Nuclear Safety
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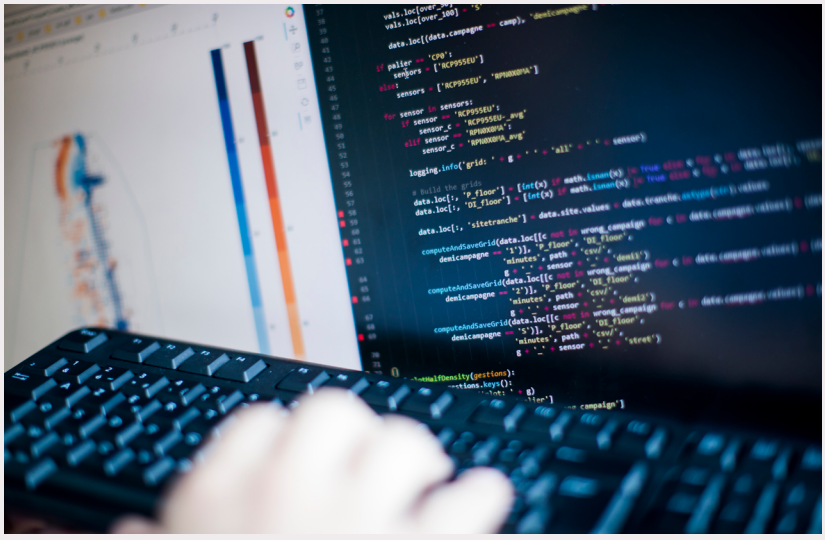

Regulatory Perspectives on Analytical Codes and Methods for Advanced Reactors

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Nuclear Safety

Regulatory Perspectives on Analytical Codes and Methods for Advanced Reactors

NEA N° 7617

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The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for the safe, environmentally sound and economical use of nuclear energy for peaceful purposes.
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Specific areas of NEA competence include the safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

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Foreword

The Nuclear Energy Agency (NEA) Working Group on the Safety of Advanced Reactors (WGSAR) has been focused on the regulatory activities in the primary area of advanced reactors and associated installations. The term "advanced reactors" in the context of this working group refers primarily to non-light-water innovative reactor designs such as those proposed by the Generation IV International Forum (GIF), including small modular non-light-water reactors.

The WGSAR aims provide regulatory perspectives on the areas in which additional or revised regulatory framework and licensing approaches, including safety research, may be needed to facilitate effective regulation of advanced reactors and to develop common understanding and approaches. This activity is pursued under the NEA Committee on Nuclear Regulatory Activities (CNRA).

This report aims to (1) identify and clarify the requirements and best practices applicable to nuclear power plant designers for the development and use of analytical codes and methods used in the design and safety analysis of nuclear power plants, and (2) identify best practices for the use of confirmatory analyses by regulatory agencies.

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This report is based on the statements and opinions provided by members of the NEA Task Group on Analytical Codes and Methods (TGACM), which is a subgroup of the NEA Working Group on the Safety of Advanced Reactors (WGSAR) and experts from their respective organisations. The TGACM members are:

- Bajorek, Stephen (Nuclear Regulatory Commission [NRC], United States);
- Drzewiecki, Timothy (NRC, United States);
- Grasso, Giacomo (Italian National Agency for New Technologies, Energy and Sustainable Economic Development [ENEA], Italy);
- Herer, Christophe (Institut de radioprotection et de sûreté nucléaire [IRSN], France);
- Ivanov, Evgeny (IRSN, France);
- Khrennikov, Nikolay (Scientific and Engineering Centre for Nuclear and Radiation Safety [SEC NRS], Russia);
- Klein-Heßling, Walter (Gesellschaft für Anlagen- und Reaktorsicherheit [GRS], Germany);
- Pivovarov, Valerii (SEC NRS, Russia);
- Plummer, David (Office of Nuclear Regulation [ONR], United Kingdom).

Organisations from Canada, France, Germany, Italy, the Russian Federation, the United Kingdom and the United States contributed to the survey.

Chapter 3 is based on contributions from the Generation IV International Forum (GIF).

The WGSAR members.

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List of abbreviations and acronyms

ADS	Automatic depressurisation system
BNL	Brookhaven National Laboratory (United States)
BWR	Boiling water reactor
CFD	Computer fluid dynamics
CHF	Critical heat flux
CNRA	Committee on Nuclear Regulatory Activities (NEA)
FHR	Fast high-temperature reactor
FR	Fast reactor
GFR	Gas-cooled fast reactor
GIF	Generation IV International Forum
Hi2Lo	High fidelity to inform low fidelity modelling
HTC	Heat transfer correlation
[V]HTR	(Very) High-temperature reactor
IAEA	International Atomic Energy Agency
IRSN	Institut de radioprotection et de sûreté nucléaire (France)
LBE	Lead-bismuth eutectic
LFR	Lead-cooled fast reactor
LOCA	Loss of coolant accident
LWR	Light water reactor
MOX	Mixed oxide
MSR	Molten salt reactor
ND	Nuclear data
NEA	Nuclear Energy Agency
NRC	Nuclear Regulatory Commission (United States)
OECD	Organisation for Economic Co-operation and Development
ONR	Office of Nuclear Regulation (United Kingdom)
ORNL	Oak Ridge National Laboratory (United States)
PWR	Pressurised water reactor
QA	Quality assessment
RCCS	Reactor cavity cooling system

