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# Regulatory Perspectives on Safety Aspects Related to Advanced Sodium Fast Reactors

## Part 2. Neutronics and Criticality Safety of Sodium Fast Reactors

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# *List of abbreviations and acronyms*

| | |
|---|---|
| ACRR | Annular core research reactor |
| ADAMS | Agency-wide Documents Access and Management System (NRC) |
| AEC | Atomic Energy Commission (United States) |
| ALWR | Advanced light water reactor |
| ANL | Argonne National Laboratory (United States) |
| AOO | Anticipated operational occurrences |
| ARDC | Advanced reactor design criteria |
| ATWS | Anticipated transient without scram |
| BDBA | Beyond-design-basis accidents |
| CAFE | Core alloy flow and erosion |
| CEFR | China Experimental Fast Reactor |
| CFD | Computational fluid dynamics |
| CFR | Code of Federal Regulations (United States) |
| CNRA | Committee on Nuclear Regulatory Activities (NEA) |
| CNSC | Canadian Nuclear Safety Commission |
| CR | Control rod |
| CRBR | Clinch River breeder reactor (United States) |
| CRPPH | Committee on Radiological Protection and Public Health (NEA) |
| CSNI | Committee on the Safety of Nuclear Installations (NEA) |
| DBA | Design-basis accidents |
| DEC | Design extension conditions |
| DiD | Defence-in-depth |
| DHR | Decay heat removal |
| ECCS | Emergency core cooling system |
| EMDAP | Evaluation model development and assessment process |
| ENEA | Italian National Agency for New Technologies, Energy and Sustainable Economic Development |

| | |
|---|---|
| EVTM | Ex-vessel transfer machine |
| FA | Fuel assembly |
| FFTF | Fast flux test facility |
| FP | Fuel pin |
| FSAR | Final safety analysis report |
| GCR | Gas cooled reactor |
| GDC | General design criteria |
| GIF | Generation IV International Forum |
| GRS | Generation random sampled |
| GSAR | Ad hoc Group on the Safety of Advanced Reactors (NEA: Joint CNRA/CSNI) |
| IAEA | International Atomic Energy Agency |
| IRSN | Institut de Radioprotection et de Sûreté Nucléaire (France) |
| KAERI | Korean Atomic Energy Research Institute |
| KASOLA | Karlsruhe Sodium Laboratory (Germany) |
| KINS | Korea Institute of Nuclear Safety |
| KIT | Karlsruhe Institute for Technology (Germany) |
| LIMTECH | Liquid metal technologies |
| LOCA | Loss-of-coolant accident |
| LWR | Light water reactor |
| MCCI | Melt coolability and concrete interaction |
| MOX | Mixed oxide fuel |
| NEA | Nuclear Energy Agency |
| NO | Normal operation |
| NRC | Nuclear Regulatory Commission (United States) |
| NUREG | NRC technical report designation |
| OECD | Organisation for Economic Co-operation and Development |
| PDSR | Package design safety report |
| PGSFR | Prototype Gen-IV sodium-cooled fast reactor |
| PWR | Pressurised water reactor |
| RCS | Reactivity control system |
| RD | Regulatory document |
| RIA | Reactivity induced accident |

| | |
|---|---|
| RSK | Reactor safety commission |
| RWMC | Radioactive Waste Management Committee (NEA) |
| SAR | Safety analysis report |
| SBO | Station blackout |
| SFR | Sodium fast reactor |
| SRP | Standard review plan |
| TMC | Total Monte-Carlo |
| TREAT | Transient Reactor Test Facility (United States) |
| TRU | Transuranium |
| ULOF | Unprotected loss of flow |
| ULOHS | Unprotected loss of heat sink |
| ULOSSP | Unprotected loss of station supply power |
| US DOE | United States Department of Energy |
| UTOP | Unprotected transient overpower |
| WGSAR | Working Group on the Safety of Advanced Reactors (NEA) |

# *Executive summary*

This report describes the regulatory perspectives on safety aspects related to the calculation and justification of neutronic characteristics and criticality safety assessment of advanced sodium fast reactors (SFR). It identifies topics that should be investigated within the framework of SFR safety regulation, potentially involving additional research and development needs.

The present report is based on the answers to the questionnaire developed by the NEA Working Group on the Safety of Advanced Reactors (WGSAR) in 2016. Eight countries contributed to this report by answering the questionnaire, thanks to their experience in licensing or their ongoing work on future SFR projects.

The following technical topics for neutronics and criticality safety were addressed: general questions, regulatory requirements and guidance related to neutronics calculations for SFRs, requirements and guidance related to criticality safety calculations for SFRs, and phenomena to be taken into account in SFR analyses. Based on a comparison of the information provided by the member countries in response to the survey, six common positions were identified. These common positions are presented in Chapter 2 of this report and represent the common approaches to the high-level safety goals and objectives, as well as safety functions to be applied for Generation IV SFRs. The essence of these common positions is mainly the following:

- The existing regulatory framework is general enough to be applicable to SFRs. Many existing neutron characteristic requirements are of sufficiently high level to be applicable universally with some specificities to be taken into account.
- All participants agreed on the parameters that should be met by SFRs.
- All participants agreed on the need to increase the use of inherent and passive safety with regard to the core design of advanced reactors including SFRs.
- The requirements for establishing and justifying neutron parameter safety limits for advanced SFRs are similar to conventional power reactor requirements.
- Accounting and evaluation of calculation parameter uncertainties are important parts of the safety analysis, and the uncertainty analysis must address all important sources of code, nuclear data and input data uncertainty. The methodology for validating and evaluating neutron code uncertainties for SFRs is the same as the methodology for other reactors.
- The experimental justification of the SFR neutronic characteristics is required for validation and verification of computer codes and cross-sections libraries used in SFR design applications.

This report further describes areas in which there was general agreement among participants, and areas in which there were significant variations in opinion, as identified in Chapter 4. In addition to the analysis outcomes that supported the identified common positions, the WGSAR noted the following:

- There is an advancement of analytical tools in predicting SFR behaviour.

- A significant amount of experimental data on neutronics of SFRs has been obtained in previous decades during the development of this reactor technology.
- International co-operation in neutron experiments for advanced SFRs on the basis of bilateral or multi-lateral agreements between countries is advisable and effective.
- Limiting neutronic parameters that bind the best-estimate plus uncertainty values are verified through surveillance as part of a facility's technical specifications. The requirements for establishing and justifying neutron parameter safety limits for advanced SFRs are similar to existing water-cooled reactor requirements.
- The requirements related to neutronic calculation methods, computer codes and group constants used for advanced SFRs are very similar to the requirements for existing conventional power reactors, with some exceptions.
- There is no uniform approach to requirements related to calculated neutronic parameters.
- The list of accident scenarios related to reactivity coefficients for all participants is approximately the same.
- Criticality safety assessments in each WGSAR participating country are based on the country's own experience in the operation of existing reactors, and its own regulatory framework. Participants' regulatory requirements for the storage and transportation of nuclear fuel are also sufficient for SFRs. There are no specific regulatory requirements regarding the analysis codes and methods for the criticality safety assessments of advanced SFRs.
- There are no SFR-specific requirements regarding subcriticality control methods and means (e.g. criticality accident alarm systems) for advanced SFRs, and the reference can be drawn from requirements on existing water-cooled power reactors.
- The criticality codes are validated by comparing benchmark experiment data against calculated results for those experiments.
- The standard list for initial start-up tests includes control rod worth confirmation (integral and detailed differential curve), isothermal temperature coefficients evaluation, feedback reactivity evaluation, measurement of power distribution, and reactivity measurements at each sub-assembly loaded.

# 1. Introduction

This technical report was developed by the NEA Working Group on the Safety of Advanced Reactors (WGSAR). It is part of the activity "Regulatory Perspectives on Safety Aspects Related to Advanced Sodium Fast Reactors", which is to develop technical reports increasing regulators' knowledge on selected safety aspects related to advanced sodium fast reactors (SFR), as well as to identify additional research and development needs to support the regulators' safety review. It was agreed to develop reports based on regulatory experiences in the following technical areas: 1) severe accident prevention and mitigation measures; 2) neutronics and criticality safety; 3) analytical codes; and 4) fuel qualifications.

This report describes the regulatory perspectives on safety aspects related to the calculation and justification of neutronic characteristics and criticality safety assessment of advanced sodium fast reactors and identifies topics that should be investigated in the frame of the SFR's safety regulation, potentially involving additional research and development needs.

Neutronics and criticality safety were identified by the group as topics to be discussed, and a questionnaire was created to gather information from the participants. The present report is based on the answers to this questionnaire.

This activity meets challenge five of the Committee on Nuclear Regulatory Activities (CNRA) Operating Plan and Guidelines (2011-2016) (NEA/CNRA/R(2011)2), i.e. safety in advanced reactor designs, which mentions, in part, that regulatory bodies should anticipate and articulate the requirements necessary for applicants to demonstrate the adequacy of the design in meeting regulations and support its licensing basis.

Eight countries have contributed to this report, taking benefit of their experience in licensing or in their ongoing work on future SFR projects:

- Canada
  - No past or ongoing licensing, but ARC-100 reactor pre-licensing assessment is in progress.
- People's Republic of China (China)
  - China Experimental Fast Reactor (CEFR), with 65 MWth, achieved first criticality on 21 July 2010 and started generating power a year later on 21 July 2011;
  - CFR-600 (1 500 MWth, 600 MWe) is under construction since late 2017.
- France
  - Rapsodie (25 MWth then 40 MWth, non-power) operated from 1967 to 1982;
  - Phénix reactor (565 MWth, 250 MWe) operated from 1973 to 2009;
  - Superphénix reactor (3 000 MWth, 1 240 MWe) operated from 1985 to 1998;
  - ASTRID project.

