Sodium-cooled Fast Reactor (SFR) Risk and Safety Assessment White Paper

R. Nakai and D. Hahn

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SODIUM-COOLED FAST REACTOR (SFR)
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Abstract

This paper, prepared jointly by Generation IV International Forum (GIF) Risk and Safety Working Group (RSWG) and the SFR Systems Steering Committees (SSCs), presents the status of the Integrated Safety Assessment Methodology (ISAM) for Sodium-cooled Fast Reactor (SFR) for further research and development. The intent is to demonstrate the adequacy of safety related design; to identify R&D needs to qualify safety related provisions, and to recommend directions for optimizing its risk and safety performance. The three main objectives of this work, identified during the first workshop which brought together SSC representatives and RSWG members, are: capturing in the short-term salient features of the design concepts that impact their safety performance; identifying crosscutting studies that assess safety design or operating features common to various GIF systems; and suggesting beneficial characteristics of the design of future nuclear power plants that should be addressed in subsequent GIF activities.

1. Introduction

In the Generation IV International Forum (GIF), risk mitigation and enhanced safety are one of goals for the Generation IV (Gen-IV) system development and serve in developing the basis to assess the performance of Gen-IV systems. A set of analytical tools have been developed for evaluating Gen-IV systems to ensure that safety is “built-in” rather than “added-on”. The Integrated Safety Assessment Methodology (ISAM) provides a comprehensive framework and guidance for carrying out a system evaluation using five distinct analytical tools. However, when undertaking a specific case study, difficulties arise from either the lack of specific information in the early stages of design or the proprietary nature of detailed information for mature designs.

This report, prepared jointly by the Risk and Safety Working Group (RSWG) and the SFR Systems Steering Committees (SSCs), presents the status of ISAM methodology for SFR for further research and development. The intent is to demonstrate the adequacy of safety related design, to identify R&D needs to qualify safety related provisions, and to recommend directions for optimizing its risk and safety performance. The three main objectives of this work, identified during the first workshop which brought together SSC representatives and RSWG members, are: capturing in the short-term salient features of the design concepts that impact their safety performance; identifying crosscutting studies that assess safety design or operating features common to various GIF systems; and suggesting beneficial characteristics of the design of future nuclear power plants that should be addressed in subsequent GIF activities.

These three objectives are further described below:

1. White papers have been developed by the SSCs to identify qualitative design features that describe safety provisions, key R&D inputs that confirm the adequacy of the design, demonstrate the structure of the safety-related provisions on the basis of defense-in-depth (DiD) philosophy, develop success criteria, and provide quantitative assessment of risks and the consistency of safety architecture with success criteria.
2. Results and recommendations of generic interest on safety features may be derived from the application of the ISAM methodology to a set of GIF systems. Specific recommendations that may be derived from such generic studies are expected to yield valuable guidelines for proceeding with further detailed conceptual studies of GIF systems.
3. Desirable features for the global architecture of future nuclear plants to minimize risks and optimize safety constitute another type of generic study that may lead to recommendations on the design of reactor coolant boundaries, control systems, layout of nuclear service

buildings, and on implementing dedicated instrumentation to achieve an effective monitoring and surveillance of safety functions required for accident prevention and mitigation.

An action plan that builds on the current ISAM evaluation methodology and its application to the case study of an example sodium fast reactor, and that makes use of the three types of above-suggested studies, is being currently followed in GIF. This leads the RSWG to interact with SSCs in order to conduct generic studies on the GIF systems, and to oversee evaluations made by the SSCs of their specific system for the sake of consistency across all GIF systems. This approach leaves to the SSCs the ultimate responsibility of the final assessment of safety features of their respective designs, with input from the RSWG subject matter experts.

2. Short overview of the ISAM assessment methodology

The RSWG has developed a methodology, called the Integrated Safety Assessment Methodology (ISAM), for use throughout the Gen-IV technology development cycle. The ISAM consists of five distinct analytical tools as follows (Ref. 1) which are intended to support achievement of safety that is “built-in” rather than “added on” by influencing the direction of the concept and design development:

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

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

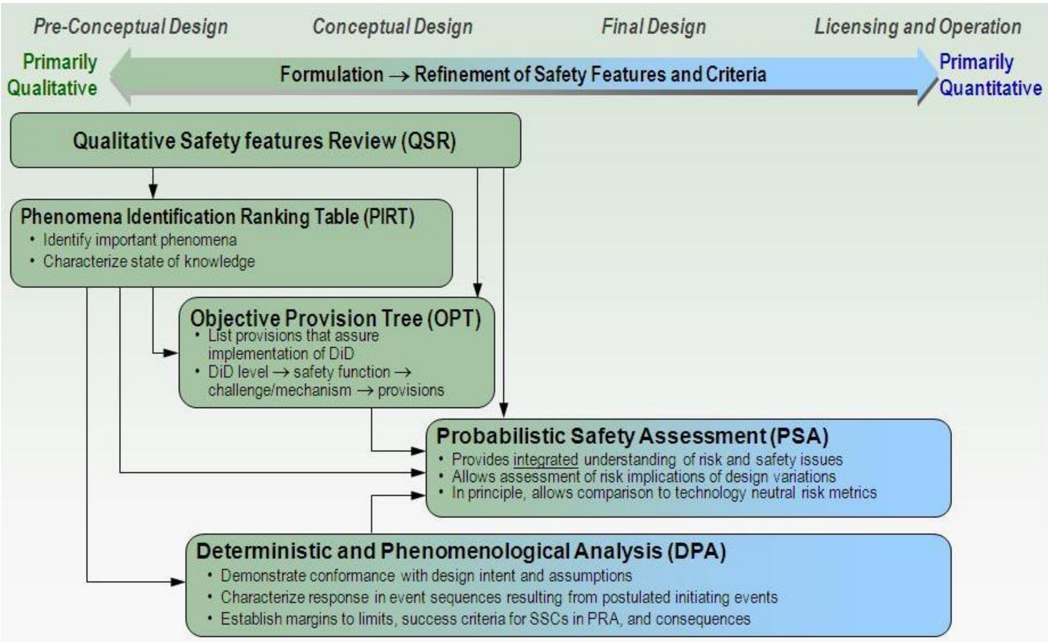

Figure 1. Proposed GIF Integrated Safety assessment Methodology (ISAM) Task Flow

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

- *Qualitative Safety Features Review (QSR)*

The Qualitative Safety Features Review (QSR) is a new tool that provides a systematic means of ensuring and documenting that the evolving Gen-IV system concept of design incorporates the desirable safety-related attributes and characteristics that are identified and discussed in the RSWG’s first report entitled, “Basis for the Safety Approach for Design and Assessment of

Generation IV Nuclear Systems” (Ref. 2), as well as in other references (e.g. the INPRO Safety methodology). Although this element of the ISAM is offered as an optional step, it is believed that the QSR provides a useful means of shaping designers’ approaches to their work to help ensure that safety truly is “built-in, not added-on” since the early phases of the design of Gen-IV systems. Using a structured template to guide the process, concept and design developers are prompted to consider, for their respective systems, how the attributes of “defence in depth”, high safety reliability, minimization of sensitivity to human error, and other important safety characteristics might best be incorporated. The QSR also serves as a useful preparatory step for other elements of the ISAM by promoting a richer understanding of the developing design in terms of safety issues or vulnerabilities that will be analyzed in more depth in those other analytical steps.