SCC	Stress corrosion cracking
SCWR	Super-critical water-cooled reactor
SEC NRS	Scientific and Engineering Centre for Nuclear and Radiation Safety (Russia)
SFR	Sodium-cooled fast reactor
SMR	Small modular reactor (or small medium reactor)
TAREF	Task Group on Advanced Reactor Experimental Facilities (NEA)
TGACM	Task Group on Analytical Codes and Methods (NEA)
TRISO	Tristructural-isotropic
UK	United Kingdom
ULOF	Unprotected loss of flow
ULOHS	Unprotected loss of heat sink
US	United States
UTOP	Unprotected total loss of power
VHTR	Very high-temperature reactor
WCR	Water-cooled reactor
WGSAR	Working Group on the Safety of Advanced Reactors (NEA)

Chapter 1. Introduction

Analytical codes and methods are used extensively in the design and safety analysis of nuclear reactors. These codes and methods are commonly used to analyse the response of a complex engineering system to postulated events with potentially severe health, financial and environmental implications. Guidance for deterministic analyses is provided in an IAEA report (IAEA, 2019b) where the use of computer codes for deterministic safety analysis is presented. Regulatory agencies establish requirements for the nuclear power plant designer for the development and use of analytical codes and methods in order to ensure the quality and credibility of the analyses produced by the analytical codes and methods. In addition, regulatory agencies have used analytical codes and methods to perform confirmatory analyses as part of due diligence during a regulatory review. These confirmatory analyses frequently require specialised expertise and can be a resource-intensive aspect of a regulatory review. The purpose of this report is to (1) identify and clarify the requirements and best practices applicable to nuclear power plant designers for the development and use of analytical codes and methods used in the design and safety analysis of nuclear power plants, and (2) identify best practices for the use of confirmatory analyses by regulatory agencies.

This technical report has been developed by the NEA Working Group on the Safety of Advanced Reactors (WGSAR). Analytical codes and methods were identified by the WGSAR as a topic for further investigation and a questionnaire was created to gather information from the participants. This report is based on the answers to this questionnaire received in September 2021. The respondents to the survey came from Canada, France, Germany, Italy, Russia, the United Kingdom and the United States.

The scope of this report covers the regulatory requirements imposed by regulatory authorities upon reactor designers and operators that use analytical codes and methods for design and analysis; the use of analytical codes and methods by regulatory authorities and/or their technical support organisations as a confirmatory analysis tool; and the existing analytical codes and methods. The regulatory requirements describe general requirements (e.g. laws/regulations and enforcement), regulatory oversight and guidance, code assessment requirements and/or expectations, and user qualifications. The confirmatory analysis discussion describes the reasons for performing confirmatory analyses, expectations for the analytical tools, and quality assurance associated with those analytical tools. Chapter 2 of this report presents the common positions identified in the survey responses, Chapter 3 provides an overview of Generation IV reactor concepts with a focus on relevant phenomena, and Chapter 4 discusses the survey and survey responses received from the WGSAR members. Additionally, Appendix A includes the survey responses from WGSAR members and lists the major codes and methods used in those NEA member countries.

Chapter 2. Common positions

2.1 Regulatory requirements

This section presents the common positions on the requirements and best practices applicable to nuclear power plant designers for the development and use of analytical codes and methods in the design and safety analysis of nuclear power plants. These common positions were established by participating WGSAR members on the basis of the questionnaire answers detailed in Section 4.1.

1. There is a general consensus that a code approval/certificate shall specify purpose and scope of the approval, range of applicability, and any applicable restrictions on the use of the code.
2. Analytical codes and methods used in the design and safety analysis for a nuclear facility should go through a process of verification and validation to establish the necessary confidence in the codes ability to determine safety margins.
3. Analytical codes and methods should be developed and maintained under a quality assurance programme. This quality assurance programme should include a comparison of code predictions against experimental data, verification of the implementation of numerical methods, and code lifecycle management.
4. Bugs and errors in analytical codes and methods should be managed using a combination of code quality assurance procedures, verification and validation, and in some cases, sensitivity analyses and/or confirmatory analyses. The comparison of code results against experimental data is seen as an important component in code validation and the elimination of defects in the code.
5. For legacy codes that were not originally developed under a quality assurance programme, the user of the code should demonstrate that the quality of code is commensurate with current good practice (i.e. verification and validation).
6. Code calculational uncertainty should be determined using statistical methods to determine the calculation error at high confidence levels (either determined by parametric [e.g. 2 sigma] or non-parametric [e.g. Wilks] methods) when sufficient data are available. There is general recognition that sufficient data may not exist to obtain statistical values with confidence levels and a conservative approach (e.g. maximum deviation) can be used.
7. Application of a code to a design calculation or safety analysis should include appropriate conservatisms to accommodate uncertainties and unknown or non-modelled phenomena, and, when applicable, insufficient qualification of the code.
8. Uncertainty assessment for passive systems is conceptually similar to active systems although the range of parameters can be very different.
9. Code users or the organisations to which the users belong should be qualified. It is recognised that the used procedures are country-specific.