- Germany
  - KNK-1 reactor (60 MWth, non-power) commissioned in 1972, retrofitted and renamed KNK-2 in 1977, final shutdown in 1991;
  - SNR-300 reactor (762 MWth, 327 MWe), for which construction began in 1972 and then was abandoned in 1991.
- Italy
  - PEC (Prova Elementi Combustibili, 120 MWth, non-power, reactor for testing fuel elements) designed in the 1970s, abandoned in 1987 after the Chernobyl accident.
- Korea
  - PGSFR project (392 MWth, 150 MWe) under design.
- Russia
  - BOR-60 reactor (55 MWth, 12 MWe) in operation since 1968;
  - BN-350 (750 MWth, 130 MWe, former USSR) in operation between 1973 and 1999;
  - BN-600 reactor (1 470 MWth, 550 MWe) in operation since 1980;
  - BN-800 reactor (2 100 MWth, 880 MWe) commissioned in 2016;
  - BN-1 200 project.
- United States
  - Hallam Nuclear Generating Station (Sodium Graphite Reactor) commissioned in 1959, operated from 1963 to 1964;
  - Fermi 1 Nuclear Power Plant (Metal Fueled Fast Breeder Reactor) commissioned in 1956, operated from 1963 to 1972;
  - EBR-II (62.5 MWth, 20 MWe) operated from 1965 to 1994;
  - Fast Flux Test Facility (FFTF, 400 MWth, non-power) operated from 1982 to 1992;
  - PRISM Design (840 MWth, 311 MWe) – The NRC issued NUREG-1368 Pre-application Safety Evaluation Report of PRISM in 1994;
  - ARC-100 (260 MWth, 100 MWe) design under development.

The United Kingdom joined the WGSAR when this report was being finalised and therefore did not participate in the development of the common positions.

## 2. Common positions

This section presents the common positions that were established by the participating NEA Working Group on the Safety of Advanced Reactors (WGSAR) members on the basis of the questionnaire answers detailed in Section 3.

Six common positions were identified:

I. The existing regulatory framework for LWRs is general enough to be applicable to SFRs. Many of the existing neutron characteristic requirements are high-level enough that they apply also to advanced reactors. However, there are some specificities to be taken into account, for example, coolant void reactivity effect, positive effect of mechanical core radial compaction, etc.

II. All participants agree that neutron parameters of SFRs should meet the following requirements:

   a. Design should pay attention to the sodium void reactivity effect, and should minimise the impact of sodium void reactivity effect.

   b. The temperature and power feedback coefficients for the entire reactor which determine the dynamic behaviour of the core during fast transients occurring in normal operation are negative. The reactor shall be designed so that, in the power operating range, the net effect of inherent nuclear feedback characteristics tends to promptly compensate for a rapid reactivity insertion.

   c. An adequate reactivity worth is required for the control rods in order to ensure subcritical shutdown at a given temperature, whatever the plant state. This value integrates a margin to cope with uncertainties.

   d. The anticipated variations of the core geometry during normal operation should be taken into account (thermal expansion, swelling, creeping, subassemblies bending) in the core neutronic design. The reactivity coefficients like the radial expansion coefficient, resulting from the change of core geometry are hard to measure and validate, so conservative approach will be applied for evaluation.

III. All participants agreed that the use of inherent and passive safety with regard to core design of advanced reactors including SFRs should be encouraged to take advantage of the enhanced reliability associated with passive systems.

IV. All participants agreed that the requirements for establishing and justifying neutron parameter safety limits for advanced SFRs are similar to conventional power reactor requirements.

V. All participants agreed that accounting and evaluation of calculation parameter uncertainties is an important part of safety analysis and the uncertainty analysis must address all important sources of code, nuclear data and input data uncertainty. Conservative uncertainty should be included in initial design based on comparisons to benchmark experiments and available operating reactor data, and uncertainty should be re-evaluated by using various core physics tests in commissioning. Technical specifications surveillance requirements should also be provided to verify the uncertainty in neutronics parameters.

VI. All participants agreed that the experimental justification of the SFR neutronic characteristics is required for validation and verification of computer codes and cross-sections libraries used in SFR design applications. Empirical evidence is also needed to establish nuclear reliability factors, which are values placed on calculated neutronic parameters to account for uncertainty.

# 3. Survey results

The survey consists of four thematic areas regarding the regulatory approach to neutronics and criticality safety for sodium fast reactors (SFRs), namely:

- general questions;
- regulatory requirements and guidance related to neutronics calculations for SFRs;
- regulatory requirements and guidance related to criticality safety calculations for SFRs;
- regulatory requirements and guidance related to phenomena to be taken into account in SFR analyses.

## 3.1. General questions

### *3.1.1. Main objectives concerning SFR reactors*

The participants were asked about the main objectives concerning SFR reactors in their country. Most countries have indicated three goals as the immediate objectives of SFRs:

1. Research/investigation. Material and fuel testing.
2. Demonstration of SFR technology reliability and safety.
3. Demonstration of feasibility of plutonium recycling and minor actinide transmutation.

Long-term objectives are to use SFRs for:

4. Commercial power production.
5. Fuel breeding.

In addition, some countries are considering the possibility of using SFRs to support industrial processes such as coal gasification, hydrogen production using high-temperature SFRs, and to power space craft.

Today, there is 67 years of international experience with SFRs since EBR-I start-up, which was the first reactor generated electricity in the world. There is still ongoing research/investigation and demonstration of SFR technology reliability and safety in many WGSAR participating countries.

In the opinion of most participants over the past decades, the analytical tools have become increasingly sophisticated and better able to predict SFR behaviour. Knowledge of and experience with passive safety features in other reactor designs is being gained. In addition, fuel and cladding technology continues to improve. Finally, advanced reactor initiatives and collaboration among the involved parties will help to address safety issues more efficiently.

### *3.1.2. Fuel cycle for advanced SFRs*

The participants were asked about the proposed fuel cycle for advanced SFR designs.

As shown by the answers, SFR development goes along with a closed fuel cycle. However, the terminology for “open fuel cycle”, “closed fuel cycle”, etc. needs to be well defined or clarified. Perhaps, the meaning of “closed fuel cycle” might differ from country to country.

Most of the countries define a closed fuel cycle with delayed fuel refabrication. This cycle shall be optimised to cope with the constraints resulting from the management of the fuel burnt in the LWR fleet.

According to the United States, proposed SFRs could be adapted to either an open or closed fuel cycle. The current status of NRC activities regarding reprocessing are discussed in the US NRC Paper SECY-13-0093, “Reprocessing Regulatory Framework–Status and Next Steps,” (USNRC, 2013) and associated staff requirements memorandum (SRM) 13-0093. Russia is developing the closed fuel cycle for advanced SFRs with prompt fuel recycling after two years of storage on the site of the nuclear power plant.

Fuel utilisation is considerably higher in SFRs than in currently operating reactors. In addition, SFRs are capable of both breeding and burning plutonium. They can also burn minor actinides, providing the implementation of adequate fuel recycling thechnologies. These properties enable a long-term reduction of plutonium stockpile and inventory of minor actinides in final repository, thus reducing radiation hazard.

### *3.1.3. Experimental base for support of advanced SFR neutronics reviews*

Concerning the experimental base for support of advanced SFR neutronics reviews, participants were presented with five questions.

***Test facilities***

The first question focused on test facilities available to provide important data on neutronics and criticality behaviour of newer fuel components and new clad and structural materials.

China stated that the China Experimental Fast Reactor (CEFR) with 65 MWth achieved criticality in July 2010, and was connected to the grid a year later. In France there is no fast reactor available at present. A zero power facility would be available in the future, after refurbishment (MASURCA). Korea does not have test facilities for SFR. It is planned that KAERI will resort to foreign test facilities.

Russia has a research fast reactor BOR-60, the reactors prototypes BN-600, BN-800, and two critical facilities BFS-1 and BFS-2 used for neutronics and criticality investigations.

The United States presented several facilities that are relevant to SFR review and analysis. These include:

- The annular core research reactor (ACRR) at Sandia National Laboratory is a water pool-type reactor used for radiation effects studies and reactor safety experiments. Capabilities include fuel transient testing under abnormal and accident conditions for SFRs, LWRs and GCRs. The ACRR has been used to perform nuclear heating tests on SFR, LWR and GCR fuels.
- The core alloy flow and erosion (CAFE) facility at Argonne National Laboratory is used for generating kilogram quantity melts of low radiological-hazard nuclear fuel materials and discharging them into various receiver geometries. This facility may be useful in SFR studies involving chemical interactions between metallic fuels and cladding, and between metallic fuels and structural materials.
- The melt coolability and concrete interaction (MCCI) facility at Sandia National Laboratory has been used to conduct integral and separate effects tests on reactor

materials and examine ex-vessel severe accident core melt issues. Being tolerant of experiments involving a high degree of hazard, the facility provides a working environment relevant to SFR testing for sodium-concrete, sodium-water and fuel coolant interaction investigations.

- At Argonne National Laboratory there is a facility for sodium technology and advanced component testing. The facility provides a flexible test bed for testing and evaluating advanced SFR components and examining the sodium compatibility of advanced materials.
- Surtsey is a facility at Sandia National Laboratory that has been used for LWR severe accident studies. Surtsey can be used for direct atmospheric heating and large tests involving molten materials and their interactions. Surtsey can also be used to generate data for sodium fire phenomena.
- The transient reactor test facility (TREAT) at Idaho National Laboratory is an air cooled irradiation test facility designed to evaluate reactor fuels and structural materials. Slow power transients under special conditions can be designed to test the behaviour of various fuels and structural materials under SFR accident conditions.
- The advanced test reactor (ATR) at Idaho National Laboratory is capable of testing fast reactor fuels and materials in a thermal spectrum by using cadmium filtering. Studies have been performed that show that ATR irradiations performed using cadmium shrouding are sufficiently prototypic that they can be used with confidence in the development and testing of fast reactor fuels (INL 2017).
- The versatile test reactor (VTR), to be located at Idaho National Laboratory, is under development and will provide a versatile, reactor-based, fast neutron source. The VTR intended to provide testing capabilities for advanced fuels, materials, and instruments and sensors to support the existing fleet and the development of net-generation nuclear reactors. The VTR is a 300 megawatt (thermal) sodium-cooled, pool-type reactor.

Germany gave information on experimental facilities for investigations of liquid metal technologies and thermo-hydraulics relevant for SFR. In Germany several research organisations formed the so-called Helmholtz alliance on liquid metal technologies (LIMTECH). Within this framework the KASOLA (KArlsruhe SOdium LAboratory) test section at the Karlsruhe Institute for Technology (KIT) is used for:

- development of free surface liquid metal targets for accelerator applications;
- development of models to describe free surface liquid metal flow;
- investigation of transition in convective flow patterns between forced, mixed and free convection modes;
- qualification of CFD and system codes to simulate adequately the transition from the channel flow to large plenum (collector tank);
- thermal-hydraulic investigations of flow patterns in fuel bundles or pool configurations at prototypical or scaled heights.

***International co-operation in neutron experiments for advanced SFRs***

The second question concerns bilateral or multi-lateral agreements between countries regarding the sharing of experimental, research and test facilities or neutronic and reactivity related data from those facilities. Korea reported that KAERI will utilise the data of ANL for the design and a mock-up facility for metallic fuel is installed in the Russian BFS critical experimental facility for tests. Russia has confirmed that it has bilateral agreement with Korea on metal fuel loading modelling at the BFS test facility.