- *Phenomena Identification and Ranking Table (PIRT)*

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

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

- *Objective Provision Tree (OPT)*

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

- *Deterministic and Phenomenological Analyses (DPA)*

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

- *Probabilistic Safety Analysis (PSA)*

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

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

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

3. Overview of Technology

A basic description of the Sodium-Cooled Fast Reactor (SFR) system is given in the Annex of the GIF SFR Systems Arrangement (Ref. 3), and the four current design “tracks” are described in the GIF SFR System Research Plan (Ref. 4). This section will provide an overview of key SFR technology features.

The SFR system was identified during the Generation IV Technology Roadmap (Ref. 5) as a promising technology to perform the actinide management mission and, if enhanced economics for the system could be realized, also the electricity and heat production missions. The main characteristics of the SFR that make it especially suitable for the actinide management mission are:

- Consumption of transuranics in a closed fuel cycle, thus reducing the radiotoxicity and heat load which facilitates waste disposal and geologic isolation.
- Enhanced utilization of uranium resources through efficient management of fissile materials and multi-recycle.

The SFR system uses liquid sodium as the reactor coolant, allowing high power density with low coolant volume fraction. While the oxygen-free environment prevents corrosion, sodium reacts chemically with air and water and requires a sealed coolant system. The primary system operates at near-atmospheric pressure with typical outlet temperatures of 500-550°C; at these conditions, austenitic and ferritic steel structural materials can be utilized, and a large margin to coolant boiling is maintained. High level of safety achieved through inherent and passive means that accommodate transients and bounding events with significant safety margins. The reactor unit can be arranged in a pool layout, a compact loop layout, or a hybrid of these two arrangements. Plant sizes ranging from small modular systems to large monolithic reactors are being considered. A wide variety of fuels and fuel cycles are being considered.

There are many predecessor SFR conceptual designs that have been developed worldwide in national advanced reactor development programs. In particular, the European Fast Reactor in EU (Ref. 6, 7, 8, 9), the Advanced Liquid Metal Reactor (PRISM) and Integral Fast Reactor Programs in USA (Ref. 10, 11), and the Demonstration Fast Breeder Reactor in Japan (Ref. 9, 12, 13) have been the basis for many SFR design studies. For the Gen-IV collaboration, several new design concepts have been contributed by the participants to guide the R&D research activities. These designs cover a wide range of reactor size and configuration options. Within the following subsections, the four contributed reactor “tracks” are briefly illustrated and described.

3.1 Loop Configuration SFR

To promote favorable economies of scale, many SFR designs have targeted large monolithic plant designs. For this approach, a prominent recent concept is the Japan Sodium Fast Reactor (JSFR) (Ref. 14, 15, 16, 17, 18) which is a sodium-cooled, MOX (or metal) fueled, advanced loop-type evolved from Japanese fast reactor technologies; the conceptual plant layout is shown in Figure 2.

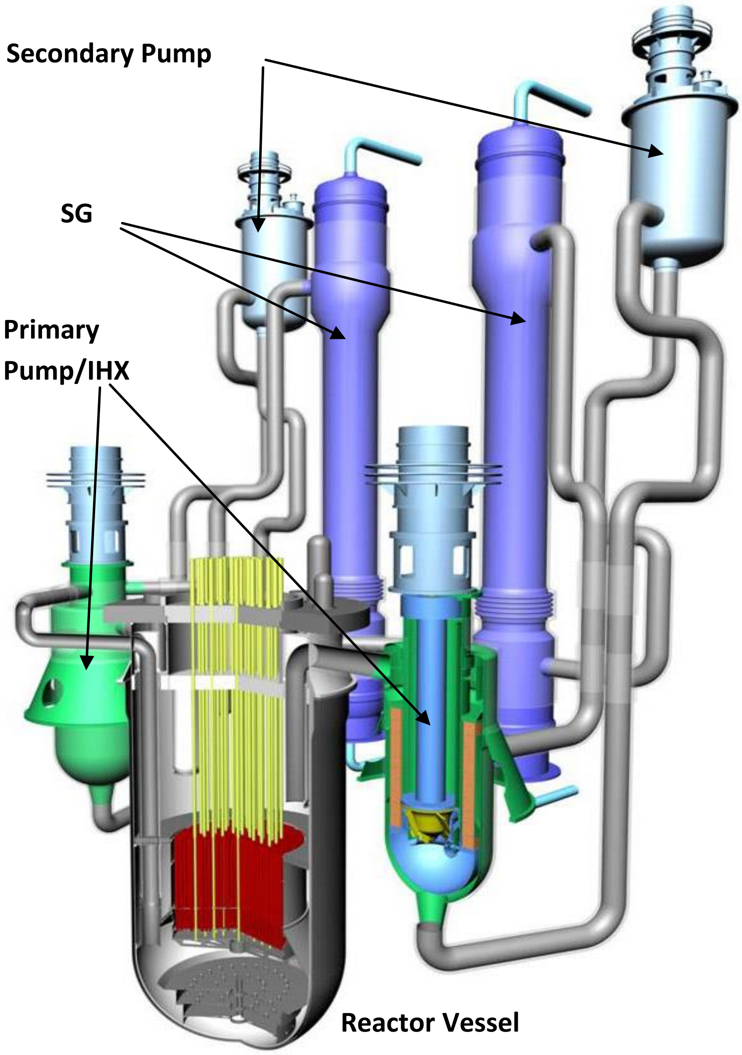

Figure 2. Japan Sodium-cooled Fast Reactor (JSFR)

The JSFR design employs several advanced technologies to reduce the construction cost: compact design of reactor structure, shortened piping layout, reduction of loop number, integration of components, and simplification of decay heat removal system through enhancement of natural circulation capability. These measures include innovative technologies such as 12Cr-steel with high strength, an advanced structural design standard at elevated temperature, three-dimensional seismic isolation, and re-criticality free core.

Despite the desirable features of a SFR such as low pressure with excellent heat transfer characteristics, the chemical reactions of sodium with air and water, and sodium's opaque characteristics make the in-service inspection and repair (ISI&R) more difficult. The JSFR design utilizes passive safety measures to increase its reliability. The improvement of ISI&R technology is concentrated to confirm the integrity of internal structures, including core support structure, and coolant boundaries. The means of access is taken into account in design.

The JSFR design studies consider plant sizes ranging from a modular system composed of medium size reactors to a large monolithic reactor. The large-scale sodium-cooled reactor utilizes the advantage of "economy of scale" by setting the electricity output to 1500MWe. On the other hand, a medium-scale modular reactor would offer advantages of flexibility in power requirements from utility companies and the reduction of development risk compared with large-scale reactors.

3.2 Pool Configuration SFR

Moderate size SFR designs have also been proposed; in this case, cost reduction relies on design simplification and factory fabrication techniques. A recent example is the KALIMER-600 pool-type reactor design, shown in Figure 3 (Ref. 19, 20), evolved from previous pool-type SFR designs such as PRISM and EFR. A pool-type reactor provides many important design advantages in plant economy and safety. The entire Primary Heat Transport System (PHTS) piping and equipment is located inside the vessel completely eliminating the possibility of PHTS piping break outside the reactor vessel. Also the large thermal inertia characteristics of a pool-type reactor enhance passive safety mechanisms and allow an increased grace period during accidents. The safety of KALIMER is enhanced further by loading its core with metal fuel which has inherent safety characteristics resulting from large negative power reactivity coefficients and a very low probability of a Core Disruptive Accident (CDA).

