the reactivity margin in the core. The application of all these features is supported by relatively small core power density, typical of a TRISO type fuel. Although the CHTR is a very small reactor of 100 kW(e), similar technologies are planned to be used in future reactors of larger capacity (up to 600 MW(th)).

The information on passive and active safety systems incorporated in the designs of the SMRs considered in this report indicates there is no single strategy, with a variety of approaches being applied in different SMRs even if they belong to the same reactor line. It is important to note that broad incorporation of the inherent and passive safety features pursued by the SMR designers to prevent certain accidents and accident consequences or reduce their scope and hazard is in several cases conditioned or facilitated by smaller reactor capacity and size. However, the design solutions used for the active and passive safety systems are, in general, not capacity-dependent. With smaller reactor capacity, it is possible to facilitate the application of passive safety features and systems, specifically, those based on the natural convection of a single-phase coolant, or those incorporating mechanisms of heat transfer by conduction and radiation.

Selection of reasonable combinations of the active and passive safety systems is based on the considerations of fitness for a specific design, validation and testing experience, regulatory practice, plant economy and plant lifetime considerations, provisions for the in-service inspection and others, and may vary from case to case.

It should be noted that all SMRs addressed in the present report incorporate redundant passive systems or passive mechanisms of decay heat removal. Regarding reactor

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balanced heat production and passive heat removal, with no failure to the barriers preventing radioactivity release to the environment; all relying on the inherent and passive safety features only, with no operator intervention and active safety systems being involved, and no external power and water supplies being necessary, and with the grace period infinite for practical purpose.

shutdown systems, a variety of approaches is proposed ranging from standard active mechanical control rods to the gravity or spring-force driven absorber insertion actuated upon de-energization or coolant flow disruption, to the passively operated safety injections, to a “passive shutdown” mechanism based on the inherent safety features of a reactor design, and to a mechanism of fuel carry over from the core in the case of a cladding failure (intended to prevent re-criticality in fast sodium cooled reactors). Depressurization and isolation systems, where applicable, often use direct action devices, e.g., check valves, to get actuated. An approach that needs to be mentioned, as it is applied in several water cooled, gas cooled and liquid metal cooled SMRs, is to have all safety systems passive and safety grade. In this, it is assumed that certain non safety grade active systems/ components of normal reactor operation are capable of making a (auxiliary) contribution to the execution of safety functions in accidents.

All SMRs considered in the present report incorporate the containment; in many cases – a double containment or a containment and a protective shell or enclosure. Compact containment design and plant embedment below the ground level are commonly mentioned as factors contributing to enhanced protection against aircraft crash.

The designers of SMRs mention that such features of their reactors as the capability to survive design basis accidents and combinations thereof relying only on the inherent and passive safety features, with no operator or emergency team interventions, and without external supplies of energy and working media, could also contribute to plant protection against a variety of natural and human induced external events.

Altogether, passive safety systems are broadly applied in the SMR designs considered. At the same time, there are potential concerns related to passive safety systems, derived from a smaller experience of reactor design with such systems. In particular, these concerns are related to the following:

- Reliability of passive safety systems may not be understood so well as that of active safety systems;
- There may be a potential for undesired interaction of active and passive safety systems;
- It may be more difficult to “turn off” an activated passive safety system, if so desired, after it has been passively actuated;
- Implications of the incorporation of the passive safety features and systems into advanced reactor designs to achieve the targeted safety goals need to be proven, and the supporting regulatory requirements need to be worked out and emplaced.

To address these and other issues related to the performance assessment of passive safety systems, the IAEA recommended coordinating a research project “Development of Methodologies for the Assessment of Passive Safety System Performance in Advanced Reactors” in 2008–2011. The objective is to determine a common analysis-and-test method for reliability assessment of passive safety system performance.

For all SMRs considered in this report, the designers expect that prototype or first-of-a-kind plants with their respective SMRs would be licensed according to the currently emplaced regulatory norms and practices in member states. Further advancement of regulatory norms could then facilitate design improvements in the next plants.

Further revisions of the IAEA safety standards toward a technology-neutral approach[20] could be of value to facilitate design development and safety qualification of non water cooled SMRs, such as the GT-MHR, the 4S-LMR, the SSTAR and STAR-LM, and the CHTR.

The designers of most of the SMRs considered in the present report foresee that safety design features contributing to defence in depth levels 1–4 [7] could be sufficient to meet the objective of the defence in depth level 5 "Mitigation of radiological consequences of significant release of radioactive materials", i.e., that the emergency planning measures outside the plant boundary might be reduced or even not needed at all. The design features of the SMRs indicated to make a contribution directly to Level 5 of defence in depth are lower fuel inventory, lower non-nuclear energy stored in the reactor, and lower integral decay heat rate of a smaller reactor as compared to the large-capacity one.

As a desired or possible feature, reduced off-site emergency planning is mentioned in the Technology Goals of the Generation IV International Forum [15], in the user requirements of the IAEA's International Project on Innovative Reactors and Nuclear Fuel Cycles (INPRO) [14], and in the recommendations of the International Nuclear Safety Advisory Group (INSAG-12) [11], with a caution that full elimination of off-site emergency planning may be difficult to achieve or with a recommendation that Level 5 of defence in depth still needs to be kept, notwithstanding its possibly decreased role. Achieving the goal of a reduced off-site emergency planning would require both, development of a methodology to prove that such reduction is possible in the specific case of a plant design, and adjustment of the existing regulations. Risk-informed approach to reactor qualification and licensing could facilitate licensing with reduced off-site emergency planning for smaller reactors, once it gets established[21]. Within the deterministic safety approach it might be very difficult to justify reduced emergency planning in view of a prescribed consideration of a postulated severe accident with radioactivity release to the environment owing to a common cause failure. Probabilistic safety assessment (PSA), as a supplement to the deterministic approach, might help justify very low core damage frequency (CDF) or large early release frequency (LERF), but it does not address the consequences and, therefore, does not provide for assessment of the source terms. Risk-informed approach that introduces quantitative safety goals, based on the probability-consequences curve could help solve the dilemma by providing for a quantitative measure for the consequences of severe accidents and by applying a rational technical and non-prescriptive basis to define a severe accident. An example of such approach is in the recently published IAEA-TECDOC-1570 *Proposal of a Technology- Neutral Safety Approach for New Reactor Designs* [13]. At the time when this report was prepared, such an approach has not been established as an IAEA safety standard.

The report provides a review of the positive/ negative effects of the incorporation of the inherent and passive safety design features of the addressed SMRs in areas other than safety, based on the inputs provided by the SMR designers in ANNEXES I–X. Toward the positive side are:

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[20] National regulations is some member states are already technology-neutral; the examples are the United Kingdom or the Russian Federation

[21] Risk-informed regulations for beyond design basis accidents are already emplaced in some member states, e.g., Argentina.

- Simplicity of plant design, resulting from a reduction of the number of systems and components, and simplicity of plant operation and maintenance, resulting from a reduced number of the systems and components requiring maintenance – both factors contribute to a reduction in plant costs;
- For many designs — reduced plant costs, resulting from a compact primary circuit design and a compact containment design;
- Simplicity of plant operation and maintenance[22], resulting from increased reactor self-control in accidents and higher margin to fuel failure, has a potential to result in reduced requirements to operating personnel and reduced plant staffing requirements — should this be accepted by the regulators, it might contribute to reduced operating costs and facilitate deployments in countries with limited infrastructure;
- For nearly all designs, a potential to benefit from cost reduction resulting from reduced or eliminated off-site emergency planning — this still needs to be proved and accepted by the regulators;
- Owing to increased reactor self-control in accidents and higher margin to fuel failure, less concerns regarding human actions of malevolent character and, potentially, the cost reduction owing to such 'inherent security' of the plant.