In Europe, multi-lateral research co-operation is also achieved through Euratom projects such as the ESFR-SMART and SESAME projects of the Horizon 2020 EU framework programme for research and innovation.

According to the US experts, the NRC has not engaged with reactor vendors whose designs depart significantly from designs where benchmark data is available (e.g. EBR-I and EBR-II). The NEA *International Handbook of Evaluated Reactor Physics Benchmark Experiments* (www.oecd-nea.org/science/wprs/irphe/handbook.html) also contains several fast reactor benchmarks.

***Regulatory experience***

The third question was focused on regulatory experience of the WGSAR member's countries with experimental, prototype or commercial SFRs. In response to this question, China reported that experience on licensing, commissioning and safety reviews is being accumulated during the design, construction and acceptance of the China Experimental Fast Reactor Project.

France pointed out that safety assessment of past SFRs was performed based on expert analysis and making reference to PWRs safety requirements. In 1983, the safety authority has issued a letter setting up the main safety requirements and characteristics of a standard SFR consistent with the requirements issued for the French 1 300 MW PWR fleet. It was noted that this document is largely obsolete and therefore not applicable to future SFRs.

Germany referred to the experience related to safety reviews and pre-licensing of the SNR-300 type reactor in Kalkar in the 1980s and the licensing of the KNK research reactor in Karlsruhe. At the time of licensing the Nuclear Reactor Safety Commission (RSK) recommendations specific to SFR were used.

Korea answered that KINS has no experience of licensing review for SFR until now.

Russia referred to the experience of safety reviews, licensing, commissioning and operation of experimental SFRs BR-10, BOR-60 and prototypes BN-600, BN-800.

Regulatory experience of the United States for SFRs was gained in the reviews of FFTF, the Clinch River Breeder reactor (CRBR), and the PRISM reactor. Review of the FFTF was initiated under the Atomic Energy Commission (AEC) and continued by the NRC during the 1970s. The FSAR was approved by the NRC in 1978. During the 1970s and early 1980s the United States Department of Energy (US DOE) attempted to license the CRBR and the NRC was reviewing the design. However, the US Congress eliminated funding of CRBR before completion. The CRBR licensing did result in a safety evaluation report (NUREG-0968 [USNRC, 1983]).

In the 1990s the NRC conducted a preliminary review of General Electric's PRISM design. The review highlighted several key regulatory issues, and a preliminary report on the review was published (NUREG-1368 [USNRC, 1994]).

***Advanced alloy materials for new SFRs***

When asked about the introduction of advanced alloy materials to reduce swelling and radiation damage to fuel clad and ducts, and the need for more confirmatory research and testing of these new materials, impacted regulation of SFRs the WGSAR members provided the following answers:

China reported that the advance alloy material is under development, and has not been employed yet.

In France, there is no regulation dealing with cladding and sub-assembly materials, but each irradiation test is subject to a safety analysis and a formal authorisation on a case-by-case basis. Cladding material foreseen for the ASTRID fuel has been tested in the Phénix reactor and is the result of a long optimisation process.

Korea answered that a lot of new materials were developed and licensed in the past. The performance of new materials for SFR fuel will be reviewed based on the past experience of PWR licensing. KINS certainly would require the same level of proven data (in-pile, out-pile) and burn-up data necessary for the licensing burn-up.

Russia stated that according to the existing rules every advanced alloy material for SFRs is tested consecutively at out-of-pile installations, in the research reactor, and in the reactor-prototype. For example, at present steel for cladding of fuel pins of BN-1200 with nitride fuel is being tested in BN-600.

The United States reported that the introduction of new alloys could be an area of importance if new SFR designs are submitted for NRC review, however the NRC is currently engaged with vendors whose designs include materials with an established database.

***Regulatory changes in response to improvements in SFR core design***

The last question on this topic was about regulatory changes have been made or perceived, including the examination or use of guidance documents, in response to improvements in SFR core design and inherent, passive heat removal features.

China and France have no regulatory change due to specific of the SFR. Korea believes that the current regulatory requirements for PWR neutronic design could be applied to SFR if take into account some SFR-specific features. Specific regulatory requirements for passive heat removal system are not fully established even for PWR. The regulator hopes to resolve the issue through a technical discussion with designer.

Russia reported that today the vast majority of Russian regulatory documents apply to all types of reactors, including the SFRs. In the “Nuclear Safety Rules for Reactor Installations of Nuclear Power Plants”, NP-082-07 (Rostechnadzor, 2007) there are only two points directly concerning SFR:

1. For SFRs it must be shown that during the normal operation and in case of deviation from the normal operation, including design-basis accidents, the formation of voids in the sodium coolant are eliminated.
2. Operational limits for damage of SFR fuel pins:
    - defects of the gas leakage – < 0.05% of the number of FPs in the core;
    - direct contact of nuclear fuel with coolant –< 0.005% of the number of FPs in the core.

    Safety limits for damage of SFR fuel pins:

- o defects of the gas leakage – < 0.1% of the number of FPs in the core;
- o direct contact of nuclear fuel with coolant –< 0.01% of the number of FPs in the core.

There is only one federal regulatory document aimed specifically at SFRs – "Requirements for the Contents of Safety Analysis Report for NPPs with Fast Neutron Reactors", NP-018-05 (Rostechnadzor, 2005a).

The United States informed that the NRC is preparing for the possible review of new SFRs. As part of this preparation, the NRC is reviewing its regulatory guidance as it applies to advanced non-LWRs and is developing a regulatory framework that is better suited to these types of designs. For example, New Regulatory Guide 1.232, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors," (USNRC, 2018a) contains advanced reactor design criteria (ARDC), including SFR design criteria (SFR-DC). The ARDC and SFR-DC are modified versions of the existing general design criteria (GDC) in title ten of the *Code of Federal Regulations* (10 CFR) Part 50, Annex A (USNRC, 2017), which were developed for LWRs.

There was general agreement that significant change in the neutronics part of the existing regulatory documents or the development of review guidance for a specific technology should not be expected.

## 3.2. Neutronics

### *3.2.1. Regulatory documents containing neutron characteristics requirements*

Participants were asked what regulatory documents in their country contain neutron characteristic requirements (e.g. reactivity effects and coefficients, control rods efficiency, reactivity balance) and what these requirements are.

Canada listed the following regulatory documents:

- REGDOC 2.5.2, Design of Reactor Facilities: Nuclear Power Plants (CNSC, 2014a);
- RD-367, Design of Small Reactor Facilities (CNSC, 2011);
- REGDOC 2.4.1, Deterministic Safety Analysis (CNSC, 2014b).

Canada also pointed out that these REGDOCs specify technology-neutral requirements at a very high level.

China replied that there are civilian nuclear safety supervision and management regulation, especially one of its enforcement regulations, the HAF102 (NNSA, 2016) and HAF201. The requirements involve the shutdown margin and defence in-depth in case of emergency.

According to France's reply, neutron characteristics of reactor cores are not specified in French regulation.

In Germany, requirements regarding the core design for example related to reactivity effect and coefficients, neutron flux distribution and control rods are given in the safety requirements for nuclear power plants. These requirements are further detailed in relevant KTA standards focusing on German LWR, e.g. KTA 3101.1-3 (Design of reactor cores of pressurised water and boiling water reactors) (KTA, 2015), KTA 3103 (Shutdown Systems for Light Water Reactors) (KTA, 1982) and KTA 3204 (Reactor pressure vessel internals) (KTA, 1998). At the time of licensing of the SNR-300 in the 1980s, in particular RSK (reactor safety commission) recommendations specific to SFR were available.

These addressed requirements for various aspects of the SNR-300 design, such as:

- neutronics calculations;
- negative power coefficient and prompt reactivity feedback;
- calibration measurements of the reactivity for both shutdown systems before each fuel cycle;
- calibration of the shutdown system and determination of the (isothermal) temperature and flow coefficient during commissioning;
- the secondary independent and diverse shutdown system, which had to function even after core deformation (loss of core geometry);
- the two independent, spatially separated reactor protection systems with independent fast scram systems consisting of a primary system with control rods (from top, 0.7s injection time, 83 cm distance) and a secondary system with absorber chains driven by a set of springs (from bottom, 0.5s injection time, 105 cm distance).

Korea answered that the requirements are stipulated in the Korean atomic law. The main contents are similar to the NRC requirements. The draft version of requirements for SFR include the reactivity coefficients and many PWR requirements might be applicable also to SFRs.

In Russia, the basic neutron characteristic requirements are contained in the federal regulation NP-082-07 (Rostechnadzor, 2007). These requirements include:

- The reactor core must be designed so that any changes of reactivity during normal and abnormal operation, including design-basis accidents, do not lead to violation of appropriate damage limits for fuel pins.
- Characteristics of the core and control rods must be such that the insertion into core for any combination of their locations during normal and abnormal operation, including design-basis accidents, provided the introduction of negative reactivity at any stretch of their movement.
- The reactor design must provide at least two reactor shutdown systems, each of which must be able, independently of one another, to ensure the transfer of the reactor to a subcritical state and maintaining it in subcritical condition considering the principle of single failure or personnel error.
- At least one of the available shutdown systems must perform the function of scram.
- The rate of reactivity increase by the control rods must not exceed $0.07\beta_{ef}$/s. For CRs with efficiency of more than $0.7\beta_{ef}$ input of positive reactivity must be quantised, with efficiency of the step is not more than $0.3\beta_{ef}$.
- Subcriticality of the reactor after rising the scram in the operating position with other CRs inserted must be not less than 0.01 in the core with the maximum $K_{ef}$.
- Minimum subcriticality of the reactor in the refuelling process, taking into account possible errors, must be not less than 0.02.
- The values of the temperature reactivity coefficient, reactor power reactivity coefficient and total coolant and fuel temperature reactivity coefficient must be negative in the whole range of parameters of reactor during normal and abnormal operation, including basis design accidents. For beyond-design-basis accidents the

range of allowed values of the sodium void effect must be justified in the design of reactor and nuclear power plant.

Most of these regulations apply to other reactor types.

The United States answered that reactivity control requirements for SFRs are addressed in the advanced reactor design criteria (ARDC), regulatory guide RG-1.232, Section three, "Reactivity Control" (USNRC, 2018a).

From the answers received it follows that the many of the existing neutron characteristic requirements are high-level enough that they apply universally. However, there are differences, and to capture those differences, it may be developed specific regulatory documents such as RG-1.232 in the United States.