For improvement of the plant economy over previous designs, KALIMER reduces the number and/or eliminates equipment by design simplification and novelty, compact design and higher plant efficiency. The decay heat removal system consists of Passive Decay heat Removal Circuit (PDRC) and Active Decay heat Removal Circuit (ADRC). Even in the active component failures of ADRC, the flow in the decay heat removal system is driven by natural circulation which maintains the primary system temperature below allowable limit.

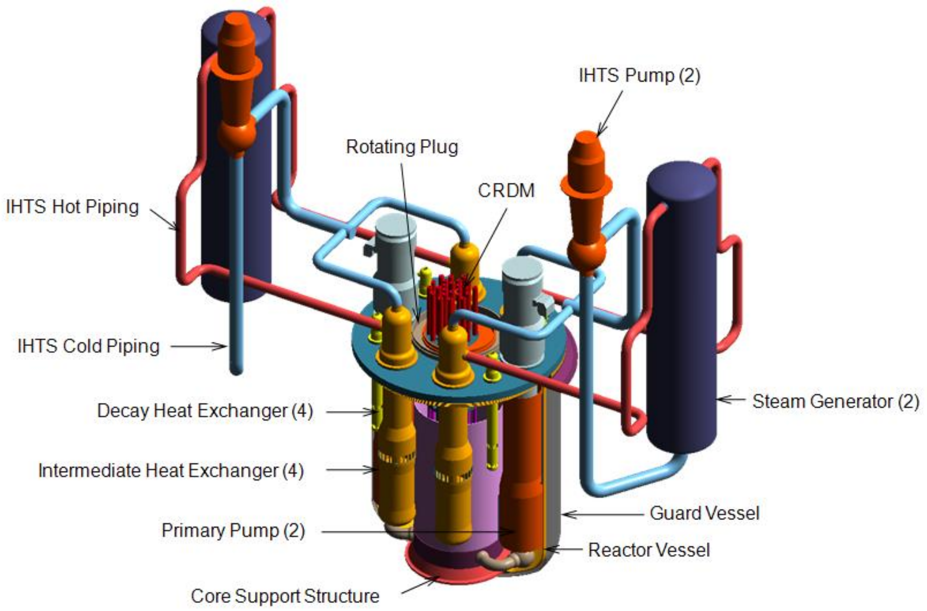

Figure 3. KALIMER-600 System Configuration

A large pool SFR is proposed for economy of scale and benefiting from generic characteristics of the pool concept including design simplification and compactness. Such a large pool SFR industrial Sodium Fast Reactor of 1500 MWe is being studied within the Collaborative Project on European Sodium Fast Reactor (CP-ESFR) of the 7th Euratom Framework program of the European commission (Ref. 21).

The design is optimized for the oxide fuel and aims at a flexible breeding and minor actinide burning strategy. The oxide ESFR core is composed of two enrichment zones of Inner/Outer Core Fuel Assemblies, and 3 rows of reflectors (1 row can be used for fertile or Minor Actinide fuel assemblies). There are two independent control rod systems composed of 24 control and shutdown device assemblies and 9 diverse shutdown device assemblies. The fuel management scheme is based on a fuel residence time of 2050 equivalent full power days with five batches. The maximum fuel burn-up is 155 GWd/tHM for an average power density of 206 W/cm^3.

The ESFR primary system is sketched in Figure 4. It is based on options already considered in previous and existing pool SFRs, with several potential improvements regarding safety, inspection and manufacturing. The reactor vessel is cooled with sodium (submerged weir) and is surrounded by a hanged safety vessel. Some provisions have been made for internal and external core catchers. The reactor vault can be inspected for maintenance. The strongback is simply lying on the vessel bottom to remove weld spots that are difficult to inspect. The pump-strongback connection is very short and robust compared to solutions used in the past. The inner vessel, with a conical part, is welded directly on the strongback to facilitate manufacturing.

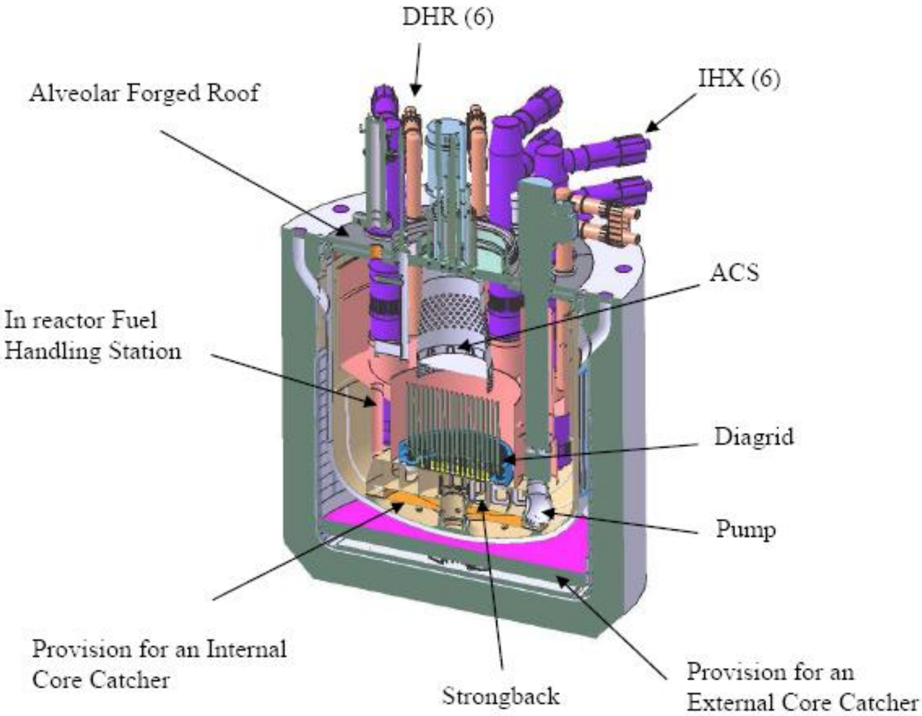

Figure 4. 3D view of ESFR primary system

An alveolar forged roof structure potentially enhances feasibility, reduces height and weight, and allows cooling. It operates at a temperature of 120°C to avoid sodium aerosol freezing. The components and fuel handling systems are resting on the roof. An ACS (Above Core Structure) has the function to guide control rods, support instrumentation, and calm sodium above the core. Other large components include 6 IHX (Intermediate Heat Exchangers) made out of stainless steel and 3 Primary Pumps.

Decay heat removal function is provided by the Direct Reactor Cooling (DRC) System which comprises six sodium loops. All loops extract heat from the primary sodium of the hot pool by means of immersed sodium/sodium dip coolers (DHX) removing each 50% of total residual power, and reject the heat to the environment using sodium/air heat exchangers situated on the periphery of the reactor building roof within air stacks. Diversity (operational and structural) and redundancy of the 6 DRC loops is ensured by 3 natural convection and 3 forced convection loops (i.e. with pumps in sodium and air to increase efficiency of the exchangers and with different component designs).

The secondary system comprises six 600 MWth parallel and independent sodium loops, each connected to an IHX. Each secondary loop hosts 6 modular Steam Generators of 100 MWth each made out of modified 9Cr1Mo (ASME grade 91). Modularity is aimed at reducing the impact of a Sodium/Water reaction and at improving overall capacity factor.