On the other side, for all designs considered, the implementation of inherent and passive safety design features results in an increase of the specific plant capital cost due to lower core power density or larger size of the reactor vessel needed to accommodate certain components of the primary circuit, etc. Elimination or reduction of liquid boron system (in PWR type reactors) or operation without on-site refuelling provided for in the sodium cooled and lead cooled SMRs, results in certain deterioration of burn-up cycle characteristics. Taller and broader reactor vessel or piping, necessary to enhance natural convection based heat removal, is also a factor contributing to the plant cost increase.

The designers expect that the abovementioned negative implications of passive safety design options could be counteracted by an enhanced option to build twin- or multi-unit plants at the same site (see Fig. 1 in Chapter 1.1.1); by enhanced pre-fabrication and, in some cases, by higher energy conversion efficiency; as well as by the positive implications highlighted in the previous paragraph.

[22] ANNEX IV gives an example of how operation complexity of a plant could be quantified and used in comparative assessments of different design solutions.

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APPENDIX I

PERFORMANCE ASSESSMENT OF PASSIVE SAFETY SYSTEMS

Background and experience

As it was already mentioned, broad incorporation of inherent and passive safety design features has become a ‘trademark’ of many advanced reactor designs, including several evolutionary designs and the majority of innovative SMR designs [1, 2, 3, 4, and 5]. In addition to various possible combinations of the inherent and passive safety features (sometimes referred to as by-design safety approaches [2]), all SMRs addressed in this report incorporate passive safety systems. Passive safety systems may include moving liquids or expanding solid structures, direct action devices, or stored energy sources. As suggested in IAEA-TECDOC-626 [6], those may be classified as passive systems of categories B, C, and D, accordingly, see Appendix 1. Passive safety systems require validation and testing to demonstrate and prove their reliable operation and quantify their reliability and, if necessary, adjust their design accordingly.

While individual processes may be well understood, the combinations of these processes, which determine the actual performance of passive safety systems, may vary depending on changes in the conditions of state, boundary conditions, and failure or malfunctioning of other components within the system, the circuit or the plant. Passive safety systems of category A, or inherent safety features incorporate no moving liquids or moving solid structures, direct action devices, or stored energy sources. There is a consensus that such systems have a strong advantage [6, 2, and 3]. Therefore, the issue of process performance reliability is most important for passive safety systems of Categories B, C, and D [6].

There are certain accomplishments regarding the testing, construction, licensing or validation of passive systems of Categories B, C, or D [6], such as the more recent VVER-1000 reactors and the KLT-40S of the Russian Federation, or the AP600, the AP1000, and the ESBWR of the USA [4, 7]. Experiment-based deterministic approaches to the validation of passive systems including separate-effect tests and integral tests of reactor models with subsequent qualification of analysis models and computer codes have been established and accepted by the regulators in some countries, in line with the conventional safety requirements also applied to active safety systems. The indicated deterministic approaches are generally successful with regulators when the basic technology involved is evolutionary, e.g., that of water cooled reactors, and backed by years of validation and testing and the reactor operation experience, and when passive systems are reasonably conventional in their design. When the technology is innovative or a passive safety system has a distinctly non-conventional set of features, the application of established deterministic approaches may require a multi-year resource consuming effort on the validation, testing and demonstration of the reliable operation of such a system, prior to licensing approval of the corresponding advanced NPP.

The regulations in Argentina, China, Japan, Germany, India, France, the Russian Federation, and the USA already incorporate provisions for accepting the results of probabilistic safety assessments (PSA) on a complementary basis. In order to ensure that the PSA used in the risk-informed decision making (RIDM) process is of acceptable technical quality, the effort is being made in different countries to provide PSA standards that define inherent technical features of a PSA acceptable for a regulatory body. An example is the ASME probabilistic risk assessment (PRA) Standard [8], recently endorsed by the United States Nuclear Regulatory Commission (US NRC). In line with the worldwide trends, the IAEA is developing a series of publications in the safety standards Series on PSA and RIDM. One of the latter, named the Safety Guide on ‘Development and Application of Level-1 PSA for NPPs” [9], planned to be published in 2008, would provide recommendations on the technical content of PSA studies to reliably support various PSA applications.

The general trend toward a more risk informed approach (e.g., see [10 and 11]) is pursued with a focus on what is really important from the safety perspective, in order to achieve a design that is more favourable from the cost-benefit perspective. A methodology for reliability assessment of passive safety systems would enable quantification of the reliability to treat both active and passive safety systems within a common PSA approach. Several such methodologies are under development in Europe, India, and the USA [12, 13, and 14]. What is important from a perspective of the overall risk assessment, these methodologies take into account uncertainties associated with unforeseen physical phenomena that may affect the operation of passive safety systems, worsening their reliability. All of the methodologies are at a preliminary stage of development and no consensus on a common approach has been established among their proponents at the time when this report has been prepared. Two of these methodologies are described in brief below.

Examples of methodologies for reliability assessment of passive safety systems

RMPS methodology

In the late 1990s, a methodology known as REPAS was developed cooperatively by ENEA, the University of Pisa, the Polytechnic of Milan and the University of Rome in Italy that was later incorporated in the European Commission’s reliability methodology for passive systems (RMPS) project within the European Commission’s 5th framework programme [12]. The RMPS methodology is based on the evaluation of a failure probability of a system to carry out the desired function for a given set of scenarios taking into account the uncertainties of those physical (epistemic) and geometric (aleatoric) parameters the deviations of which can lead to a failure of the system. The RMPS approach considers a probability distribution of failure to treat variations of the comparative parameters considered in the predictions of codes.

Schematics of the RMPS are shown in Fig. 1.

The RMPS methodology has been developed to evaluate reliability of the passive systems incorporating a moving fluid and using natural convection as an operation mechanism. The reliability evaluation for such systems is based, in particular, on the results of thermal-hydraulic calculations. The RMPS methodology could be structured as follows:

- Identification and quantification of the sources of uncertainties;
- Reliability evaluation of a passive system;
- Integration of passive system reliability in PSA.

The methodology is applied to a specific accident scenario in which the operation of a certain passive safety system is foreseen. When the scenario to be examined is specified, the first step — identification of the system — requires full characterization of the system under investigation to be carried out. This step includes specifying the goals of the system, the modes via which it may fail, and providing the definition of a system failure, i.e., the definition of the success/ failure criteria. Modelling of the system is also required, which is accomplished using best-estimate computer codes. Numerous sources of uncertainties present in the modelling process have to be identified. Such sources are related to the approximations in modelling of physical processes and the system geometry, and the uncertainties in input variables, such as the initial and boundary conditions. Identifying the most important thermal-hydraulic phenomena and parameters, that have to be investigated for the system, is an important part of the methodology. Such identification could be accomplished via a brainstorming of the experts with good understanding of the system functions and best-estimate code calculations, and with the use of a method of the relative ranking of phenomena. The ranking technique implemented in the RMPS project is the analytical hierarchy process (AHP). After identifying the important thermal-hydraulic parameters, the next step is to quantify their uncertainties. In the case when experimental data are not available, expert judgement would be required to identify the range of uncertainties and to select the appropriate probability density functions for a given set of variables. The methodology incorporates a sensitivity analysis, which is to determine, among all uncertain parameters, the main contributors to the risk of a system failure.