### *3.2.2. Rules, regulations and guidance to increase core inherent and passive safety of advanced SFRs*

As in the case of neutron characteristics there are no special requirements or guidance to increase the use of inherent and passive safety of advanced SFRs. Designers use recommendations developed for existing reactors.

Canada presented the following list of RD's and rules aimed at increasing inherent safety:

REGDOC 2.5.2 (CNSC, 2014a), Section 4.2.4, *Accident Mitigation and Management:* "measures shall be taken to mitigate the radiological consequences of accidents" and the measures shall include "consideration of inherent safety features";

REGDOC 2.5.2, Section 4.3.1, *Defence-in-Depth* and Section 6.1, *Application of Defence-in-Depth*:

- DiD level two: control of postulated initiating event (PIE) using both inherent and engineered design features to minimise or exclude uncontrolled transients to the extent possible;
- DiD level three: minimisation of the consequences of accidents by providing design inherent safety features;

REGDOC 2.5.2, Section 6.3, Accident Prevention and Plant Safety Characteristics:

"Following a PIE, the plant is rendered safe by:" inherent safety features, passive safety features (among others);

REGDOC 2.5.2, Section 8.1, Reactor Core: "The design of the core shall be such that:" "maximum degree of positive reactivity and its maximum rate of increase by insertion in operational states and DBAs are limited by a combination of the inherent neutronic characteristics of the core", etc.

In addition, if inherent design characteristics are not sufficient to compensate for PIEs other compensatory design features must be provided.

REGDOC 2.5.2, Section 7.6.2, Single-Failure Criterion ("All safety groups shall function in the presence of a single failure"): "Unintended actions and failure of passive components shall be considered as two of the modes of failure of a safety group."

"Exemptions for passive components may be applied only to those components that are designed and manufactured to high standards of quality, that are adequately inspected and maintained in service, and that remain unaffected by the PIE"; extensive justification of the exemption is required.

China answered that it uses civilian nuclear safety supervision and management regulation, especially one of its enforcement regulations, the HAF102 (NNSA, 2016). At the same time, China is participating in the development of the safety design criteria/guidelines of SFR within the Generation IV International Forum, which would benefit the safety design of SFRs in China.

France reported that designers aim at developing the use of appropriate combinations of passive and active systems for the ASTRID project, but there is no regulatory guidance on this topic.

Italy noted that the incorporation of passive shutdown systems is being assessed in the frame of passive systems implementation studies in future SFRs: to this aim an ad hoc group has been established at IAEA.

According to Korea’s answer, inherent safety designs are generally required even for the current PWR design. The regulations on technical standards for nuclear reactor facilities, etc., of Korean atomic law (“The Atomic Energy Law”, the Republic of Korea, 1958) stipulates in the Article 26 (inherent protection of reactor): “The reactor core and associated coolant systems shall be designed so that, in all power operating range, the net effect of prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity”.

On the other hand, regulators are not sure whether passive safety features can be activated reliably and increase safety in real accident case.

According to Section 3.1.10 of Russian regulation “General Safety Provisions for Nuclear Power Plants”, NP-001-15 (Rostechnadzor, 2016), in the design of systems (elements) of nuclear power plants and reactors, preference must be given to systems (elements) and devices with passive principle of action and properties of the inherent safety (self-regulation, thermal inertia, natural circulation and other natural processes). At present the special regulations, and guidance for the increase of use of inherent and passive safety of advanced SFRs are absent, although BN-1200 designers incorporated them in their project (passive shutdown system, sodium plenum above the core, passive emergency cooling system). The US Nuclear Regulatory Commission, in their advanced reactor policy statement (73 FR 60612; 14 October 2008 [USNRC, 2007]) states that, “the Commission expects that advanced reactors [including SFRs] will provide enhanced margins of safety [over existing light water reactors] and/or use of simplified, inherent, passive or other innovative means to accomplish their safety and security functions.”

All participants agreed on the need to increase the use of inherent and passive safety of advanced reactors including SFRs. This requirement is contained in the existing regulatory documents of the WGSAR member’s countries and applies to all the types of reactors. The special regulations and guidance for the increase of use of inherent and passive safety of advanced SFRs are absent.

### *3.2.3. Advanced SFR fuel designs*

When asked about the advanced SFR fuel design which are currently under review or are expected to be submitted for review in the near future, the WGSAR members provided different answers.

China and France plan to use traditional MOX fuel.

Germany explained that the development process of the SNR-300 in the 1970s started with the usage of metallic fuel and an external breeder design. At that time, the doppler effect for SFR was not understood completely and the negative power coefficient of metallic fuel due to expansion was the safer option. This changed with time and the final design was an internal breeder with MOX fuel. Oxide fuel was at that point the more economic choice and allowed higher burn-up. Metallic fuel burn-up then was limited to 10-15 GWd/t due to swelling.

Korea (KAERI) plans to load U-Zr metallic fuel for the initial core and then it hopes to move to load the U-TRU-Zr fuel finally, in the future.

Russia stated that currently in the BN-600 nine experimental FAs with nitride uranium-plutonium fuel for BN-1200 and lead cooled reactor BREST-300 are tested.

In the United States, a potential vendor design is based on fuel similar to that used in EBR-II.

The following comments were received on additional questions regarding the prospects for licensing of new types fuel for SFR (metallic and nitride) and the use of new fuels for projects PRISM, terra power traveling wave reactor and BN-1200. In the United States, metallic fuel seems to be the top choice for research and development and for commercial use because it does not react aggressively with sodium coolant in the event of fuel failure, as oxide fuel would, and the higher fuel thermal conductivity of metallic fuel leads to lower temperatures. Review of a new fuel type will commence if and when a new reactor application using that fuel is received. NRC is currently engaged in pre-application activity with a vendor that is proposing metallic fuel. Use of oxide fuel has not been discussed. There is no information about any current regulatory engagement on the PRISM and standing wave reactor designs. The Russian BN-1200 project provides for possibility of using both oxide and nitride fuels.

### *3.2.4. Requirements for neutronic calculations*

The participants were asked about the requirements for neutronic calculations pertaining to normal operation, anticipated operating occurrences, DBAs and BDBAs.

Canada answered that these requirements and guidance are given in Section 8.1, *Reactor core*, of REGDOC 2.5.2 (CNSC, 2014a).

France declared that the former SFR cores of the Phénix and Superphénix reactors fulfilled the following criteria:

- be subcritical with a minimum USD 10 margin in cold shutdown state with all neutron absorber rods inserted;
- be equipped with redundant shutdown system (cold shutdown);
- have in addition, a complementary system (other control rods or device insuring the insertion in the core of existing control rods) that enable to achieve hot shutdown even in case of light degradation of core geometry;
- guarantee subcriticality for the most penalising core configuration (maximum number of fresh subassemblies) in case of single error in loading pattern;
- have negative global feedback coefficients associated to inlet temperature, mass flow and power, in all configurations.

Any new project will have to fulfil at least these criteria. Up to now, they have not been included in a regulatory document.

According to Korea the regulatory requirements for neutronic calculations for SFR are absent at present time. The following are general requirements pertaining to reactor design in the draft version of safety requirements for SFR. Main requirements are similar to PWR requirements with changes for SFR-specific design features.

In the United States, the requirement on neutronic calculations during normal operations are driven by technical specifications, which typically require that the reactivity balance be within 1 000 pcm of predicted values. For anticipated operational occurrences (AOOs) and design-basis accident analyses, applicants and licensees are required to demonstrate compliance with acceptance criteria (associated with fission product barriers) using suitably conservative modelling techniques and input parameters. The use of suitably conservative modelling is driven by the design criteria (GDCs and ARDCs) associated with the fission product barriers. Simple point kinetics models are frequently used for these analyses. For beyond-design-basis analyses, applicants and licensees have been permitted to perform best-estimate calculations.

As shown by the answers, the main requirements for neutronic calculations are similar to conventional power reactor requirements with small changes for SFR-specific design features.

### *3.2.5. Requirements for establishing and justifying neutron parameter safety limits*

The participants were asked about the requirements for establishing and justifying neutron parameter safety limits for advanced SFRs.

China, France and Russia reported that there are no specific requirements to establish and justify neutron parameter safety limits for advanced SFRs. The safety limits related to core physical parameters must be presented and justified by the designer. In this case the requirements of conventional power reactors are taken as reference.

Korea said that "we do not impose any safety limits for neutron parameters, even for PWR".

In the United States, applicants and licensees account for the uncertainty in significant neutronic parameters (e.g. reactivity balance, control rod worth, power shapes, kinetics parameters). This uncertainty includes both code bias and bias uncertainties, as determined through comparisons to experimental data, and measurement uncertainties. This uncertainty is initially treated conservatively and is then updated as operating data is accumulated. Limiting neutronic parameters that bound the best-estimate plus uncertainty values are used in the plant's safety analyses. Limiting neutronic parameters are verified by surveillance as part of a facilities technical specifications. The requirements on these limits are captured by design criteria (GDC and ARDC) associated with reactor inherent protection and fission product barriers (see Section 3.4).

As in the previous paragraph it follows from the answers that the requirements for establishing and justifying neutron parameter safety limits for advanced SFRs are similar to conventional power reactor requirements.

### *3.2.6. Requirements related to neutronic calculation methods, computer codes and group constants*

This question was focused on the requirements related to neutronic calculation methods, computer codes and group constants used for advanced SFRs in WGSAR participating countries.

As follows from the answers received from China there are no specific requirements on this point established for advanced SFRs at present. In France, it is up to the designer to develop a computation methodology that is able to describe the behaviour of the core during any operational or accidental transient. Then, the designer has to define code domain of validity and justify its representativeness regarding physical phenomena. Guidance exists on the general methodology to validate a computer code used for the safety case accidental studies. Neutron parameters that are calculated in the design phase shall be verified by appropriate start-up tests.

Korea reported that the requirements and guidelines of NUREG-0800 Section 4.3 (USNRC, 2007) will be referenced, basically, taking into account the SFR-specific features when reviewing the safety of PGSFR nuclear design. The uncertainty of nuclear data for TRU and sodium related nuclides, not reviewed seriously in case of PWR will be evaluated in detail.

Since fast reactor design resorts heavily to middle range of neutron energy spectrum, compared to two-group (thermal and fast neutrons) for PWR nuclear design, and utilises the multi-group cross-sections in the nuclear design, the adequacy of multi-group energy approximation used in the design will be evaluated with high priority.