One of the most structuring options of the Nuclear Island layout is the twinning of two reactors with a shared Fuel Handling Building and Component Maintenance Building to reduce investment cost.

3.3 Small Modular SFR

The Small Modular Fast Reactors (SMFR) aim at exploiting characteristics inherent to fast reactors in application to locations with small grid capacity. In a recent United States study (Ref. 22), a reactor size of 50 MWe was selected for a specific niche market where industrial infrastructure is not sufficient for larger systems and the unit cost of electricity generation is very high with conventional technologies. Examples of this situation are remote areas in Alaska, small grid systems in developing countries, and Pacific-basin islands. The basic goal is to make the operation, safety, and fuel management as simple as possible; for example, by the application of a long-lived reactor core that eliminates the need for refueling. The SFR characteristics that enable this approach are:

- The non-corrosive character of sodium coolant does not degrade the reactor core material and primary system components even over very long residence times.
- The excellent neutron economy of fast spectrum and metal fuel can be exploited to design a small core with a conversion ratio near unity, obviating the need for refueling to compensate for reactivity losses over an extended lifetime.

Innovative design features have been incorporated into the SMFR design including a metallic fueled core with high internal conversion ratio, inherent passive safety characteristics, simplified reactor configuration for modular construction and transportability, and supercritical CO_2 Brayton cycle power conversion system. The primary and intermediate heat transport systems and Brayton power conversion are depicted in Figure 5; the primary and intermediate systems are embedded below ground level for more robust physical protection. The primary system is configured as a typical pool arrangement with the core, pumps, intermediate heat exchangers, and auxiliary cooling decay heat exchangers all contained within the reactor vessel. The intermediate sodium exits the vessel and flows to the sodium-to-CO_2 heat exchangers.

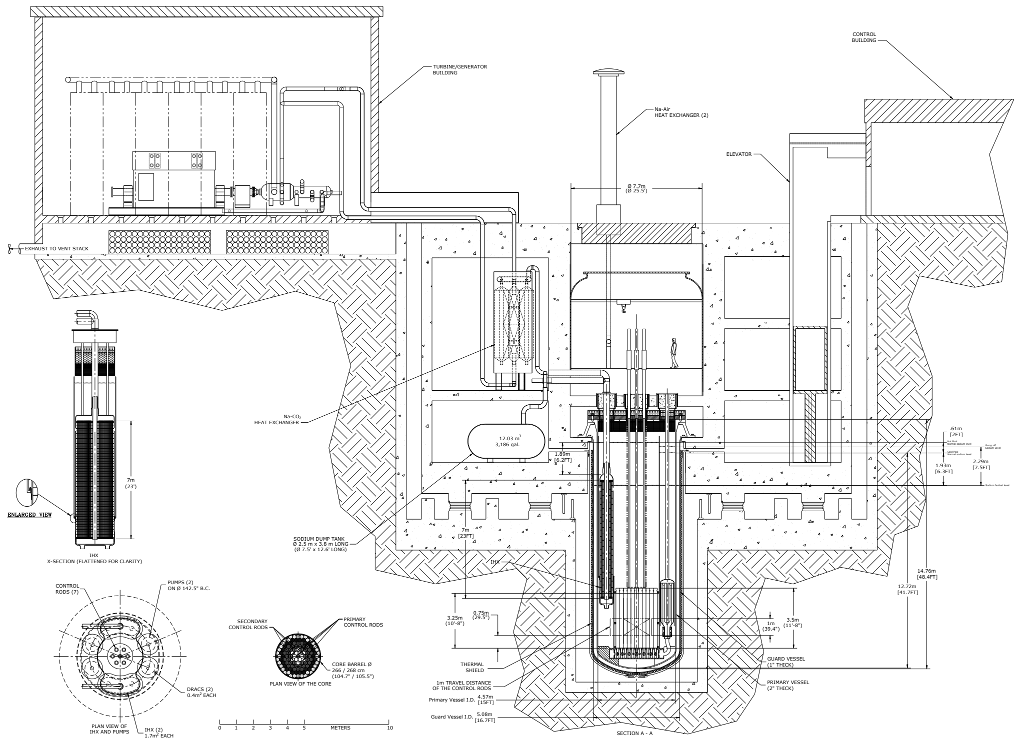

Figure 5. Elevation View of SMFR System

A key design feature of the SMFR is the long-lived core – 30 years with no refueling. This long lifetime improves proliferation resistance by eliminating all aspects of on-site fuel management: new fuel acceptance, spent fuel handling, and out-of-reactor storage. The SMFR incorporates all the passive safety features developed for SFR applications to avoid plant damage; this includes a passive decay heat removal system directly from the primary coolant pool.

The SMFR utilizes a metal fuel form with similar burn-up and fluence limits as employed for the KALIMER design. However, the SMFR operates at a significantly reduced power density to achieve the 30 year lifetime design goal. Thus, the system size is increased compared to a conventional SFR high power density design, and this results in a higher system cost per unit power generation. However, the SMFR energy generation cost is acceptable for the intended niche market application where the small size and design simplicity are more important considerations.

3.4 Summary of Generation-IV SFR Tracks

Table 1 summarizes the key design parameters of the SFR design concepts identified in the previous three subsections. It is important to note that all of these SFR systems are designed with a large degree of flexibility in size, specific fuel design, and fuel loading configuration. These particular designs are indicative of current international SFR design studies that cover a wide range of power applications (sized from 50-1500 MWe). The question of size involves a cost reduction approach by economies of scale for large systems as compared to modular factory fabrication for small systems. Other factors like capital investment limits or electrical grid limitations may dictate the optimal deployment system power rating.

Table 1. Key Design Parameters of Generation IV SFR Concepts

Design Parameters	JSFR	KALIMER	SMFR	ESFR
Power Rating, MWe	1 500	600	50	1 500
Thermal Power, MWt	3 570	1 500	125	3 600
Plant Efficiency, %	42	40	~38	~42
Core outlet coolant temperature, ^{o}C	550	545	~510	545
Core inlet coolant temperature, ^{o}C	395	390	~355	395
Main steam temperature, ^{o}C	503	503	480	490
Main steam Pressure, MPa	16.7	16.5	20	18.5
Cycle length, years	1.5–2.2	1.1	30	1.25
Fuel reload batch, batches	4	5	1	5
Core Diameter, m	5.1	4.2	1.75	4.9
Core Height, m	1.0	0.89	1.0	1.0
Fuel Type	MOX(TRU bearing)	Metal(U-TRU-10%Zr Alloy),	Metal(U-TRU-10%Zr Alloy),	MOX(TRU bearing)
Cladding Material	ODS	HT9M	HT9	ODS
Pu enrichment (Pu/HM), %	13.8	25.2	15.0	15.2
Burn-up, GWd/t	150	139	~87	155 (max)
Breeding ratio	1.0–1.2	0.74	1.0	1.0-1.2

With regard to the fuel and loading, any of the systems can be designed for different actinide management missions. The reactor performance noted in Table 1 is for converter mode designs; each concept could readily be modified to breeder or transmuter configurations by changing the fuel assembly design to modify the uranium loading. Furthermore, the SFR reactor performance can be achieved with different fuel forms, depending on the success of the advanced fuels research to develop and demonstrate recycle fuels.

4. Overview of the Safety Architecture's characteristics and performances

SFR, is fast neutron sodium-cooled reactor, which has less neutrons absorption and moderation characteristics than conventional light water reactors (LWRs), and makes it possible to effectively produce nuclear fuel or burn Pu and minor actinides while generating electric power. Hence, SFR is one of the most promising concepts for next-generation nuclear systems.