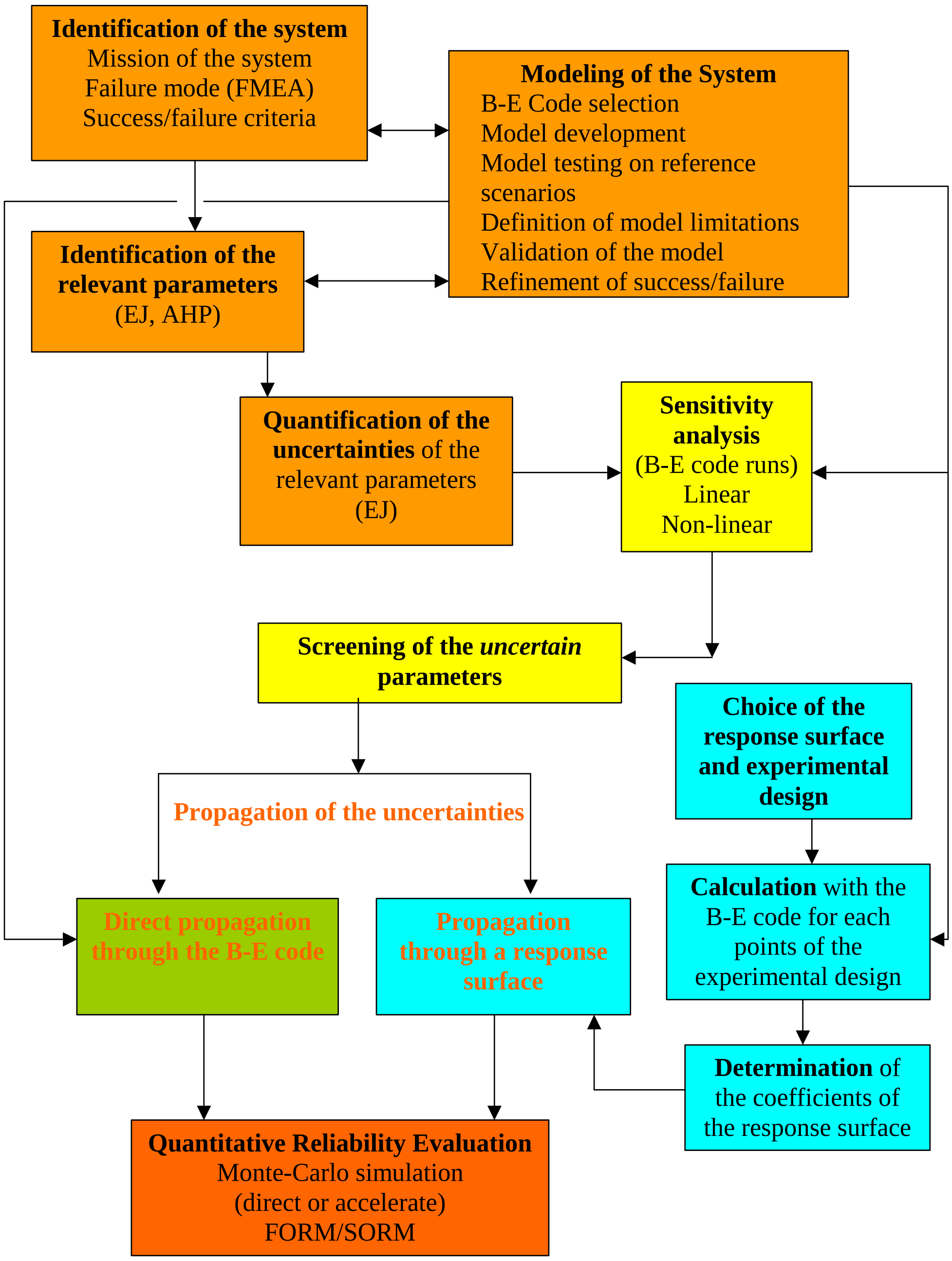

FIG. 1. Schematics of the RMPS methodology.

The second part of the methodology requires evaluating the uncertainty in the expected performance of the passive system as predicted by the thermal-hydraulic code and according to the studied scenario. Such uncertainty evaluation could be performed using the confidence intervals or the probability density functions. Within the RMPS studies, it has been found that methods giving an uncertainty range of the system performance are not very efficient for reliability estimation. Therefore, the use of a probability density function was selected as an approach to be implemented. Then, the probability density function of system performance could be directly used for the reliability estimation, once a failure criterion is given. The existing methods for such quantitative reliability evaluation are generally based on the Monte-Carlo simulations. Monte-Carlo simulations consist in drawing of the samples of the basic variables according to their probabilistic density functions and then feeding them into the performance function evaluated by a thermal-hydraulic code. An estimate of the probability of failure could then be determined by dividing the number of simulations leading to a failure of the system, by the total number of the simulation cycles. Monte-Carlo simulations require a large number of calculations; as a consequence, the calculations may be prohibitively time-consuming. To avoid the problem, two approaches are possible, which are (i) the variance reduction techniques used in Monte-Carlo methods, or (ii) the use of response surfaces. It is also possible to use approximate methods, such as first and second order reliability methods (FORM/SORM).

The final part of the methodology focuses on the development of a consistent approach for the quantitative reliability evaluations of passive systems, which would allow introducing such evaluations in the accident sequence of PSA. In the PSA of innovative reactor projects carried out until recently, only the failures of the passive system components (valves, pipes, etc.) were taken into account and not the failures of combinations of the physical phenomena on which the system performance is based. It is a difficult and challenging task to treat this aspect of passive system failure in the PSA models, with no commonly accepted practices available. Different options have been discussed within the framework of the RMPS project, but no real consensus between partners has been found. In line with the emplaced standards for Level 1 PSA models, the approach currently followed by the CEA and Technicatome of France is based on accident scenarios being presented in the form of static event trees. The event tree technique makes it possible to identify the whole variety of chains of the accident sequences, deriving from initiating events and describing different basic events corresponding to a failure or to a success of the safety systems. This method has been applied to a fictitious PWR type reactor equipped with two types of passive safety systems. The analyses of failures carried out for this reactor made it possible to characterize both, technical failures (those of valves, heat exchanger pipes, etc.), and ranges of variation of the uncertain parameters affecting the physical process. A simplified PSA has been performed starting from a single initiating event. The majority of the sequences addressed by this event tree were analysed by deterministic evaluations, using enveloping values of the uncertain parameters. For some sequences, where the definition of the enveloping cases was impossible, basic events corresponding to the failure of physical processes were added to the event tree, and quantitative reliability evaluations, based on the Monte Carlo simulations and on the thermal-hydraulic code analyses, were carried out to evaluate the corresponding failure probability. The failure probabilities obtained by these reliability analyses were fed into the corresponding sequences. Such approach allows evaluating the impact of a passive safety system on the accident scenario. In particular, for the example studied, a new design basis of the system has been proposed in order to meet in full the global safety objective assigned to the reactor.

The RMPS methodology has been applied to three types of passive safety systems, including the isolation condenser system of a boiling water (BWR) reactor, the residual heat removal system on the primary circuit of a PWR reactor, and the hydro-accumulator (HA) systems of PWR and VVER type reactors.

In the RMPS applications performed by the CEA and Technicatome of France, the thermal-hydraulic passive system acts as an ultimate system in the management of an accident scenario. Under this assumption, the characteristics of the current Level 1 PSA models remain adequate.

A test case of the RMPS methodology is currently underway for a CAREM like passive residual heat removal system within the ongoing IAEA's coordinated research project "Natural circulation phenomena, modelling and reliability of passive systems that utilize natural convection".