In Russia, there are no specific requirements on this point established for advanced SFRs. The general recommendation is to use modern, well-verified and validated codes and nuclear data. At present time new generation codes for advanced fast reactor calculations are being developed and verified.

As discussed in Section 3.4, in the United States, design-basis analyses are performed using suitably conservative modelling techniques and input parameters. The use of suitably conservative modelling is driven by the design criteria (GDC and ARDC) associated with the fission product barriers. Outside of quality assurance requirements (Annex B to 10 CFR Part 50), there are no regulations that require specific modelling practice. However, review guidance (Section 4.3 of the standard review plan [SRP], NUREG-0800 [USNRC, 2007]) clarifies that the analytical methods and database should reflect the state-of-the-art. In addition, SRP Section 15.0.2 and regulatory guide 1.203 (USNRC, 2005) provide guidance on transient and accident analysis methodology, including the evaluation model development and assessment process (EMDAP).

As can be seen from the answers the requirements related to neutronic calculation methods, computer codes and group constants used for advanced SFRs are very similar to the requirements for conventional power reactor with the exception of the requirement to use multi-group cross-sections in the calculation of SFRs instead of the two-group commonly used in PWR calculations.

### *3.2.7. Requirements related to calculated neutronic parameters*

The participants were asked about the requirements related to calculated neutronic parameters, including:

- reactor core power distribution;
- fuel burn-up distribution;
- fuel temperature and power reactivity effects and coefficients;
- reactivity effects due to changing of core geometry and dimensions;
- reactivity effects due to changing of sodium density;
- control rods efficiency, including passive reactor shutdown means;
- reactivity change with fuel burn-up;
- impact of the presence of actinides in the fuel on reactivity feedback effects during normal operation, DBAs, BDBAs, and during reactor shutdown conditions;

- sensitivity of core reactivity to core geometry, assuming this sensitivity is used to give negative feedback during an accident;
- minimising the impact of positive sodium void reactivity in core design;
- subcriticality at fuel reloading;
- characteristics of delayed neutrons;
- characteristics of residual heating.

In accordance with the Canadian response, the REGDOC 2.5.2 (CNSC, 2014a) provides generic high-level core design requirements and guidance in Section 8.1, *Reactor core*, which address most of the list above in the following subsections: defence-in-depth, core power densities and distributions, reactivity coefficients, criticality, core stability, analytical methods, core internals and vessel. In addition, relevant Canadian CSA standards are provided. In RD-367 Section 7.13, *Guaranteed shutdown state* (CNSC, 2011), requirements for shutdown depths are provided. It is emphasised that in general terms the design requirements demand consideration of all plant states (NO, AOO, DBA, DEC) in derivation of key design parameters.

China reported that there are no explicit requirements in the regulation level on these technical points, except for the overall safety goals.

France provided the requirements for several characteristics from the above list, namely:

- The sum of temperature feedback coefficients is negative. The same requirement was given for the resulting power feedback coefficient.
- The temperature and power feedback coefficients for the entire reactor which determine the dynamic behaviour of the core during fast transients occurring in normal operation are negative.
- A minimal efficiency is required for the control rods in order to ensure subcritical shutdown at a given temperature, whatever the core configuration. This value integrates a margin to cope with uncertainties.
- Although the expected variations of the core geometry during normal operation are taken into account (thermal expansion, swelling, creeping, subassemblies bending) in the core neutronic design, there is no mandatory requirement.
- Core must be subcritical at reloading state, to ensure cold shutdown (around 250°C) providing that all control rods are inserted and taking into account the inadvertent replacement of an absorber rod by a fissile sub-assembly.

Korea believes that the need to establish requirements might be decided after performing licensing review at least once. It is expected that the uncertainty of nuclear data for MA and $^{23}$Na etc. might be large and have a significant impact on nuclear characteristic parameters. It will be evaluated as a priority. In addition, Korea provided comments on the calculation of a number of neutron characteristics from the list above:

- The reactivity coefficients like the radial expansion coefficient, resulting from the change of core geometry, are hard to measure and validate, so a conservative approach will be applied for evaluation.
- The uncertainties for sodium density and sodium void coefficient appears to be large because of the uncertainties in the nuclear data of $^{23}$Na etc. Thus we expect detailed evaluation for these parameters is needed.

- The sensitivity of reactivity coefficient induced by the change of core geometry has direct effect in producing the so-called “negative reactivity trip (AURN of Phénix)”, so detailed evaluation is needed to have confidence against this kind of phenomena occurring.
- Design should minimise the sodium void reactivity, but there is a need to balance the design taking into account other safety parameters. The final decision should be made through the safety analysis evaluation, so to not impose any limit values against sodium void reactivity for a moment.

Russia reported that at present time there are regulatory requirements only for fuel temperature and power reactivity coefficients, control rods efficiency and subcriticality at fuel reloading. Design should minimise the positive sodium void reactivity. Regulatory requirements for the rest of listed above characteristics are absent, but for some of them, design criteria for BN-1200 are established. For example, total sodium void reactivity effect must be negative, reactivity change with fuel burn-up must be less than $\beta_{ef}$, the efficiency of a passive reactor shutdown system must be sufficient for the reactor shutdown without actuation of the scram.

In the United States, calculated neutronics parameters need to account for uncertainty (as described in Section 3.5). Neutronics calculations are necessary to demonstrate compliance with the GDC and ARDC. For example, as required by GDC 11 and ARDC 11, the systems that contribute to reactivity feedback shall be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity. In addition, many of the neutronic parameters and core power distributions are surveyed in accordance with 10 CFR 50.36 (USNRC, 2018b), “technical specifications,” as they represent initial conditions to a design-basis accident or transient analysis.

In summary, there is no uniform approach to requirements related to calculated neutronic parameters. In some countries there are no explicit neutronic parameter requirements in the regulation levels, except for the overall safety goals. In one case provided to the WGSAR, regulatory requirements reflect the sign and the values of several characteristics from the above list. Some requirements are established as design criteria in the SFR project.

### *3.2.8. Accident scenarios related to reactivity coefficients*

The participants were asked what accident scenarios related to reactivity coefficients are considered by their country.

China considers mainly leakage of loops or main equipment, accidental insertion of reactivity, ULOF, and the breaking of fuel element.

In Germany, design-basis accident of the SNR-300 have considered the Bethe-Tait accident scenario:

- loss-of-flow accident through failure of main coolant pumps;
- failure of both SCRAM systems.

Through boiling of sodium and the positive void coefficient a power excursion would be caused which stops only through fuel expansion. The regulatory body issued a requirement that for the SNR-300 a mechanical stress due to an energy release of 370 MJ had to be considered. The maximum energy release was later determined to be a factor 4 lower.

According to Italy the accident scenarios related to reactivity coefficients concern the ATWS events, as for instance UTOP, ULOF and ULOHS, where the various reactivity feedbacks (Doppler, thermal expansion of core components, coolant density variation, neutron leakages, etc.) play a fundamental role in the evolution of the accident itself.

In Russia, the following accident scenarios related to reactivity coefficients are considered DBAs: the unauthorised movement of the rods; the penetration of gas bubbles or hydrogen-containing substances into the core; and the deterioration of the heat removal. BDBAs are also considered such as UTOP, ULOF, ULOHS without scram. It is emphasised that practically all emergency scenarios which affect the core are related to reactivity coefficients.

In the United States, reactivity feedback is considered for all transient and accident analyses. The most significant reactivity events are the reactivity induced accidents (RIAs), which are typically evaluated using higher fidelity modelling techniques (e.g. 3-D kinetics). Further information regarding neutronic calculations in the United States is provided in Section 3.4.

As can be seen from the responses received, the list of accident scenarios for all participants is very similar. This list includes DBAs caused by unauthorised movement of the rods, the penetration of gas bubbles or hydrogen-containing substances into the core, the deterioration of the heat removal and BDBAs such as UTOP, ULOF, ULOHS without scram (ATWS). In all of these scenarios the various reactivity feedbacks (Doppler, thermal expansion of core components, coolant density variation, neutron leakages, etc.) play a fundamental role in the evolution of the accident itself.

### *3.2.9. Requirements regarding experimental justification of SFR neutronic characteristics*

The question was focused towards the requirements regarding experimental justification of the SFR neutronic characteristics used in SFR design applications.

China answered that these requirements are contained in the HAF201, one of the codes of Nuclear Safety Regulation.

France reported that there is no specific requirement. The calculation codes used for safety demonstration must be validated on a representative experimental database.

According to Korea uncertainty of the nuclear design computer codes depends heavily on the nuclear characteristics of specific core type. Similarity and applicability (in terms of critical mass, spectral index, reaction rate, etc.) of the designed core with the experimental facility used in validating the computer codes needs to be evaluated.

Russia explained that experimental justification of the SFR neutronic characteristics (critical mass, control rod worth, reaction rate and power distributions) in validating the computer codes needs are used test facilities BFS-1 and BFS-2, the results of the start-up experiments (the last one obtained on the BN-800) and the results of the measurements on operating SFRs (BOR-60, BN-600, BN-800).

In the United States, applicants and licensees account for the uncertainty in significant neutronic parameters (see Section 3.5). Suitably conservative neutronic parameters are used in the plant’s safety analyses to demonstrate compliance with design criteria (GDC and ARDC) associated with reactor inherent protection and fission product barriers. Empirical evidence is used to justify the conservatism of the neutronics parameters. The uncertainty in the neutronic parameters is verified and updated as operating data is accumulated through start-up testing and normal surveillance.

All participants agreed that the experimental justification of the SFR neutronic characteristics is required for validation and verification of computer codes and cross-sections libraries used in SFR design applications. Empirical evidence is also needed for establishing nuclear reliability factors, which are values placed on calculated neutronic parameters to account for uncertainty.

### *3.2.10. Requirements regarding accounting and evaluation of calculation parameters uncertainties*

The participants were asked about the requirements regarding accounting and evaluation of calculation parameter uncertainties in their countries.

Canada explained that REGDOC 2.5.2 (CNSC, 2014a) defines the uncertainty analysis as: "The process of identifying and characterising the sources of uncertainty in the safety analysis, evaluating their impact on the analysis results, and developing–to the extent practicable–a quantitative measure of this impact."

The design calculation should cover and be supported by:

- uncertainty analyses for nominal values, including the magnitude of the uncertainty and the justification of the magnitude (by examination of the accuracy of the methods used in calculations), and comparison, where possible, with reactor experiments;
- a combination of nominal values and uncertainties to provide suitably conservative values for use in reactor steady-state analysis (primarily control requirements), stability analyses, and the AOO and accident analyses.