Good heat-transport characteristics of sodium can allow us to design a compact, high-performance and low-pressure reactor system. There is a relatively large thermal inertia of the primary coolant, and a large margin to coolant boiling is achieved by design. A decay heat removal (DHR) by natural circulation is possible by ensuring the cooling circuit with optimum layout to ultimate heat sink. This enables simpler and highly reliable DHR systems with no dependence on support systems such as electric systems and component cooling systems. Another major safety feature is that the primary system operates near the atmospheric pressure, pressurized only to the extent needed to circulate fluid. Therefore it is easy to maintain coolant inventory for core cooling because of no rapid flushing out of primary coolant even upon the primary boundary failure.

On the other hand sodium reacts chemically with air and water, and thus the design must limit the potential for such reactions and their consequences. A secondary sodium system acts as a buffer between the radioactive sodium in the primary system and the energy conversion system. The opaqueness of sodium and the prevention of local blockage in fuel subassembly should be considered in the design.

Since an SFR core is not in its most reactive configuration, recriticality due to a core compaction is a possibility. Therefore, a hypothetical core disruptive accident (HCDA) is an important consideration for commercialization of SFRs. It should be robustly demonstrated that the consequences of an HCDA can be managed with various design options of the core and the reactor. Toward commercialization of SFR technology, it is essential to practically eliminate the occurrence of re-criticality resulting in a significant mechanical energy release.

The first three levels of the DiD emphasize prevention, detection and control of accidents, which collectively ensure that the plant remains within so-called "design basis." Accident prevention is the first priority, because provisions to prevent deviations from normal operation state are generally more effective and more predictable than measures aimed at mitigation of the consequences of abnormal conditions. With a primary emphasis on preventing and detecting abnormal occurrences, safety design provisions shall be provided for control of postulated abnormal conditions. These provisions include the appropriate means to shut down the reactor core and remove the residual heat from the reactor core. HCDAs shall be excluded from design basis accident (DBA) by implementation of these safety provisions within the first three levels of DiD. Ensuring the independence of different levels of protection is a key element to avoid the propagation of failure into subsequent levels. This might be accomplished by use of passive systems and reliance on diverse systems for reducing common cause failures.

For the purpose of meeting the ambitious GIF safety and reliability goals, we also need to strengthen the safety design of the fourth level of DiD by considering preventive and mitigative measures against design extension conditions (DECs). A set of DECs shall be derived on the basis of engineering judgement, operationg experiences, deterministic assessments, and probabilistic assessments for further improving the safety of the nuclear power plant by enhancing the plant's capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than DBAs or involve additional failures. In these situations, it is essential to provide design measures to prevent accident progression and to mitigate its consequences. An HCDA that could lead to a significant radioactive release shall be practically eliminated by appropriate provisions. Safety demonstrations of practically eliminated situations shall be robust and based on deterministic and/or probabilistic analyses that address uncertainties and covers a large spectrum of events.

Although the final goal is the identification of the provisions which, for each safety function, participate in the control, management, and mitigation (if needed) of all the plausible abnormal situations through the application of ISAM, the preliminary example application of ISAM in the SFR systems are presented to facilitate the designer's further application in the following sections.

4.1 Qualitative Safety Features Review (QSR)

A comprehensive set of qualitative recommendations / criteria will be supplied to the designer by the QSR. These recommendations / criteria are structured following the principle of the DiD. Each of these recommendations / criteria are detailed as far as feasible (i.e. step by step), first with a "technology neutral" logic (i.e. applicable to all the technologies) and, further, looking for technology specific guidelines. Following this logic, a comprehensive "check list" is established for the different safety functions and, once defined, for the different concept technologies. The designer will so be asked to go through the check list in order to qualitatively assess the options under examination identifying strong and weak characteristics.

The system and option characteristics and features are compared to the check list's items. The options' characteristics can be rated as **"favourable" (⇧)**, **"unfavourable" (⇩)** or **neutral (⇔)** to satisfy or meet each specific recommendation of the check list. For a given option, "favourable" rating will be used to support its implementation while the identification of "unfavourable" rate will either be used to discard its selection or to motivate further R&D effort to reduce the identified drawbacks.

4.2 Phenomena Identification and Ranking Technique (PIRT)

As applied to Gen-IV nuclear systems, the PIRT is an expert elicitation process used to identify:

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

It is recognized that PIRT could be helpful in demonstrating adequacy of analysis models and parameters that are used in DPA, which is necessary for defining success criteria of level-1 PSA. In particular, PIRT could be useful in identifying and considering important factors or phenomena affecting the safe shutdown or decay heat removal upon accidents. JAEA performed preliminary application of PIRT to examine the reactor safe shutdown by means of Self Actuated Shutdown System (SASS), during an unprotected loss of flow (ULOF) accident, where the term of "unprotected" means "with a failure of conventional reactor shutdown system". SASS is installed above the core and it holds the control rods above the core in normal operation. Once the core outlet temperature rises abnormally, SASS loses the holding force, when the temperature becomes above the Curie point, without actuation of any instrumentation and control devices and the control rods are inserted into the reactor core by gravity force.

Following the individual step described in Chapter 2 of ISAM report, PIRT of reactor shutdown by SASS upon the ULOF accident was conducted as shown below (Ref. 1).

1) The issue was defined as identifying the priority R&D issues related to innovative safety features specific to JSFR.
2) The specific objectives were defined as identifying phenomena and factors having a significant impact on reactor safe shutdown by means of SASS, in order to confirm effectiveness of SASS, which is expected to be actuated under Design Extension Conditions.
3) Database information was obtained: e.g., R&D results concerning the holding force that is dependent upon the temperature of the main device constituting SASS, design specifications of SASS, design information about neutronics and thermal/hydraulics characteristics of reactor and PHTS.

4) Hardware and scenario were defined as follows:
   i. Hardware: SASS in the backup reactor shutdown system (BRSS), reactor, reactor power control system (RPCS) and PHTS.
   ii. Postulated scenario: a ULOF (unprotected loss of flow) accident.
5) The figure of merit was defined as the maximum temperature of core coolant, which represents the safety criterion of preventing severe core damage under the ULOF accident condition.
6) Phenomena to be considered were identified i.e. phenomena, characteristics and state variables that affect maximum temperature of core coolant upon ULOF accident, by experts who are familiar with accident and thermal/hydraulics analyses and SASS.
7) The importance ranking was rated by considering sensitivities of uncertainty included in each phenomenon in terms of the maximum core coolant temperature upon the ULOF accident. The ranking scales defined in Table 1 of ISAM report were applied.
8) The knowledge level was assessed by considering whether we have sufficient knowledge to simulate precisely the identified individual phenomena and state variables. The knowledge level ranking scales defined in Table 2 of ISAM report were applied.

The results showed the high priority-ranked R&D issues are SASS performance, such as uncertainty and/or accuracy of actuation temperature and time constraint of temperature response from the coolant to the SASS device. The innovative feature of new technology is identified as the key element to enhance the safety of JSFR. The application of PIRT to the other families of accidents could be done in a similar manner.