APSRA methodology

A different approach followed is the "APSRA" methodology developed at the Bhabha Atomic Research Centre (BARC) of India [13]. In this approach, the failure surface[23] is generated by considering the deviation of all those comparative parameters which influence the system performance.

Schematics of the APSRA methodology are shown in Fig. 2, 3.

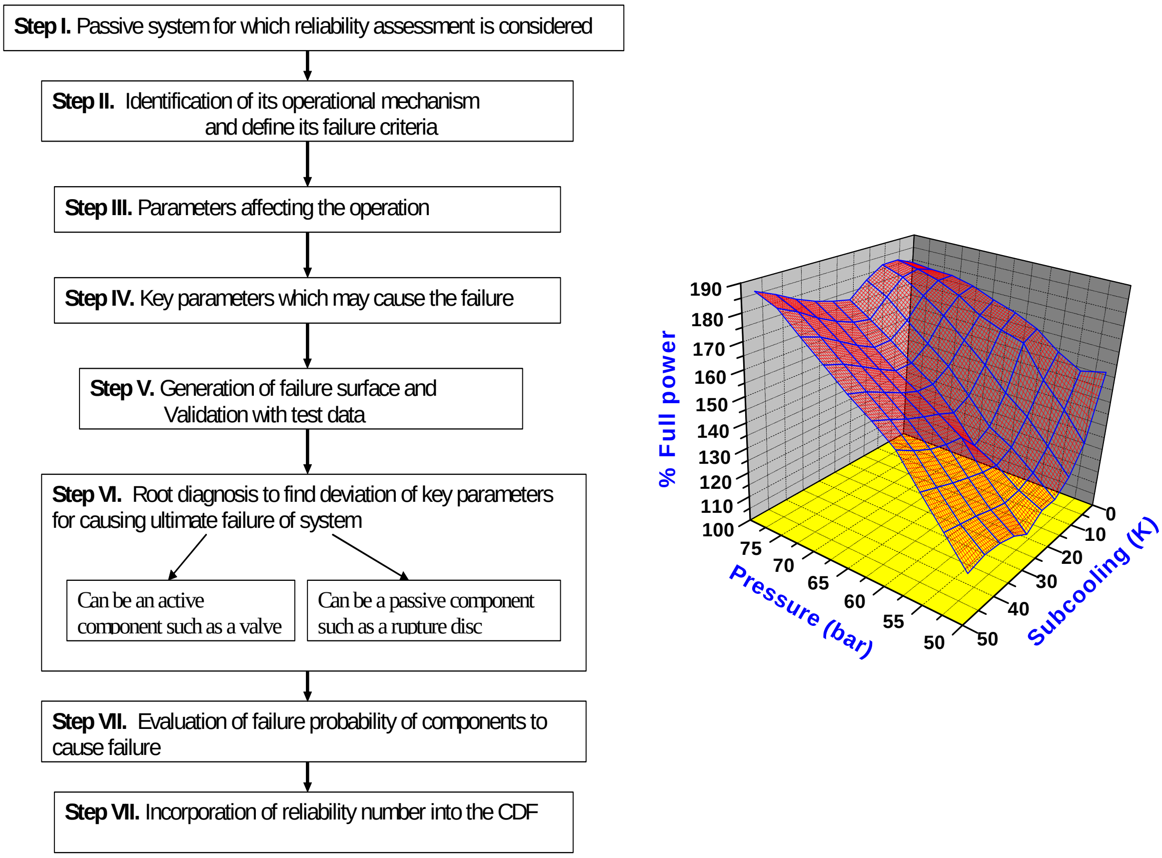

FIG .2. Flowchart of the APSRA methodology (left) and typical failure surface for natural circulation (right).

Like the RMPS methodology described above, the APSRA methodology developed in BARC, India, is primarily intended to analyze reliability of the passive systems employing natural convection. The driving head being small, a natural convection based system is susceptible to deviation from the performance of an intended function by a small change in the key parameters. Because of this, there has been a growing concern about the reliability of natural convection based systems.

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[23] Failure surface [23] is an experiment-backed predicted boundary of reliable operation of a passive safety system defined against all variables that may affect performance of such system and used to support the subsequent root cause analysis (actually, the failure surface defined in [23] is of iterative nature, also supporting the identification of those tests that are still missing).

The methodology named assessment of passive system reliability (APSRA) starts with selection of the system followed by the understanding of its operational mechanism. Using simple computer codes, the key parameters causing functional failure of the system are identified. Failure criteria are determined. Best estimate codes, such as RELAP5, etc., are then used to determine the key parameter ranges, a deviation from which may cause system failure. These ranges of parameters are then fine-tuned based on the data generated in test facilities. This is done by performing uncertainty analysis for the predictions of a best estimate code, using the in-house experimental data obtained in integral and separate effect test facilities.

In the next step, the possible causes of deviation of these parameters are revealed through root diagnosis. It is attributed that the deviation of such physical parameters occurs only due to a failure of the mechanical components, such as valves, control systems, etc. Then, the probability of failure of a system is evaluated from the failure probability of these mechanical components, through a classical PSA treatment.

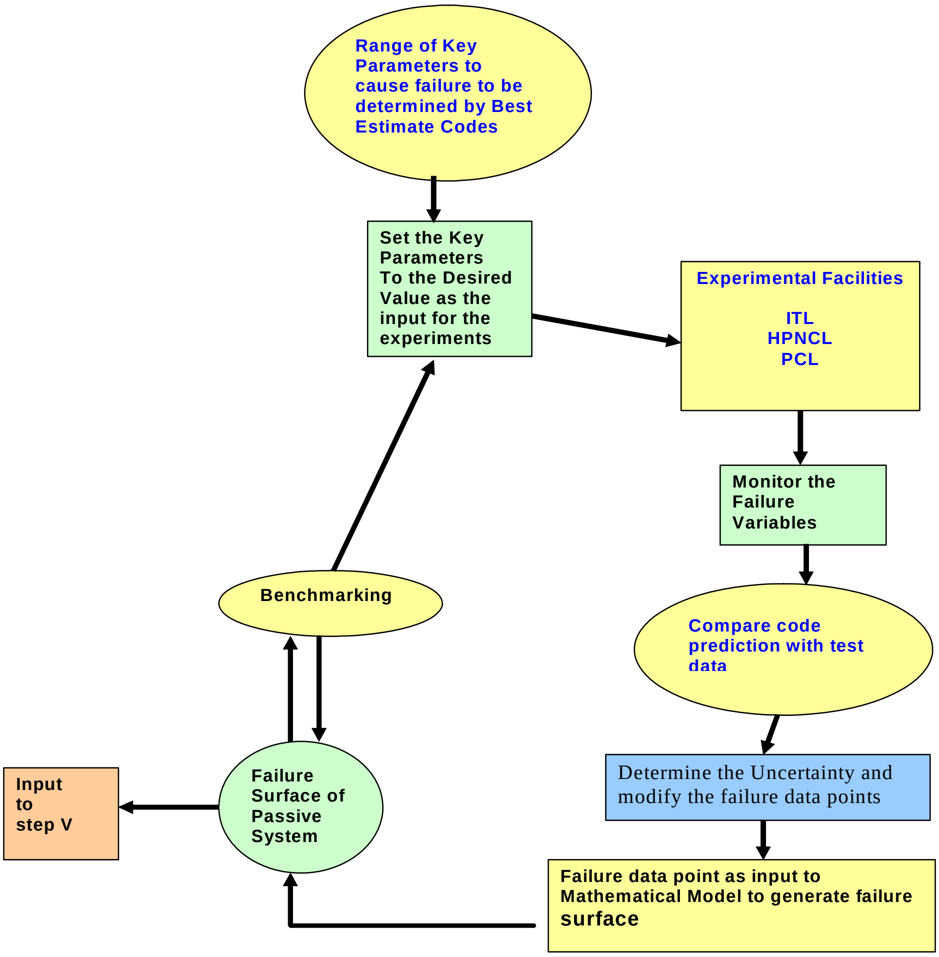

FIG. 3. APSRA methodology: flowchart of the programme for benchmarking of the failure surface based on experimental data.