Code uncertainties must be quantified in the analysis. Safety margins are required to cover the uncertainties of the analysis.

REGDOC-2.4.1 "Deterministic safety analysis" (CNSC, 2014b), Section 4.4.2.7: The uncertainties should be accounted for accordingly, either in the conservative analysis or in the best-estimate-plus-evaluation-of-uncertainties methodologies.

In the safety analyses for level two and level four defence in-depth (where a realistic, best-estimate analysis method may be used) it is not necessary to account for uncertainties to the same extent.

China answered that the uncertainty of calculation parameters should be provided for review, but there is no explicit requirement on the value of the uncertainty.

France answered that analysis of code numerical accuracy and impact of nuclear data uncertainties are essential in the safety demonstration when computing the neutronic parameters of the reactor. However, no SFR-specific requirements have been developed in France.

Korea expects the experimental data for which the similarity and applicability are validated to be used in the SFR core design are not sufficient, so the current statistical methodology applied to the PWR design should not be used in SFR design. It hopes that the methodology to evaluate the uncertainty of the nuclear parameters to be developed through international research programme.

Russia stated that according to Section 1.2.9 "General safety provisions for nuclear power plants", NP-001-15 (Rostechnadzor, 2016), deterministic analyses of design-basis accident must be based on a conservative approach, which in particular involves using the most unfavourable values of the calculation parameters. Safety analyses must be accompanied by estimates of errors and uncertainties in the obtained results. All codes used for the safety analysis must be certified. Certificates of the codes contain the evaluated calculation parameters uncertainties that should be taken into account in the safety analysis.

According to the United States, the uncertainty analysis associated with safety analyses must address all important sources of code uncertainty, including the mathematical models in the code and user modelling. The major sources of uncertainty must be addressed consistent with their importance to the model and to the figures of merit for the calculation. When the code is used in a licensing calculation, the combined code and application uncertainty must be less than the design margin for the safety parameter of interest. The analysis must include a sample uncertainty evaluation for a typical plant application.

Germany and Italy did not provide answers to this question.

All participants agreed that accounting and evaluation of calculation parameter uncertainties is an important part of safety analysis and the uncertainty analysis must address all important sources of code and input data uncertainty.

### *3.2.11. Uncertainties in nuclear data*

The participants were asked how their country considers the uncertainties in nuclear data, including:

- nuclear data of MA, $^{23}$Na;
- fission data of Pu isotopes;
- $^{238}$U and Fe inelastic cross section, etc.

China answered that the covariance matrix of nuclear data should be taken into consideration during the design phase of the reactor.

France answered that although no mandatory requirements have been set up, it has been recommended that SFR designers present their methodology for assessing uncertainties on the neutronic parameters of the core as well as the code validation test matrix. Uncertainties in nuclear data should be evaluated based on extensive set of benchmarks and are the result of international work of experts. As an example, the ASTRID designer proposed to submit to the safety authority a document describing the validation of the numerical codes used for neutronic parameters computation, including the evaluation of the uncertainties on the parameters and the neutronic data, and, as necessary, the analysis of benchmarks.

Russia explained that the impact of the uncertainty of nuclear data on the integral parameters such as $K_{eff}$, the breeding ratio, sodium void reactivity effect is estimated by using the covariance matrix. To assess the impact of these uncertainties on distributed characteristics, such as the FAs power distribution, statistical approach is used, the so-called Total Monte-Carlo (TMC) method or generation random sampled (GRS) method.

In the United States, the code assessment described in Section 3.12 of this report implicitly captures nuclear data uncertainties since the models that use certain nuclear data are compared against experimental benchmarks or operating data.

All participants agreed that taking into account uncertainties in nuclear data plays an important role in safety analysis and use different method for this purpose.

### *3.2.12. Methodologies used both for validation and for uncertainty evaluation of neutronics code*

The participants were asked about the methodologies used both for validation and for uncertainty evaluation of neutronics code.

China, France and Italy answered that comparisons are used between mock-up experiment and target reactor (critical mass, geometry, spectral index, reaction rate distribution, etc.). China also uses:

- detailed evaluation on uncertainty quantification of sodium void worth with large uncertainty;
- conservative uncertainty in initial design, and re-evaluation of uncertainty using various core physics test in commissioning.

In Italy, some calculations concerning the similarity evaluation between mock-up experiment and target reactor have been performed for fast spectrum systems, the experimental small-scale reference system being the TAPIRO reactor of the Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA). However, these calculations are for now only targeted at the assessment of the method itself.

Korea stated that no methodologies are, as yet, fully established. All the methods mentioned above should be taken into account together. Large uncertainties will be imposed at the initial stage of operation and the values will be re-evaluated through the power ascension and nuclear tests. This requires well established nuclear test plans. It is expected that the measurement of excess reactivity, calibration of control rod worth, measurement of isothermal temperature coefficient, etc., will be performed during the low power nuclear tests.

In Russia, the methodology and regulatory requirements for verification and validation of the neutronics codes are contained in the following documents:

- "Requirements to the structure and content of a report on verification and validation of codes used for safety justification of nuclear energy use enterprises", RD-03-34-2000 (Gosatomnadzor of Russia, 2000);
- "Guidance on the verification and review of codes for neutronic calculations", RB-061-11 (Rostechnadzor, 2011).

As a measure of the uncertainty of calculation result it is recommended to use the deviation of the calculated parameters from the measured values and values calculated by more precise (e.g. Monte-Carlo code) or certificated code. The value of uncertainty can be characterised as a mean square deviation, confidence coefficient, and confidence interval or as a maximum deviation according to the expert assessment in the case when number of comparisons is not enough. It is important that in the evaluation of calculation parameters uncertainty, the uncertainty value with which a comparison is made should be considered. Evaluation of reliability on predicted uncertainty is performed by the method of mathematical statistics (statistical verification of the hypothesis about the probability distribution is Gaussian), the recommended number of tests for this check ~100 or more.

In the United States code validation and uncertainty evaluation is accomplished primarily through comparisons to benchmark experiments and operating reactor data. Start-up tests and technical specifications surveillance requirements verify the uncertainty in neutronic parameters, and the uncertainty values may be updated as operating data is collected.

In general, the answers show that rules and regulations to increase the inherent and passive safety of advanced SFRs, as well requirements for neutronic calculations, establishing and justification neutron parameter safety limits, requirements related to neutronic calculation methods and computer codes, experimental justification of neutronic characteristics, accounting and evaluation of calculation parameters uncertainties and methodologies used for validation and for uncertainty evaluation of neutronics codes are very similar to the requirements for existing reactors. The development of specific requirements for neutronics of advanced SFRs apparently is not necessary. However, regulatory guidance on specific issues may be useful (for example, on coolant void effect and potential reactivity in case of core compaction).

## 3.3. Criticality safety

### *3.3.1. Safety requirements for storage and transportation of nuclear fuel*

The participants were asked about the regulatory documents which establish nuclear safety requirements for storage and transportation of nuclear fuel.

As follows from the answers in Canada these documents are RD-327 (*Nuclear Criticality Safety)* (CNSC, 2010a), for storage and RD-364 (*Fissile Material Transportation Packages)* (CNSC, 2009), for transportation, as well as Regulations "Packaging and Transport of Nuclear Substances" (CNSC, 2015). Guidance is provided in RD-327.

In China, regulation HAF701 gives the detailed requirements on transportation of nuclear fuel. It does contain special guidance for nuclear criticality safety.

France reported that concerning civil transportation of nuclear materials in general, including spent fuel, French regulations are based on the IAEA SSR-6 specific safety requirements (IAEA, 2018). Three guides have been issued on behalf of the safety authority, dealing with requests for approval, the content of a package design safety report (PDSR) and compliance with European regulations of the packages which are classified as "not submitted to standard approval". The second guide, in particular, provides requirements and guidance on the content of the PDSR regarding criticality safety for nuclear material transportation.

General requirements for interim storage of fuel and radioactive waste are given in the "Basic Nuclear Installation" (order of 7 February 2012) (ASN, 2012). Otherwise, technical guides exist that address waste management and storage but not interim storage of spent fuel.

In Germany there is KTA 3602 "Storage and handling of fuel assemblies and associated items in nuclear power plants with light water reactors" (KTA, 2003) which is applicable to light water reactor concepts only. The regulations describe:

- dry storage and handling of new fuel assemblies;
- wet storage and handling of new and irradiated fuel assemblies.

For the specified normal operation the distance between the fuel assemblies in the storage racks shall be chosen such that the neutron multiplication factor $k_{ef}$ shall not exceed 0.95. If a conservative approach is proven, the calculation of $k_\infty$ is sufficient.

For the design-basis accidents the same rules apply, but in well substantiated cases a higher value, the maximum permissible value being $k_{ef} = 0.98$, may be used.

Italy stated that no regulatory documents specifically issued for the safety requirements for the storage and transportation of nuclear fuel are in force in Italy. For the transportation of fissile material in general, IAEA safety requirements TS-R-1 (IAEA, 2009) are usually applied; these contain guidance for criticality safety. Special laws are in force concerning this storage of irradiated or spent nuclear fuel (for instance, Decreto Legislativo 15 Febbraio 2010, n. 31) however these don't contain guidance for nuclear criticality safety. Also, for storing, handling and processing of fissile materials, ISO-1709 (ISO, 2018) is typically used.

Korea answered that the documents used for PWR evaluation will be referenced, but the kind of these documents is not indicated.

In Russia, the requirements for nuclear criticality safety for storage and transportation of nuclear fuel are established in “Safety rules for storage and transportation of nuclear fuel at nuclear facilities”, NP-061-05 (Rostechnadzor, 2005a). The special requirements to safety analysis report of storage facilities for nuclear materials are presented in “Requirements to safety analysis report of storage facilities for nuclear materials”, NP-066-05 (Rostechnadzor, 2005b).

The United States indicated that NUREG-1609, “Standard review plan for transportation packages for radioactive material,” (USNRC, 1999) and NUREG-1617, “Standard review plan for transportation packages for spent nuclear fuel,” (USNRC, 2000) contain guidance for criticality safety of transportation packages. Packages of arrayed spent fuel casks are governed by NUREG-1646, “Criticality analysis of transportation-package arrays” (USNRC, 1999b). SRP Section 9.1.1 provides guidance on criticality safety of new and spent fuel storage and handling in a nuclear power plant outside of the reactor. NUREG/CR-5661, “Recommendations for preparing the criticality safety evaluation of transportation packages” (USNRC, 1997a), NUREG/CR-6361, “Criticality benchmark guide for light-water-reactor fuel in transportation and storage packages” (USNRC, 1997b), and NUREG/CR-6698, “Guide for validation of nuclear criticality safety calculational methodology” (USNRC, 2001) address determination of biases and uncertainty in criticality analyses.