4.3 Objective Provision Tree (OPT)

The Objective Provision Tree (OPT) is a practical tool which should be applied to design and/or to assess the safety architecture of innovative plants in line with the defence in depth philosophy. This is achieved through visual presentation and systematically inventorying the plant's safety capabilities; i.e., the systematic identification of the provisions that contribute to the safety mission. Full range of the knowledge of the installation's characteristics, the phenomenology associated with the abnormal situations, and the associated risks are not always required in the use of the OPT.

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

The OPT method is a top-down approach with a tree structure which:

- for each level of DiD (normally level 1 to 4),
- and for each safety objective/function (in general, control of reactivity, removal of heat from the core, and confinement of radioactive materials),

Identify:

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

All this is done by expressing this hierarchy structure in a tree form.

The availability of the OPT can greatly help and simplify the preparation of the PSA.

Following the individual step described in the Chapter 2 of the ISAM report, the OPT of JSFR safety features were developed as shown below (Ref. 1).

1) The objectives were defined as assessing the structure of safety architecture of the JSFR in a systematic and comprehensive manner consistent with the defence-in-depth philosophy.
2) The design, research and safety assessment documentation were collected.
3) OPTs were developed by considering the three fundamental safety functions and the levels 1 to 4 of the defence in depth.
4) The developed OPT was illustrated in a tree structure and also expressed in a different representation with unique numbering as shown in Appendix 7 of the ISAM report.

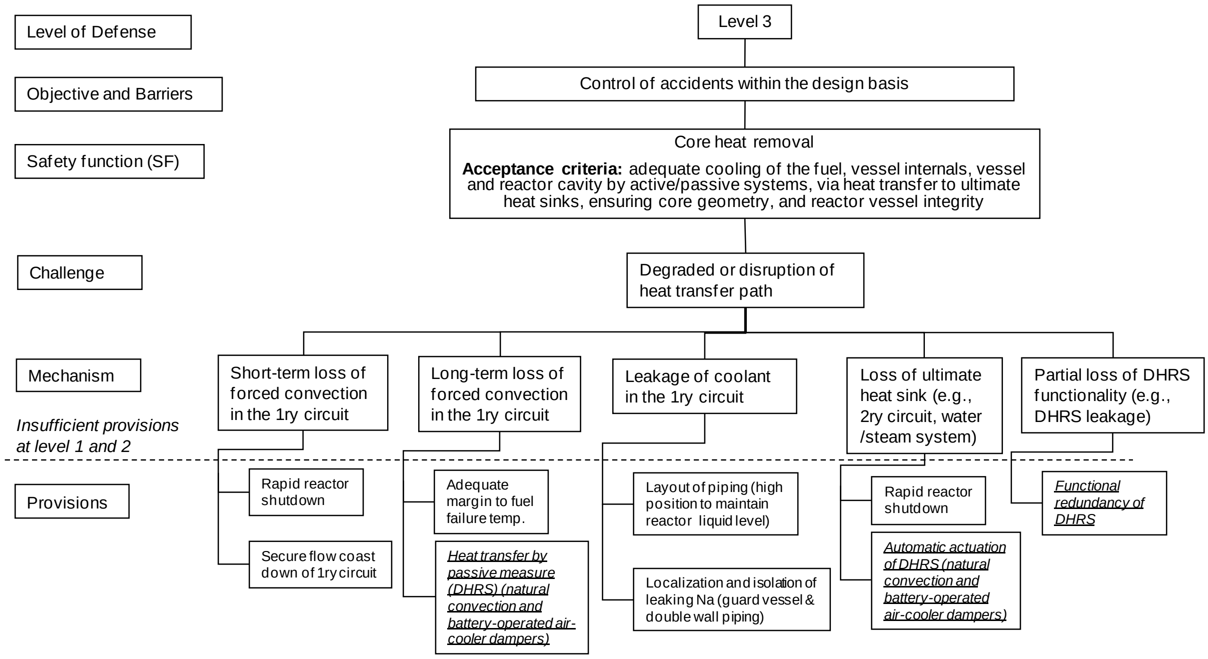

Figure 6. Example of OPT developed for JSFR safety function 2 (core heat removal) at level 3 of defence in depth

As shown in Figure 6, OPT is the organized structure of safety-related provisions based on the defence in depth philosophy. In particular, the provisions that are expressed with italic letters and underlined in Figure 6 are characterized with the safety design features and recommendations of decay heat removal function as shown below.

- Decay heat removal system (DHRS) should have capability of long-term decay heat removal without forced circulation of the PHTS sodium coolant.
- When loss of ultimate heat sink such as the secondary/intermediate heat transport system (SHTS), main feedwater system, main steam system, etc., occurs during normal reactor operation, the ultimate heat sink should be switched to DHRS automatically following the reactor scram.
- Even assuming complete loss of function in a single train of DHRS, DHRS should have sufficient decay heat removal capability.
- In the process of developing the PSA model, these features became key inputs to specify plant responses upon the initiating event and success criteria of DHRS during the decay heat removal operation. In addition, adequacy in meeting the requirements was confirmed by conducting DPA associated with the decay heat removal function.

The JSFR is equipped with three trains of reactor auxiliary cooling systems for decay heat removal so that the decay heat can be removed only by way of the decay heat removal system. One of them is the direct reactor auxiliary cooling system (DRACS) that is directly connected to the reactor vessel, and the others are the primary reactor auxiliary cooling system (PRACS) that is connected to the primary cooling system. These trains are operated in a fully passive condition (i.e. natural circulation of sodium coolant and natural air flow at the heat sink).

4.4 Deterministic and Phenomenological Analyses (DPA)

Classical deterministic and phenomenological analyses, including thermal-hydraulic analyses, CFD analyses, reactor physics analyses, accident simulation, materials behaviour models, structural analysis models, and other similar analysis tools collectively constitute a vital part of the overall Gen-IV ISAM.

These traditional deterministic analyses are used as needed to understand a wide range of safety issues that guide concept and design development, and to form input into the PSA. These analyses typically involve the use of familiar deterministic safety analysis codes.

It is anticipated that DPA will be used from the late portion of the pre-conceptual design phase through ultimate licensing and regulation of the Generation IV system.

Following the individual step described in Section 3.2 of the ISAM report, DPA of JSFR DHRS was conducted as shown below (Ref. 1).

1) Facility, objectives and scope of the analysis were specified:
   i. Target system is the DHRS in JSFR.
   ii. Objective is to determine the consequences in terms of “success or failure” of different event sequences modeled in the PSA.
   iii. Scope was specified to the analysis of decay heat removal characteristics upon the typical accident with reactor scram.
2) Approach was selected: i.e. best estimate analyses using best estimate code to cope with innovative safety features (i.e. natural circulation).
3) A computer code in the category “(c) thermo-hydraulic codes” was selected: i.e. one-dimensional flow network code that has been developed and used for Japanese SFRs.
4) Methodology of the accident analysis is as follows:
   i. Physical model to be applied is a one-dimensional flow network model.
   ii. Examples of initial and boundary conditions are initial transient power history in a short time, no heat loss from the system, conservative inlet air temperature at the air cooler, no heat exchange at the SG tubes.
   iii. Acceptance criteria are defined as maintaining core coolable geometry: i.e.
      – Coolant boundary temperature: $\leq$ 650 ºC (tentatively)
      – Core coolant temperature: $\leq$ 900 ºC (tentatively)
5) Data for analysis were collected, which is associated with the plant systems operating characteristics (e.g., heat balance), neutronics, thermal and hydraulics characteristics (e.g., reactivity coefficients, material properties), design specifications of systems and components (e.g., geometry).
6) A database containing the above data was developed and has been updated corresponding to progress of the design work.
7) The engineering handbook was developed, which describes how to convert the data included in the above database into input of the analysis code.
8) The plant model was developed.
9) The models and methods in the analysis code were applied, which are equivalent to those of the code already verified and validated that was used in the safety evaluation of the prototype sodium-cooled fast reactor “Monju”.
10) As a basic scenario the reactor scram followed by the DHRS operation was supposed. Systems and components available were determined, corresponding to the accident

sequence that was developed in the event trees of the level-1 PSA of JSFR by considering the key information obtained from the OPT. No modification of the plant model was needed as the check of the result in step 12.