To demonstrate the methodology in a test case, it has been applied to the main heat transport system of the AHWR reactor, described in ANNEX VI to this report. This system employs a boiling (two-phase) light water coolant in natural circulation. To find the code uncertainties, the code predictions were compared with the data generated from experimental natural circulation facilities, and the uncertainties were evaluated from the error distribution between the code predictions and the test data. The facilities mentioned for the generation of the required experimental data were the integral test facility ITL, the high pressure natural circulation loop HPNCL, and the flow pattern transition instability loop FPTIL [13].

The effects of variation of the key parameters on system performance were evaluated, and a multi-dimensional failure surface was generated. The probability of the system to reach the failure surface was elaborated using generic data for the failure of components.

The APSRA methodology is being applied to other passive systems of the AHWR, such as the decay heat removal system using isolation condensers, passive containment cooling system, passive containment isolation system, etc.

Common issues and recommended further R&D

The approaches presented in short in the previous section were discussed by their proponents and other experts at a dedicated IAEA technical meeting, convened on 12-16 June 2006 in Vienna (Austria) with experts from interested member states and international organizations — Argentina, Brazil, China, France, India, Italy, Japan, the Russian Federation, the USA, and the European Commission as an observer. In the conclusions to this meeting, it was noted that the APSRA and the RMPS methodologies are complementary in the following:

- APSRA incorporates an important effort on qualification of the model and use of the available experimental data. These aspects have not been studied in the RMPS, given the context of the RMPS project;
- APSRA includes in the PSA model the failure of those components which cause a deviation of the key parameters resulting in a system failure, but does not take into account the fact that the probability of success of a physical process could be different from unity;
- RMPS proposes to take into account, in the PSA model, the failure of a physical process. It is possible to treat such data, e.g., the best estimate code plus uncertainty approach is suitable for this purpose;
- In fact, two different philosophies or approaches have been used in the RMPS and in the APSRA and the two developed methodologies are, therefore, different. At the same time, the proponents of the RMPS conclude that certain parts of the APSRA and the RMPS could be merged in order to obtain a more complete methodology.

During the IAEA technical meeting mentioned above and after it several other distinct approaches for reliability assessment of passive safety system performance were noted [14 and 15], and the consensus was that a common analysis-and-test based approach would be helpful for the design and qualification of future advanced nuclear reactors. The inclusion of tests appears to be a must once new designs of passive systems and, especially, when non water cooled reactors are considered, for which validated codes and sufficient data for validation of the codes might be a priori not available. The approach itself is expected to streamline and speed up the process, and improve the quality, of validation and testing of passive safety system performance.

Reflecting on these developments in Member States, the IAEA is implementing a coordinated a research project (CRP) on "Development of Methodologies for the Assessment of Passive Safety System Performance in Advanced Reactors" in 2008-2012.

The objective is to determine a common method for reliability assessment of passive safety system performance. Such a method would facilitate application of risk-informed approaches in the design optimization and safety qualification of the future advanced reactors, contributing to their enhanced safety levels and improved economics.

In addition to what was discussed above, it will likely be necessary to confirm that over the plant lifetime passive safety systems retain capability to perform its safety function as designed. As it has already been mentioned, such confirmation would be facilitated if possible ageing effects on passive safety systems are considered in plant design and if passive safety systems are designed with a provision for easy in-service inspection, testing, and maintenance. In addition to this, new approaches might be needed to perform this confirmation, different than with active safety systems. One possible approach to deal with this issue is outlined in a short paper contributed by D.C. Wade of the Argonne National Laboratory (USA), enclosed as Appendix 1.

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APPENDIX II

PERIODIC CONFIRMATION OF PASSIVE SAFETY FEATURE EFFECTIVENESS

D. Wade (ANL, USA)

Technical Specifications that govern plant operations require that active safety systems be periodically validated and/or recalibrated as a means to assure that they continue to perform their required safety function. Passive safety features are subject to ageing phenomena over the multi-decade life of the plant, and so a means is needed to periodically reconfirm that they also remain always capable to perform their required safety function.

The means to accomplish this reconfirmation is specific to the safety function being performed and to the plant design, but the philosophy of periodic checking of passive safety features under Technical Specification requirements can be illustrated for the specific case of liquid metal cooled fast reactors that rely on a reactor vessel auxiliary cooling system (RVACS) for passive decay heat removal and thermo-structural reactivity feedbacks to self regulate power output to match externally-imposed heat removal rates.

First, for the case of the RVACS, performance degradation might occur due to partial clogging of the ambient air circulation channels with dust, rodent nests, flooding of the lower regions of the ducting, etc.; additionally, changes of the emittance properties of radiation surfaces due to oxidation or dust layers, and etc., might increase heat transport impedance. But continuous heat balances on the always-operating RVACS heat rejection rate can be performed in a completely straightforward manner by monitoring air flow rate and temperature rise versus reactor power level. The heat balance instrumentation will, of course, require periodic recalibration in its own right.

The thermo-structural reactivity feedbacks that govern power self regulation are integral feedbacks which depend on temperature profiles in the reactor; they affect reactivity directly through Doppler and density coefficients of reactivity and indirectly through structural displacements which affect neutron leakage rates. Their components change versus burn-up and age due to changing fuel composition and due to structural relaxations of core support structure, core clamping mechanisms, and creep of the fuel wrapper. The periodic reconfirmation to show that the thermo-structural feedbacks remain in the range necessary to assure passive matching of power to external heat removal rate rests on the fact that such feedbacks are composite feedbacks with respect to externally-controllable variables. These externally-controllable variables are the inlet coolant temperature, the forced circulation flow rate, and the reactivity vested in control rods. Specifically, asymptotically – after transients die away – normalized power, P, depends on these external variables via a quasi reactivity balance as:

$$\Delta\rho \equiv 0 = (P-1)\,A + \left(\frac{P}{F}-1\right)B + \delta T_{in}C + \Delta\rho_{ext}\,,$$

where F is normalized primary flow rate, and δT_{in} is change in coolant inlet temperature from its operating value.

The integral reactivity coefficients A, B, C have the following physical interpretations:

C is the reactivity vested in the deviation of core inlet temperature from its nominal value;

B is the reactivity vested in the coolant average temperature rise above the coolant inlet temperature;

A is the reactivity vested in the fuel average temperature rise above the coolant average temperature.

They are measurable in-situ on the operating power plant in a non-intrusive way by introducing step changes in flow rate, coolant inlet temperature and external (rod) reactivity and then measuring the asymptotic value of the normalized power after the transient dies away [1]. For example, three measurements might be made wherein the external variables are changed one at a time:

- If $\Delta\rho_{ext}$ is changed while inlet temperature and flow remain fixed, the power will asymptotically self-adjust to:

$$P_1 = 1 + \frac{-\Delta\rho_{ext} / B}{1 + A / B};$$

- If flow rate is changed while inlet temperature and $\Delta\rho_{ext}$ remain fixed, the power will asymptotically self-adjust to:

$$P_2 = \frac{-\left(A + B\right)}{\left(A + \frac{B}{F_2}\right)};$$

- If inlet temperature is changed, δT, while $\Delta\rho_{ext}$ and flow rate remain fixed, the power will asymptotically self-adjust to:

$$P_3 = \frac{\left(A + B\right)}{\left(A + \frac{B}{F_o}\right)} - \left(\frac{C\,\delta T_{inlet}}{A + \frac{B}{F_o}}\right).$$

This procedure would yield three equations in the three unknowns, A, B, C, — which would determine their current values on the operating reactor itself. The efficacy of such measurements in determining the values of A, B, and C on an operating reactor connected to the grid was demonstrated [2] at EBR-II.