Thus, we can say that most WGSAR member countries have a fairly solid regulatory base for nuclear safety of storage and transportation of nuclear fuel.

### *3.3.2. Safety requirements for main stages of fuel handling*

This question was focused towards the nuclear criticality safety requirements at the main stages of nuclear handling, including:

- fresh fuel storage and handling;
- core reloading system, especially for frequent reloading without shutdown;
- spent fuel handling (in-core, spent fuel pool, spent fuel cleaning system);
- nuclear fuel transportation at power plant site.

According to Canada there are about 200 pages of requirements and guidance in total in RD-327 (CNSC, 2010a) and GD-327 (CNSC, 2010b) related to the topics mentioned above.

China answered that the code of the regulation, HAF103/01-1994, gives the detailed regulation on transportation of nuclear fuel.

France stated that the requirements for fresh fuel storage and handling as well for nuclear fuel transportation at power plant site were to be completed in the future. With regard to core reloading system it is reported that fuel reloading is authorised only during shutdown states.

Russia stated that the regulatory documents NP-061-05 (Rostechnadzor, 2005a) and NP-066-05 (Rostechnadzor, 2005b) give the detailed regulation on storage and transportation of fresh and spent nuclear fuel. The frequent reloading without shutdown in the Russian SFRs is not available and not planned.

Finally, the United States has general criticality requirements for the fuel storage, handling, fuel loading and spent fuel storage. Annex A to ten CFR Part 50 (USNRC, 2007), GDC 61, “Fuel storage and handling and reactivity control,” requires, in part, that fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. GDC 62, “Prevention of criticality in fuel storage and handling,” requires the prevention of a criticality accident by physical systems or processes, preferably using geometrically safe configurations. GDC 63, “Monitoring fuel and waste storage,” requires, in part, the provision of systems in fuel storage systems and associated handling areas to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and to initiate appropriate safety actions.

Furthermore, ten CFR 70.24 (USNRC, 2018c) requires a licensee to maintain an alarm system capable of detecting a criticality accident. Ten CFR 50.68 (USNRC, 2018b) aims to prevent a criticality accident or mitigate the consequences of an accident. A nuclear power reactor applicant or licensee may choose to comply with Part (b) of ten CFR 50.68, which provides specific limits on the effective multiplication factor ($k_{eff}$) for the new and spent fuel storage racks, in lieu of ten CFR 70.24.

Guidance on complying with the above requirements in the various stages of fuel handling are provided in multiple SRP sections, as follows: new and spent fuel storage and handling – Section 9.1.1; spent fuel pool clean-up system – Section 9.1.3; refuelling – Section 9.1.4.

As shown by the answers, no special nuclear criticality safety requirements for advanced SFRs currently exist. Only France reported on the development of such requirements for SFRs at the stages of fresh fuel storage and nuclear fuel transportation which have to be completed. Other participants use the requirements developed for existing reactors.

### *3.3.3. Requirements regarding codes and methods*

A question focused on the requirements regarding codes and methods used for nuclear safety justification at all stages of fuel handling.

Canada answered that codes and methods used for nuclear safety are to be compliant with requirements of Section two of RD-327 (CNSC, 2010a).

In China, there are no detailed requirements regarding codes and methods on the fuel handling at all stages at present.

France reported that development of such requirements has yet to be completed.

In Germany, there is standard DIN 25478:2014 “Application of computer codes for the assessment of criticality safety” (German Institute for Standardisation, 2014) according to which:

- The computer code needs to be able to describe the parameters necessary for the neutronics of the arrangement of fissile material in question and the physical effects that characterise the arrangement.
- It further needs to be able to determine the parameters relevant to safety of the arrangement as well as the measured data that is necessary to validate the former.
- The computer code needs to be validated for each application by comparing the results of a calculated benchmark case. The recommended source for the benchmark data is the “International Handbook of Evaluated Criticality Safety Benchmark Experiments” (NEA, 2016).

Further requirements are made in standard DIN 25471:2009 “Criticality safety taking into account the burn up of fuel elements when handling and storing nuclear fuel elements in fuel pools of nuclear power plants with light water reactors” (German Institute for Standardisation, 2009).

Russia reported that according to the 7.5.1 NP-066-5 (Rostechnadzor, 2005c) the safety report must contain a description of methods and codes used with information about their certification and nuclear data library. In practice the Monte-Carlo codes are used for nuclear safety analysis at all stages of fuel handling.

In the United States, no specific requirements on codes and methods for fuel handling exist. However, guidance on codes and methods is provided in the several documents listed in Section 4.1 of this report.

As in the previous paragraph, no specific requirements for SFRs were noted in the answers of participants.

### *3.3.4. Initiating events in fuel handling and transportation safety analysis*

The participants were asked about the postulated initiating events considered in fuel handling and transportation safety analysis reviews for advanced SFRs.

Canada answered that the postulated initiating events considered in fuel handling and on-site transportation safety analysis are any events or event sequences which are considered credible, i.e. the frequency of occurrence is once in a million or higher. The text of one of the requirements states that “…before a new operation with fissionable material is begun, or before an existing operation is changed, it shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions” (CNSC, 2019).

China as a source of initiating events indicated the disastrous weather including heavy rain, penetrating by external force, falling from high altitude.

In France, for the safety assessment of handling phase, the following events were considered in the analysis of past reactors:

- fissile sub-assembly drop off in the handling corridor and in the carrying case;
- carrying case drop out of protected zones;
- inadvertent opening of the isolation valve of the fuel pit (making a connection between the primary contaminated argon and some work spaces);
- leak of the carrying case (filled with sodium);
- sub-assembly or carrying case blockage in the handling chain.

It is noted that French SFRs were equipped with an interim storage vessel.

Russia presented the recommended list of initiating events of DBAs and BDBAs during storage and handling of fuel assemblies, include:

***Design-basis accidents***

1. Internal and external impacts of natural and anthropogenic origin.
2. Blackout of the object using atomic energy.
3. The fire in the storage of nuclear fuel and/or at the transportation of nuclear material.
4. Dropped items, which can change the spacing of fuel assemblies and fuel pins to violation of the integrity of fuel claddings and FA.

5. Dropping of individual FA, boxes and covers with spent FA with transport-technological operations.
6. Possible leaks from the spent nuclear fuel pool.
7. The impact of flying objects, generated as the result of accidents (for example, as a result of destruction of the systems operating under pressure).
8. Failure of ventilation leading to the formation of explosive mixtures in the storage of spent nuclear fuel.
9. Violation (Interruption) of heat removal during storage and transportation of nuclear fuel.
10. Violation of fastening of packages during transportation of nuclear fuel.

***Beyond-design-basis accidents***

1. The occurrence of self-sustaining chain reaction at storage and handling of nuclear fuel.
2. Complete dehydration of the repository for spent nuclear fuel.
3. Accidental dropping of equipment and building structures overlapping the storage compartments or stored nuclear fuel.
4. Flooding of nuclear fuel storage.

The United States stated that for the Clinch River Breeder reactor, two fuel handling accidents were proposed and analysed: a non-mechanistic cover gas release accident, and a non-mechanistic EVTM (ex-vessel transfer machine) accident. For both accidents, it was assumed that the refuelling hatch connecting the reactor service building and the reactor containment building remained open. Furthermore, SRP Sections 9.1.1 and 9.1.4 provide guidance on evaluation of fuel handling accidents.

Safety assessment of handling and transportation phases of each participant is based on their own experience in the operation existing reactors, their own regulatory frameworks, taking into account the specific of the sodium-cooled reactor such as leak of the carrying case filled with sodium.

### *3.3.5. Subcriticality control method and means*

A question focused on the requirements regarding subcriticality control methods and means (e.g. criticality accident alarm system) for advanced SFRs. Only Canada and the United States provided some information about requirements regarding subcriticality control.

Canada reported that full scope requirements regarding criticality safety are listed in Sections two, and four – to 16 of RD-327 (CNSC, 2010a). Requirements on criticality accident alarm system are set by Section three of RD-327: "Subject to the evaluation of the overall risk described above, a criticality alarm system meeting the requirements of this regulatory document shall be installed in areas where: *(i)* inadvertent criticality can occur, and *(ii)* excessive radiation dose to personnel is credible should the inadvertent criticality occur."

In the United States, ARDC 26 addresses reactivity control systems in the reactor. As discussed in Section 4.2 of this report, ten CFR 50.68 (USNRC, 2018b) and ten CFR 70.24 (USNRC, 2018c) prescribe requirements related to alarm systems capable of detecting a criticality accident. RG 1.97 (USNRC, 2006) provides guidance on accident monitoring instrumentation.

Italy declared that nothing specific exists for advanced SFRs; more generally, reference is typically made to ISO-7753 (ISO, 1987) for criticality detection and criticality accident alarm systems.

Other participants reported that there are no specific requirements for advanced SFRs and the reference can be drawn from requirements on conventional power reactors.

### *3.3.6. Experimental data to validate codes and methods*

This question focused on the requirements regarding the use of experimental data to validate codes and methods used in fuel handling nuclear safety analyses.

Canada reported that requirements regarding the use of experimental data to validate codes and methods used in fuel handling nuclear safety analyses are set in Section 2.3.4 of RD-327 (CNSC, 2010a) and state that "Bias shall be established by correlating the results of critical and exponential experiments with results obtained for these same systems by the calculation method being validated." When no experimental data are available, establishment of the bias for a calculation method is not possible and the requirements of this section cannot be satisfied. Validation of a calculation method by comparing the results with those of another calculation method, for example, is unacceptable.

In China and France, there are no detailed requirements on this point at present. According to these participants the reference can be drawn from requirements on conventional power reactors.

Russia stated that general guidelines on the verification neutronic codes, including using the experimental data, are contained in the regulatory documents NP-061-05 (Rostechnadzor, 2005a) and NP-066-05 (Rostechnadzor, 2005b).

In the United States, no regulations exist for the validation of criticality analysis codes and methods. However, relevant guidance is provided in NUREG/CR-5661 (USNRC, 1997a), NUREG/CR-6361 (USNRC, 1997b), and NUREG/CR-6698 (USNRC, 2001), as discussed in Section 4.1 of this report. In general, criticality codes are validated by comparing benchmark experiment data against calculated results for those experiments. Trending analyses should be performed to determine any dependence of the code bias on model parameters, such as fuel enrichment and poison content. The resulting code bias and bias uncertainty is applied to criticality calculations to ensure conservative results.