11) The calculation was executed by following the code manual.

12) The analysis results were checked using the supervisory review by performing some sensitivity analyses.

13) The analysis results were presented as shown in Figure 7.

Figure 7 indicates that a successful accident sequence developed in the event trees results in the reactor coolant boundary integrity representing core integrity being maintained. In addition, the accident consequence of the other sequences was regarded conservatively as core damage by considering uncertainty. Thus, DPA serves as a determination of success criteria in the level-1 PSA model. It is for future work to implement sensitivity analyses to establish margins to the limits and to cover imprecision in actual parameters at the design stage.

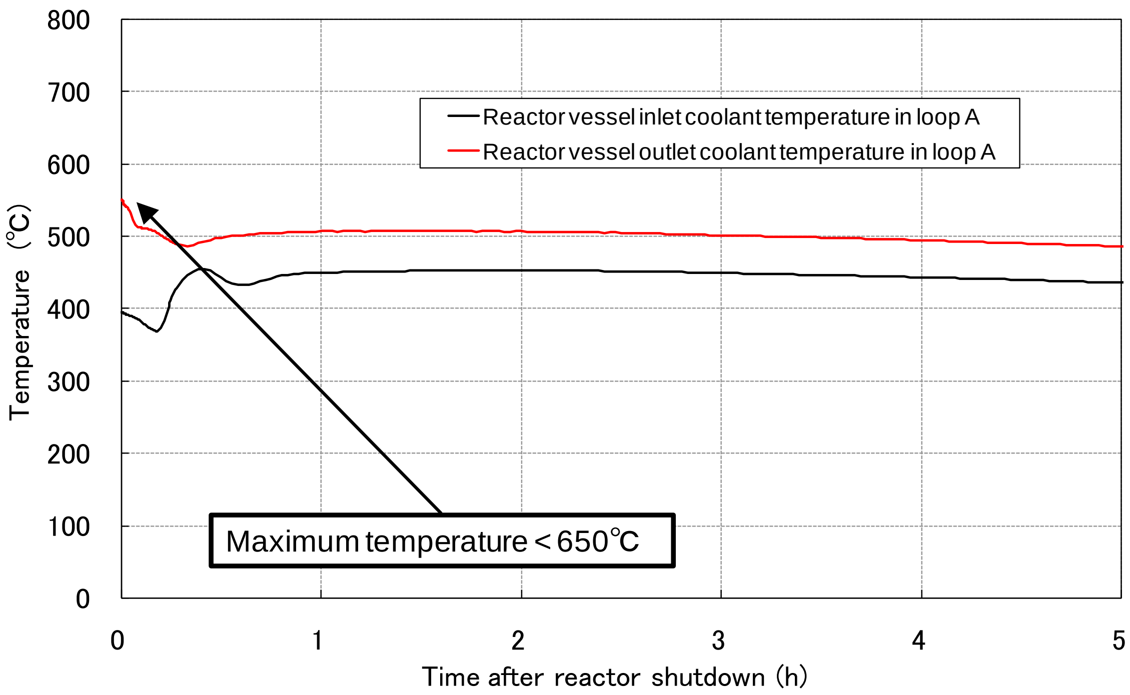

Figure 7. Example of DPA results: Passive cooling scenario by using DRACS & PRACS-A with a single air cooler damper failure

4.5 Probabilistic Safety Assessment (PSA)

The PSA is recognized as an effective means to identify accident scenarios that could occur for a given design and, with the associated assessment tools, as effective means to quantitatively assess the weight of the uncertainties associated with various aspects of those scenarios.

The complementary deterministic and phenomenological analyses confirm that design provisions forming each line of protection can adequately perform their expected functions and determine the “success criteria” for the performance of the system, modeled in the PSA together with a clear definition of the consequences for event sequence scenarios included in the PSA.

Aside the verification of the probabilistic safety/risk criteria, such as the frequency of the undesired events (e.g., core damage frequency), the PSA and the associated tools are also used to assess the interaction of design features that may be proposed to provide defence in depth in response to the identified accident scenarios.

Finally PSA could contribute to assess the whole consistency of the safety architecture versus criteria such as

- An exhaustive defence, i.e.: the identification of the risks, which leans on the fundamental safety functions, should look for exhaustiveness.
- A graduated and progressive defence; in order to avoid “short sequences” such that the failure of a particular provision entails a major increase without any possibility of restoring safe conditions at an intermediate stage.
- A tolerant defence: no small deviation of the physical parameters outside the expected ranges can lead to severe consequences (i.e., rejection of “cliff edge effects”).
- A forgiving defence, which guarantee the availability of a sufficient grace period and the possibility of recover during accidental situations.
- A balanced or homogeneous defence, i.e.: no sequence participates in an excessive and unbalanced manner to the global frequency of the damaged plant states.

Obviously the elaboration of a full scope PSA is possible only when the full design will be available. Nevertheless it is considered that with the support of the OPT data complemented with indication about the characteristics of the implemented provisions, both in term of physical performances as well as in terms of reliability, it will be possible to implement a “simplified PSA on-line assessment” which will be essential to help the design’s phases.

The scope of JSFR PSA was focused on the level-1 PSA related to internal initiators and specific to decay heat removal after successful reactor shutdown. PSA for the DHRS in JSFR was conducted following the steps below (Ref. 1).

1) Initiating events were identified and categorized, based on the plant design information and using master logic diagram method.
2) The mitigation systems were defined and the event trees (ET) were developed, based on the plant design specifications linked with the key information that was obtained from the OPTs and on the DPA results.
3) The fault trees (FT) were developed based on the system design information with some assumptions related to support systems.
4) Common cause failures (CCF) of major active failure modes of redundant components were considered e.g. damper failure to open, battery failure to supply electricity to damper drivers.
5) Human error in operator’s recovery action was considered.
6) The component failure rate was estimated, based on the reliability database for sodium-fluid components and on the domestic LWR reliability data.
7) The occurrence frequency of the initiating events was quantified, based on the failure rate and the operating experiences of nuclear reactor systems (i.e., SFRs, LWRs).
8) CCF parameters and human error probability were determined, based on the methodology used in LWR PSA.
9) Quantification of the accident sequences with combining ET and FT was executed.

Contributors to the protected loss of heat sink (PLOHS) frequency were broken down by time phases with different success criteria. The dominant contributor is loss of two out of three trains of DHRS within 24h after reactor shutdown. Obviously this is because the success criterion is different after the 24h period. If the designer enhances the heat removal capacity of a single train of DHRS in this time period so as to become less-demanding success criteria, there is potential to reduce at most 99% of the total PLOHS frequency. Based on this information, the designer and analyst examined the possibility of introducing non-safety-related blowers at the air cooler inlet to enhance PRACS and DRACS capability with consideration of both lower cost increase and significant safety improvement. By conducting additional DPA, it was confirmed that the consequence of the decay heat removal scenario with sodium natural circulation and forced-air flow by using DRACS alone would be adequate for maintaining the reactor coolant boundary integrity. In addition, the PSA with design improvement showed quantitatively that introduction of the air cooler blowers in both PRACS and DRACS can reduce significantly the PLOHS frequency; i.e. improve the reliability of decay heat removal.