Some small and medium sized reactors rely on natural circulation in which case flow, F, is not externally controllable but instead is a function of power F= f(P). Assuming that f(P) is could be represented as a quadratic:

$$F = a + bP + cP^2,$$

several additional step changes in $\Delta\rho_{ext}$ and/ or δT_{inlet} would be sufficient to determine the values of a, b, and c.

More elegant methods have been developed based on continuous monitoring and noise analysis techniques — taking advantage of spontaneous fluctuations or small purposeful power spectral density inputs to the externally controlled state variables.

These examples for liquid metal cooled fast reactors illustrate the approach that can be taken for periodic reconfirmation of the ability of passive safety features to perform their safety function. Other reactor types with different passive features may employ alternative approaches.

REFERENCES TO APPENDIX II

[1] D. C. WADE AND R. N. HILL, *The Design Rationale of the IFR*, Progress in Nuclear Energy, Vol. 31, No. 1/2, pp 13-42, (1997).

[2] PLANCHON, SACKETT, GOLDEN, & SEVY, *Implications of the EBR-II Inherent Safety Demonstration Test*, Nuclear Engineering and Design, 101, p.75 (1987).

APPENDIX III

TERMS USED

Small and medium sized reactors (SMRs)

According to the classification currently used by the IAEA, small reactors are the reactors with an equivalent electric power less than 300 MW, medium sized reactors are the reactors with an equivalent electric power between 300 and 700 MW [1][24].

Small reactors without on-site refuelling

According to the definition given in [1], small reactors without on-site refuelling are the reactors designed for infrequent replacement of well-contained fuel cassette(s) in a manner that prohibits clandestine diversion of nuclear fuel material.

Safety related terms

Definitions from IAEA safety standards

The format to describe passive safety design options for SMRs, provided in Appendix 3 and used in ANNEXES I – X contributed by member states, was developed reflecting the definitions used in the IAEA safety standard NS-R-1 *Safety of the Nuclear Power Plants: Design Requirements* [7]:

ACTIVE COMPONENT. A component whose functioning depends on an external input such as actuation, mechanical movement or supply of power.

PASSIVE COMPONENT. A component whose functioning does not depend on an external input such as actuation, mechanical movement or supply of power.

PLANT EQUIPMENT (see Fig. 1).

SAFETY SYSTEM. A system important to safety, provided to ensure the safe shutdown of the reactor or the residual heat removal from the core, or to limit the consequences of anticipated operational occurrences and design basis accidents.

PROTECTION SYSTEM. System which monitors the operation of a reactor and which, on sensing an abnormal condition, automatically initiates actions to prevent an unsafe or potentially unsafe condition.

PLANT STATES (see Fig. 2).

NORMAL OPERATION. Operation within specified operational limits and conditions.

POSTULATED INITIATING EVENT. An event identified during design as capable of leading to anticipated operational occurrences or accident conditions.

ANTICIPATED OPERATIONAL OCCURRENCE. An operational process deviating from normal operation which is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions.

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[24] References are given in line with the list of references for the main part of the report.

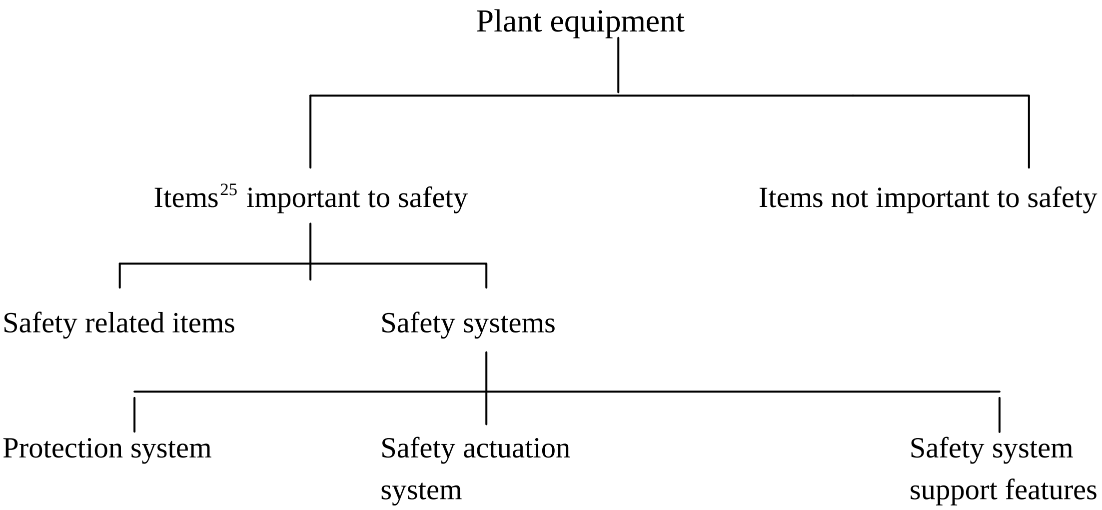
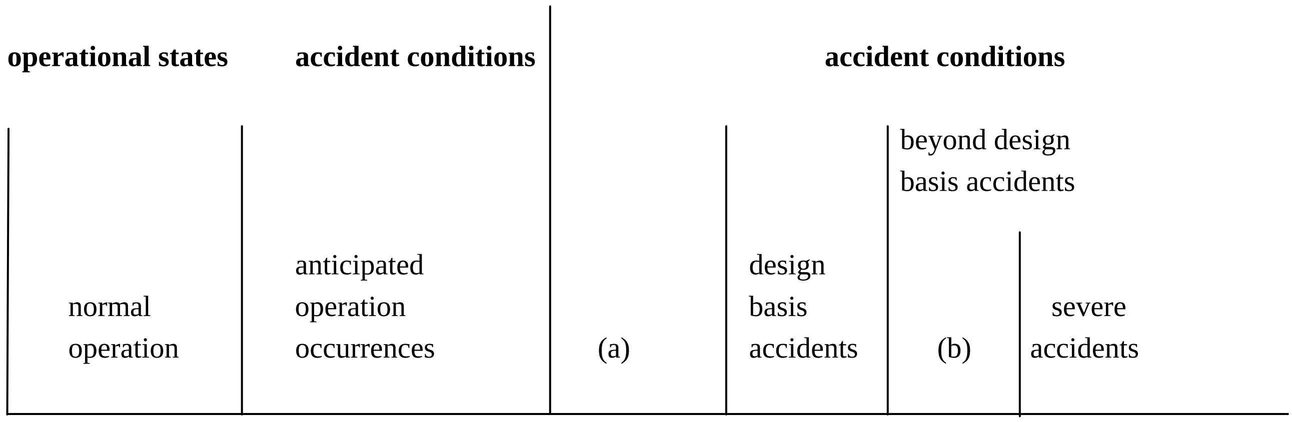


FIG. 1. Plant equipment [7].



(a) Accident conditions which are not explicitly considered design basis accidents but which they encompass;

(b) Beyond design basis accidents without significant core degradation.

FIG. 2. Plant states [7].