As in the previous section on the basis of the received answers we can conclude that the development of specific requirements for criticality safety of advanced SFRs are not required. Specific features of these reactors (for example increased content of fissile isotopes in the fuel) can be taken into account in the regulatory framework.

## 3.4. Phenomena

### *3.4.1. Negative reactivity shutdown*

The participants were asked about provisions in core designs to prevent "negative reactivity shutdown" (for instance negative reactivity shutdowns which occurred at Phénix in 1989 and 1990, the causes of the decreases in reactivity that occurred at the EBR-II in 1974 and at Rapsodie in 1978). France answered that this issue remains open as negative shutdown incidents have never been explained. At present stage of ASTRID development, several design provisions are deemed favourable by the applicant to master this risk. Analysis of these provisions is yet to be done.

According to Korea this is a serious safety issue for SFR licensing and have reviewed the report issued by Joël Guidez, "PHÉNIX: The Experience Feedback", 2013 (Joël Guidez, 2013). The following shows the power history at the time AURN occurred in Phénix. Korea expert understanding is that it is not really a matter of “negative reactivity”. If the control rods had not been inserted quickly, the power could have become positive again (power oscillation). Korean experts suppose the failure probability of control rod insertion for SFR might be rather high. The Section 4.3 of the safety review guideline for SFR, under development, requires reviewing the following in detail:

- sensitivity of fast reactors to a reactivity change in the event of sub-assembly movement;
- protection against positive changes using anti-compaction systems;
- better understanding of the phenomena of gas flow in a reactor and related protection;
- better monitoring of core movements;
- care in setting up irradiation devices in areas where the hydraulic, thermal and neutron fluxes are not well known, not well calculated and not well monitored.

Russian experts participated in the international group investigating the causes of the incident at Phénix, but the exact cause of the negative reactivity shutdowns at Phénix have not been established.

### *3.4.2. Problem of the sodium aerosol deposits in the cold section of the control rod mechanism*

Operating experience feedback shows that the control rod drop times or rod drop failure due to sodium aerosol deposits in the cold sections of the rod mechanism could be a safety issue. The participants were asked how they consider this risk in regulation or in design. Most of the participants did not answer this question.

France states that a priori, the risk of control rod jamming due to aerosols was deemed negligible for the latest French design. Nevertheless, at the stage of the safety report, this issue shall be examined and specific operating procedures might be required to ensure the correct performance of the control rod mechanisms.

Korean experts expect this is also an important safety issue for SFR. As a requirement, it is necessary in a very general way that the design shall minimise the impact of sodium aerosol effect. The experimental data and the computer code analyses should be able to support the design.

Russian experts reported that they have not encountered that problem in existing SFRs.

All participants agreed that it would be useful to get more information and discuss the operating experience related to problem of the sodium aerosol deposits in the cold section of the control rod mechanism and impact of the deposition of sodium aerosols on the operability of small and large rotating plugs.

### *3.4.3. Initial core physics test plans*

Participants were asked about the initial core physics test plans and how to evaluate the validity of the plans, including:

- control rod worth confirmation;
- excess reactivity measurement;
- reactivity shutdown margin measurement;

- isothermal temperature coefficient evaluation;
- feedback reactivity evaluation;
- applicability evaluation on subcriticality measurement, etc.

From the above physical tests, China and Russia indicated: control rod worth confirmation, isothermal temperature coefficient evaluation, feedback reactivity evaluation and applicability evaluation on subcriticality measurement.

France considered the following tests: measurement of the integral worth and detailed differential curve; measurement of power distribution; reactivity measurement at each sub-assembly loaded and validation of subcriticality in case of error in sub-assembly loading.

Korea reported that the initial core physics test plans for PGSFR will be submitted in the future when an official licensing review starts. Korea has summarised tests from the Joyo and Monju test plans but are still concerned whether these tests are enough to validate the nuclear design of SFR with large uncertainties.

In the United States, physics testing performed as part of plant startups addresses the above. Guidance in SRP Section 14.2 (USNRC, 2007) provides a framework for reviewing proposed initial start-up tests.

# 4. Conclusions

The analyses undertaken by the NEA Working Group on the Safety of Advanced Reactors (WGSAR) and the countries participating in the questionnaire exercise have led to some important conclusions. For example, most participants agreed that, over the past decades, analytical tools have become increasingly sophisticated and more accurate in predicting sodium fast reactor (SFR) behaviour. New technologies that use materials with an appropriate database (i.e. irradiation exposure, energy spectrum and material temperatures) have been proposed to the regulatory bodies. Advanced reactor initiatives and collaboration among the involved parties will also help to achieve goals (reliable and safe SFRs) more efficiently than has been the case in the 20$^{th}$ century.

It was noted that countries participating in the WGSAR have many experimental facilities for materials science including irradiation test facilities designed to evaluate reactor fuels and structural materials, test facilities for investigations of liquid metal technologies and thermo-hydraulics relevant for SFR, for sodium-concrete, sodium-water and fuel coolant interaction investigations, modelling the behaviour of fuel under SFR accident conditions. SFR neutronics test installations exist now in China (SEFR), France (MAZURKA) and Russia (BFS-1, BFS-2, BOR-60, BN-600 and BN-800) only. However, a significant amount of experimental data on neutronics of SFRs has been obtained in previous decades during the development of this reactor technology. In particular, the NEA *International Handbook of Evaluated Reactor Physics Benchmark Experiments* (NEA, 2016) contains several fast reactor benchmarks.

The WGSAR members agreed that the international co-operation in neutron experiments for advanced SFRs on the basis of bilateral or multi-lateral agreements between countries is advisable and effective. In Europe, multi-lateral research co-operation is achieved through Euratom projects such as the ESFR-SMART and SESAME projects of the Horizon 2020 EU Framework Programme for Research and Innovation.

The answers to the questionnaire revealed that the existing regulatory framework is general enough to be applicable to SFRs. Many existing neutron characteristic requirements are of sufficiently high level to be applicable universally. However, there are differences, and to capture those differences, specific regulatory documents could be developed, such as RG-1.232 in the United States. Specific features of these reactors can also be taken into account in the regulatory framework.

The answers also showed the similarity in application at the plant's safety analyses of limiting neutronic parameters that bound the best-estimate plus uncertainty values. Limiting neutronic parameters are verified through surveillance as part of a facility's technical specifications. The requirements on these limits are captured by design criteria (e.g. general design criteria [GDC] and advanced reactor design criteria [ARDC]) associated with reactor inherent protection and fission product barriers. The requirements for establishing and justifying neutron parameter safety limits for advanced SFRs are similar to existing water-cooled reactor requirements.

Also, all participants agreed on the need to increase the use of inherent and passive safety with regard to the core design of advanced reactors including SFRs. This requirement is contained in the existing regulatory documents of WGSAR participating countries and applies to all types of reactors.

Another similarity between SFRs and existing conventional power reactors requirements is notable for the requirements related to neutronic calculation methods, computer codes and group constants used for advanced SFRs, with the exception of the requirement to use multi-group cross-sections in the calculation of SFRs instead of the few-group (usually two groups) which are commonly used in the Pressurised Water Reactor (PWR) calculations.

The list of accident scenarios related to reactivity coefficients for all participants is approximately the same. This list includes design-basis accident (DBAs) caused by unauthorised movement of the rods, the penetration of gas bubbles or hydrogen-containing substances into the core, the deterioration of the heat removal and beyond-design-basis accidents (BDBAs) such as UTOP, ULOF, ULOHS without scram (ATWS). In all of these scenarios, the various reactivity feedbacks (Doppler, thermal expansion of core components, coolant density variation, neutron leakages) play a fundamental role in the evolution of the accident itself.

All participants agreed that accounting and evaluation of calculation parameter uncertainties are important parts of the safety analysis, and the uncertainty analysis must address all important sources of code, nuclear data and input data uncertainty. The methodology for validating and evaluating neutron code uncertainties for SFRs is the same as the methodology for other reactors. It includes similarity evaluations between the mock-up experiment and target reactor (critical mass, spectral index, reaction rate distribution, etc.). Conservative uncertainty should be included in the initial design, based on comparisons to benchmark experiments and operating reactor data, and uncertainty should be re-evaluated using various core physics tests in commissioning. Start-up tests and technical specifications surveillance requirements should be provided to verify the uncertainty in neutronics parameters, and the uncertainty values may be updated as operating data is collected.

There is common understanding, that the requirements regarding experimental justification of the SFR neutronic characteristics apply primarily to validation and verification of computer codes and cross-section libraries used in SFR design applications. Empirical evidence is also needed to establish nuclear reliability factors, which are placed on calculated neutronic parameters (e.g. reactivity balance, power distribution, shutdown margin) to account for uncertainty.

As for the criticality codes, they are validated by comparing benchmark experiment data against calculated results for those experiments. Trending analyses should be performed to determine any dependence of the code bias on model parameters, such as fuel enrichment and poison content. The resulting code bias and bias uncertainty are applied to criticality calculations to ensure conservative results.

The standard list (similar to other types of reactors) for initial start-up tests includes control rod worth confirmation (integral and detailed differential curve), isothermal temperature coefficients evaluation, feedback reactivity evaluation, measurement of power distribution, and reactivity measurements at each sub-assembly loaded.

The questionnaire answers also revealed the areas in which there were significant variations in opinion of WGSAR participants. For example, there is no uniform approach to requirements related to calculated neutronic parameters. In some countries, there are no explicit requirements to neutronic parameters in the regulation levels, except for the overall safety goals. In one case, regulatory requirements are provided on the sign and value of several characteristics identified in the previous paragraph. Some requirements are established as design criteria for SFR projects.

Criticality safety assessments in each WGSAR participating country are based on the country's own experience in the operation of existing reactors, and its own regulatory framework. All participants have a fairly solid regulatory base for nuclear safety in terms of the storage and transportation of nuclear fuel, which is also sufficient for SFRs. There are no specific regulatory requirements regarding the codes and methods for the criticality safety assessments of advanced SFRs.

Finally, it was revealed that there are no specific requirements regarding subcriticality control methods and means (e.g. criticality accident alarm systems) for advanced SFRs, and the reference can be drawn from requirements on existing water-cooled power reactors.

Moreover, the answers provided by participants identified the reliability of shutdown mechanisms as one topic worthy of further discussion. This issue is an area where WGSAR members believe that additional work is needed, given the potential for common mode failure. Such work would include understanding the impact of phenomena such as sodium aerosol deposition, tribology issues, which could, for example, prevent control rod insertion.

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