There are some uncertainty issues in the PSA. In order to address the issue that cumulative component operating time is still short, compared with the target reliability level, and further effort will be made to collect the empirical reliability data for SFRs. In order to minimize the uncertainty due to shortage of such empirical reliability data, the margin to the safety target is ensured by introducing redundancy and diversity in the core cooling measures.

The second issue is related to epistemic uncertainty due to the fact that the component to be considered is a new type even if empirical reliability data of a similar type are available. Reliability and safety performance of new type of components would be tested and demonstrated to some extent in the research and development process of those components, although the operating time would be limited. Sensitivity of the uncertainty in the reliability of new components needs to be examined.

Phenomenological uncertainty associated with the passive cooling is not assessed explicitly yet. Sensitivity of the uncertainty in DPA will be analyzed and if the sensitivity is significant, the uncertainty will be quantified (e.g., with the Monte Carlo calculation based on the evaluation of the response surface and uncertainty in individual analysis parameters).

4.6 Summary of preliminary application and future work

Following the method described in Chapter 2 of the ISAM report, applicability of PIRT, OPT, DPA and PSA to various SFR systems was examined to demonstrate the adequacy of safety-related design and to guide the future R&D activities.

- PIRT provides a framework to confirm the appropriateness for key R&D items.
- OPT helps organize a structure of safety-related provisions based on the defence-in-depth philosophy.
- DPA provides key information for the success criteria to be defined in the PSA model.
- PSA provides the quantitative assessment of the level of safety and provide useful information for the system design improvement.

The Fukushima Daiichi nuclear power plants accidents were caused by an extreme tsunami that followed a very large offshore earthquake, which resulted in long-term station blackout and loss of ultimate heat sink. The plant design for Gen-IV nuclear systems shall be robust against such extreme external events, although the site-specific conditions for external hazards are unavailable in the early design stage. The assessment methodology for external events beyond design basis needs to be provided with a more balanced use of deterministic and probabilistic approaches (e.g., Stress test, external event PSA, etc.).The SFR systems will incorporate passive safety feature in the design in order to achieve high reliability. For a passive safety system, the statistical assessment methodology seems to be more applicable taking into account the phenomenological uncertainty. In particular, the best estimate approach is recommended for DECs.

5 Current System Development Status

The SFR has the highest technical maturity level among Gen-IV systems. Its development approach builds on technologies already developed and demonstrated for SFRs and associated fuel cycles in fast reactor programs worldwide; test SFRs have successfully been built and operated in Japan, France, Germany, the United Kingdom, Russia, and the United States. A major benefit of previous investments in SFR technology is that the majority of the R&D needs are related to performance rather than viability of the system. Accordingly, the Gen-IV collaborative R&D focuses on a variety of design innovations for actinide management, improved SFR economics, development of recycle fuels, in-service inspection and repair, and verification of favorable safety and operational performance.

The GIF System Research Plan covers the needs of the viability R&D phase and the performance R&D phase for the SFR system envisioned in the GIF Technology Roadmap (Ref. 5). The viability phase has extended to assess promising new technology features such as supercritical CO_2 energy conversion. The performance phase aims at the design inclusion and refinement of key SFR innovative design features by the end of 2015. These research activities have been arranged by the SFR Signatories into five “Projects” to organize the joint GIF research activities:

1) **System Integration and Assessment**: This project will carry out the design and safety studies needed to define technical requirements for safety, fuels, and components of the SFR system. The results of the technical R&D projects will be integrated into generalized design concepts (contributed by the Members), and evaluated against Generation-IV goals and criteria.
2) **Safety and Operation**: This project includes the verification of safety tools, evaluation of the effectiveness of inherent mechanisms and design features, and identification of bounding events to consider in SFR licensing and containment design. This project also includes reactor operation and technology testing campaigns in existing SFR reactors.
3) **Advanced Fuels**: This project includes the development of high burn-up fuel systems (fuel form and cladding) to complete the SFR fuel database; research on remote fuel fabrication techniques for recycle fuels that contain minor actinides and possibly trace fission products; and the consideration of alternate fast reactor fuel forms for special applications (e.g., high temperature).
4) **Component design and BOP**: This project includes the development of advanced energy conversion systems to improve thermal efficiency and reduce secondary system capital costs. It also includes the development of advanced in-service inspection and repair (in sodium) technologies.
5) **Global Actinide Cycle International Demonstration (GACID)**: This project will demonstrate that the SFR can effectively manage all actinide elements in the fuel cycle, including uranium, plutonium, and the minor actinides (neptunium, americium and curium). This technical demonstration will be pursued using existing fast reactors in a comparatively short time frame.

In addition to the Gen-IV SFR research and development projects identified above, several of the GIF member countries have plans to build prototype or demonstration SFR systems in the 2020-2030 time frame. These prototype designs typically reduce risk and system cost by employing more conventional technology options with reduced power output (for monolithic approach) or single module application (for modular approach). Thus, these modern systems provide unique opportunities to test the SFR technology innovations, and initial demonstration of the Generation-IV SFR performance potential.

6 Conclusions: System’s Issues, Concerns and Benefits

With regard to reactor safety of SFR, technology gaps center around two general areas: assurance of inherent/passive safety response, and techniques for prevention and mitigation of severe accidents. The advanced SFR designs exploit inherent/passive safety measures to increase reliability. The system behaviour varies depending on system size, design features, and fuel type. The R&D for inherent safety investigate phenomena such as axial fuel expansion and radial core expansion, and passive design features such as self-actuated shutdown systems and passive decay heat removal systems. The ability to measure and verify these inherent and passive features must be demonstrated. Supplementary R&D may also be required for some designs to evaluate severe accidents and investigate the fundamental phenomena to mitigate their consequences.

The favorable passive safety behavior of fast reactors is expected to virtually exclude the probability of severe accidents with potential for core damage. Nevertheless, design measures to mitigate the consequences of severe accidents may also be considered in level 4 of DiD. A safety approach considering the impact of the physical and chemical characteristics of the SFR materials (chemical activity sodium coolant, for example) on systems, structures, and components important to safety should be established. The goal is to render the risk for deployment of SFR systems much lower than the risk of other alternatives. Achieving this level of safety should lead to licensing and regulatory simplifications, that may in turn result in reduced system cost. To achieve this, probabilistic safety evaluations will be needed to ensure design tradeoffs that yield very high levels of safety to avoid core damage.

The concept of DiD shall be applied to the safety design of advanced SFRs. A safety level can be further improved especially enhancing prevention features with more emphasis on passive safety features. Through prevention, detection, and control of accident CDA shall be excluded from DBAs. Toward a commercialization of SFR technology, mitigation of core damage consequences will also be needed for some systems. In particular, the safety approach for practical elimination of re-criticality will be crucial for establishing public acceptance of the SFRs. The demonstration of safety approach should be achieved by deterministically with the supplemental use of probabilistic method in initial design stage. A more use of probabilistic assessment could be expected in the detailed design stage.

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