ACCIDENT CONDITIONS. Deviations from normal operation more severe than anticipated operational occurrences, including design basis accidents and severe accidents.

DESIGN BASIS ACCIDENT. Accident conditions against which a nuclear power plant is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

SEVERE ACCIDENTS. Accident conditions more severe than a design basis accident and involving significant core degradation.

ULTIMATE HEAT SINK. A medium to which the residual heat can always be transferred, even if all other means of removing the heat have been lost or are insufficient.

SINGLE FAILURE. A failure which results in the loss of capability of a component to perform its intended safety function(s), and any consequential failure(s) which result from it.

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[25] In this context, an ‘item’ is a structure, system or component [7].

COMMON CAUSE FAILURE. Failure of two or more structures, systems or components due to a single specific event or cause.

SAFETY FUNCTION. A specific purpose that must be accomplished for safety.

Non-consensus definitions from IAEA TECDOCs

At the moment, the IAEA safety standards do not provide a complete set of definitions necessary for the description of safety features of NPPs with innovative reactors. In view of this, some missing definitions related to passive safety features could be taken from IAEA-TECDOC-626 [12]:

INHERENT SAFETY CHARACTERISTIC. Safety achieved by the elimination of a specified hazard by means of the choice of material and design concept.

PASSIVE COMPONENT. A component, which does not need any external input to operate.

PASSIVE SYSTEM. Either a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation.

GRACE PERIOD. The grace period is the period of time during which a safety function is ensured without the necessity of personnel action in the event of an incident/accident.

Recommendations from International Nuclear Safety Advisory Group (INSAG)

Although the IAEA safety standard NS-R-1 [7] provides a consensus definition of the defence in depth levels, the definitions suggested in INSAG-10 [10] may better suit for NPPs with innovative reactors. For the future reactors, reference [11] envisages the following trends of the different levels of defence in depth:

"— Level 1, for the prevention of abnormal operation and failures is to be extended by considering in the basic design a larger set of operating conditions based on general operating experience and the results of safety studies. The aims would be to reduce the expected frequencies of initiating failures and to deal with all operating conditions, including full power, low power and all relevant shutdown conditions.

—Level 2, for the control of abnormal operation and the detection of failures, is to be reinforced (for example by more systematic use of limitation systems, independent from control systems), with feedback of operating experience, an improved human-machine interface and extended diagnostic systems. This covers instrumentation and control capabilities over the necessary ranges and the use of digital technology of proven reliability.

—Level 3, for the control of accidents within the design basis, is to consider a larger set of incident and accident conditions including, as appropriate, some conditions initiated by multiple failures, for which best estimate assumptions and data are used. Probabilistic studies and other analytical means will contribute to the definition of the incidents and accidents to be dealt with; special care needs to be given to reducing the likelihood of containment bypass sequences.

—Level 4, for the prevention of accident progression, is to consider systematically the wide range of preventive strategies for accident management and to include means to control accidents resulting in severe core damage. This will include suitable devices to protect the containment function such as the capability of the containment building to withstand hydrogen deflagration, or improved protection of the basemat for the prevention of melt-through.

— Level 5, for the mitigation of the radiological consequences of significant releases, could be reduced, owing to improvements at previous levels, and especially owing to reductions in source terms. Although less called upon, Level 5 is nonetheless to be maintained."

Terms to be avoided

The designers were not requested to adjust safety related terminology of their projects accordingly when preparing the design descriptions for this report; they had rather followed the definitions accepted in their respective member states. However, in line with the recommendations of [5] and upon the approval from designers, terms such as "revolutionary design", "passive, simplified and forgiving design", "inherently safe design", "deterministically safe design", "catastrophe free design" etc. were edited out from design descriptions, except for the cases when they appear in the names of certain reactor concepts.

Categorization of passive systems

At the moment, there is no consensus definition of a passive safety system.

In IAEA-TECDOC-626 [12], four different categories of passive safety features have been proposed, as described below.

Category A passive safety features are those, which do not require external signal inputs of "intelligence", or external power sources or forces, and have neither any moving mechanical parts nor any moving working fluid. Examples of safety features included in this category are:

- Physical barriers against the release of fission products, such as nuclear fuel cladding and pressure boundary components and systems;
- Hardened building structures for the protection of a plant against external event impacts;
- Core cooling systems relying only on heat radiation and/ or convection and conduction from nuclear fuel to outer structural parts, with the reactor in hot shutdown; and
- Static components of safety related passive systems (e.g., tubes, pressurizers, accumulators, surge tanks), as well as structural parts (e.g., supports, restraints, anchors, shields).

Category B passive safety features are those, which do not require external signal inputs of "intelligence", or external power sources or forces, and have no moving mechanical parts. They do, however, have moving working fluid. Examples of safety features included in this category are:

- Reactor shutdown/emergency cooling systems based on injection of borated water produced by the disturbance of a hydrostatic equilibrium between the pressure boundary and an external water reservoir;

- Reactor emergency cooling systems based on air or water natural circulation in heat exchangers immersed in water reservoirs (inside containment) to which the decay heat is directly transferred;
- Containment cooling systems based on natural circulation of air flowing around the containment walls, with intake and exhaust through a stack or in tubes covering the inner walls of silos of underground reactors; and
- Fluidic gates between process systems, such as "surge lines" of PWRs.

Category C passive safety features are those, which do not require external signal inputs of “intelligence”, or external power sources or forces. They do, however, have moving mechanical parts whether or not moving working fluids are present. Examples of safety features included in this category are:

- Emergency injection systems consisting of accumulators or storage tanks and discharge lines equipped with check valves;
- Overpressure protection and/ or emergency cooling devices of pressure boundary systems based on fluid release through relief valves;
- Filtered venting systems of containments activated by rupture disks; and
- Mechanical actuators, such as check valves and spring-loaded relief valves, as well as some trip mechanisms (e.g., temperature, pressure and level actuators).

Category D passive safety features, referred to as "passive execution /active initiation" type features, are those passive features where the execution of the safety function is made through passive methods as described in the previous categories except that internal intelligence is not available to initiate the process. In these cases an external signal is required to trigger the passive process. Since some desirable characteristics usually associated with passive systems (such as freedom from external sources of power, instrumentation and control and from required human actuation) are still to be ensured, additional criteria such as the following are generally imposed on the initiation process:

- Energy must only be obtained from stored sources such as batteries or compressed or elevated fluids, excluding continuously generated power such as normal AC power from continuously rotating or reciprocating machinery;
- Active components in passive systems are limited to controls, instrumentation and valves, but valves used to initiate safety system operation must be single-action relying on stored energy; and manual initiation is excluded.

Examples of safety systems, which may be included in this category, are:

- Emergency core cooling/ injection systems, based on gravity driven or compressed nitrogen driven fluid circulation, initiated by fail-safe logic actuating battery-powered electric or electro-pneumatic valves;
- Emergency core cooling systems, based on gravity-driven flow of water, activated by valves which break open on demand (if a suitable qualification process of the actuators can be identified); and
- Emergency reactor shutdown systems based on gravity driven, or static pressure driven control rods, activated by fail-safe trip logic.

Some non-conventional terms used in this report

(1) The wording ‘reactor line’ is used to denote the totality of known designs of reactors of a given type, e.g., the reactor lines considered in the present report are pressurized water reactors, pressurized light water cooled heavy water moderated reactors, high temperature gas cooled reactors, sodium cooled and lead cooled fast reactors, and non-conventional reactor designs.

(2) Several designers of SMRs addressed in this report use the wording ‘passive shutdown’ to denote bringing the reactor to a safe low-power state with balanced heat production and passive heat removal, with no failure to the barriers preventing radioactivity release to the environment; all relying on the inherent and passive safety features only, with no operator intervention, no active safety systems being involved, and no external power and water supplies being necessary, and with the grace period infinite for practical purpose.

(3) The wording ‘reactor self-control’ is used by the designers of SMRs to refer to the capability of a reactor to self-adjust the reactivity and power level in a way that prevents the progression of a abnormal operation occurrence or a design basis accident into a more severe stage, without the operation of active safety systems and operator intervention.

(4) Descriptions of the passive safety design features of SMRs, contributed by member states and given in ANNEXES I–X to this report, may occasionally include the following terms that are not accepted internationally but are in use in certain member states:

- In India they may use the term ‘incident conditions’ instead of ‘accident conditions’ defined in the NS-R-1 [7];
- In France they may use the term ‘intrinsic safety feature’ with a meaning corresponding to the ‘inherent safety feature’ used by the IAEA [7];
- In the Russian Federation, the term ‘self-protection feature’ is sometimes used to denote a capability of a reactor to bring itself in safe state in a certain unprotected transient without human intervention. It is used to denote a combination of the inherent and passive safety features and also includes passively actuated or permanently operating passive safety systems.
- Also in the Russian Federation, the term ‘self-defence principle’ is sometimes used in application to innovative reactors to define the use of the reactor inherent and passive safety features and passive safety systems to ensure the “deterministic type” of protection from more important severe accidents;
- In the USA, within I-NERI and Generation IV programmes, the term ‘passive safety’ is used in a meaning that is very close to what IAEA-TECDOC-626 defines as inherent safety characteristic. Specifically, ‘passive safety’ includes the phenomena, e.g., the core always covered with coolant, or elimination of a possibility to lose the flow of a primary system.

- The IRIS team led by Westinghouse (USA) uses the term “safety-by-design” to characterize an inherent safety where postulated accident by design: 1) are outright eliminated, or 2) have reduced probability of occurring, and/or 3) have reduced consequences.
- Regarding passive design options not related to safety, the term ‘passive load follow’ is used in the USA to denote the self-adjustment of reactor power due to reactivity feedbacks following changes of heat removal.
- In the USA, the term ‘pre-conceptual design’ is used to denote early design stage, referred to as ’feasibility study’ in [2].
- Also in the USA, the term ‘to design-out certain events’ is used to denote essential suppression or elimination of certain events by design.

APPENDIX IV

OUTLINE TO DESCRIBESAFETY DESIGN FEATURES OF SMRs

1. Reactor full and abbreviated name

2. Brief description of the design and safety design concept with a reference to previous publications

3. Description of inherent (by-design) and passive safety features, passive and active systems

- Inherent and passive safety features (Category A in IAEA-TECDOC-626)
- Passive systems (Categories B, C, D in IAEA-TECDOC-626)
- Active systems

IMPORTANT: For each passive and active system, please, indicate whether it is safety grade or back-up system

4. Role of inherent and passive safety features and passive and active systems in defence-in-depth (NS-R-1, with a reference to questionnaire Q4)

Level 1: Prevention of abnormal operation and failure

Level 2: Control of abnormal operation and detection of failure

Level 3: Control of accidents within the design basis

Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of consequences of severe accidents

Level 5: Mitigation of radiological consequences of significant release of radioactive materials

Note: Please, try to follow this IAEA-supported DID structure, even if your domestic practice the concept of DID is different

5. Acceptance criteria for design basis accidents (DBA) and beyond design basis accidents (BDBA)

- List of DBA and BDBA (NS-R-1)
- Acceptance Criteria for DBA and BDBA (deterministic and probabilistic, if applicable)
- Protection against the impacts of external events, combinations of events considered in the design (NS-G-3.3, and NS-G-1.5)
- Probability of unacceptable radioactivity release beyond the plant boundaries
- Measures planned in response to severe accidents

6. Questionnaires

Q1. List of safety design features considered for/incorporated into a SMR design

#	SAFETY DESIGN FEATURES	WHAT IS TARGETED?

Q2. List of internal hazards

#	HAZARDS THAT ARE OF SPECIFIC CONCERN FOR A REACTOR LINE	EXPLAIN HOW THESE HAZARDS ARE ADDRESSED IN A SMR

Q3. List of initiating events for safety analysis

#	(1) LIST OF INITIATING EVENTS FOR SAFETY ANALYSIS (BOTH TYPICAL FOR THIS REACTOR LINE AND CHARACTERISTIC OF THIS INDICVIDUAL DESIGN) MARK INITIATING EVENTS THAT ARE SPECIFIC TO THIS PARTICULAR SMR	SPECIFY DESIGN FEATURES OF A SMR USED TO PREVENT PROGRESSION OF INITIATING EVENTS TO AOO /DBA / BDBA, USED TO CONTROL DBA, USED TO MITIGATE BDBA CONSEQUENCES, ETC.

Q4. Safety design features attributed to defence in depth levels

#	SAFETY DESIGN FEATURE	(1) INDICATE AOO / DBA / BDBA OF RELEVANCE (2) INDICATE CATEGORY: A-D* (FOR PASSIVE SYSTEMS ONLY)	RELEVANT DID LEVEL ACCORDING TO NS-R-1**

* Categories A-D correspond to IAEA-TECDOC-626

** An outline of approaches to DID for advanced NPPs is provided in INSAG-10 and IAEA-TECDOC-1434

Q5. Positive/ negative effects of passive safety design features in areas other that safety (if any)

PASSIVE SAFETY DESIGN FEATURES	POSITIVE EFFECTS IN ECONOMICS, ETC.	NEGATIVE EFFECTS IN ECONOMICS, ETC.

CONTRIBUTORS TO DRAFTING AND REVIEW

Delmastro, Dario	Comisión Nacional de Energía Atómica, Centro Atómico Bariloche Argentina
Carelli, Mario	Westinghouse Science and Technology USA
Petrovic, Bojan	Westinghouse Science and Technology USA
Mycoff, Clark	Westinghouse Science and Technology USA
Gautier, Guy-Marie	DER/SESI/LESA CEA Cadarache, France
Delpech, Marc	CEA-Saclay-DEN-DDIN, France
Naviglio, Antonio	The University of Rome “La Sapienza”, Italy
Cumo, Maurizio	The University of Rome “La Sapienza”, Italy
Nishimura, Satoshi	Central Research Institute of Electric Power Industry (CRIEPI) Japan
Nayak, A.K.	Reactor Engineering Division, Thermal Hydraulics Section, Bhabha Atomic Research Centre India
Devictor, Nicolas	CEA/DEN/DER/SESI/LCFR France
Saha, Dilip	Reactor Engineering Division, Bhabha Atomic Research Centre India
Dulera, I.V.	Reactor Engineering Division, Bhabha Atomic Research Centre (BARC), Trombay, Mumbai, India
Shepelev, Sergey	Experimental Design Bureau of Machine Building (OKBM), Russian Federation
Lepekhin, A.N.	Experimental Design Bureau of Machine Building (OKBM), Russian Federation

Sienicki, James J.	Innovative Systems Development, Nuclear Engineering Division, Argonne National Laboratory (ANL), USA
Wade, David C.	Nuclear Engineering Division, Argonne National Laboratory (ANL) USA
Minato, Akio	Nuclear Energy Strategy Office, Central Research Institute of Electric Power Industry (CRIEPI), Japan
Sinha, R.K.	Reactor Engineering Division, Bhabha Atomic Research Centre (BARC), Trombay, Mumbai, India
Kuznetsov, Vladimir	International Atomic Energy Agency, Vienna, Austria