2.2 Confirmatory analyses

Confirmatory analyses consist in cross calculations carried out by the reviewer on the same transient, using the same or different tools.

This section presents the common positions on the best practices for the use of confirmatory analyses by regulatory agencies. These common positions were established by participating WGSAR members on the basis of the questionnaire answers detailed in Section 4.2.

1. Use of analytical codes and methods by a regulatory authority or their technical support organisation enhances the safety review by (1) increasing the regulatory authorities' understanding of the plant design, integrated system performance, safety margins, sensitivities, and analytical methods; (2) allowing for an efficient assessment of accident scenarios not presented in the licensee safety case; and (3) supporting the analysis of the safety case.
2. Regulatory decisions are based upon analyses and data submitted by the applicant and not confirmatory analyses. Issues identified through confirmatory analyses should be brought to the attention of the design authority (i.e. plant designer or facility owner) and resolved through the safety review process.

Chapter 3. Technology overview

3.1 Introduction to the reactor concepts considered by GIF

This report on the regulatory perspectives on analysis codes and methods covers the six Generation IV reactor concepts selected by the Generation IV International Forum (GIF) in the GIF Technology Roadmap:

- gas-cooled fast reactor (GFR);
- lead-cooled fast reactor (LFR);
- molten salt reactor (MSR);
- sodium-cooled fast reactor (SFR);
- supercritical-water-cooled reactor (SCWR);
- very-high-temperature reactor (VHTR).

The selection of these reactor concepts is based on the proposed objectives for the next ("fourth") generation of nuclear systems, namely sustainability, safety and reliability, economic competitiveness, proliferation resistance and physical protection. Worldwide, numerous research projects are ongoing to develop these six concepts. However, the status of development of the different concepts is quite different. This is also the case for the required specific analytical codes and methods. Therefore, the possible existing knowledge gaps for the different codes is not an indication of the capabilities and safety of the different reactor concepts themselves.

In the following section, a brief description of the six reactor concepts is provided. The information is mainly based on the GIF Roadmap (GIF, 2018). A first view of the safety specific aspects and specific needs of the analytical codes and methods is given. Further details are given in Section 3.2.

3.1.1 *Sodium-cooled fast reactor (SFR)*

In SFRs, liquid sodium is used as coolant, allowing reactors designs with a fast neutron spectrum and a high-power density without coolant pressurisation. Due to the high boiling point of sodium and its high thermal conductivity, core outlet temperatures of 500-550°C can be reached. Plant size options under consideration range from small, 50 to 300 MW_e modular reactors to larger plants up to 1 500 MW_e.

SFR designs can be categorised into two types:

- "Pool-type" reactors (such as Phénix, Superphénix, PFR and CEFR) where the primary circuit components (i.e. pumps and heat exchangers) are all contained in the primary vessel, or
- "Loop-type" reactors (such as Joyo and Monju) where the primary sodium flows in loops connecting a main vessel with other vessels where the large components are located (Couturier, 2012).

Important safety features of SFRs include a long thermal response time, a reasonable margin to coolant boiling, a primary system that operates near atmospheric pressure, and an intermediate sodium system between the radioactive sodium in the primary system and the power conversion system. The sodium reacts chemically with air and water and thus requires a sealed coolant system. Water/steam is generally used as working fluid in the power conversion system but nitrogen and supercritical carbon dioxide (CO_2) are also considered for future applications to rule out the risk of sodium/water reactions in steam generators and achieve high thermal efficiency (supercritical CO_2).

Much of the basic technology for SFRs has been established in former fast reactor programmes (France, Japan, Russia, the United Kingdom and the United States). As for recent achievements, the Chinese Experimental Fast Reactor (CEFR) reached full power in 2014 (Russian design). In Russia, the BN-800 achieved full power in 2016. As for new construction plans, the People's Republic of China is building a second CFR-600 (IAEA, 2025), fuel loading of the first having taken place in 2023 (IAEA, 2024), Russia is planning the construction of the BN-1200 (World Nuclear News, 2024; Figure 1), and India began loading fuel to operate its prototype fast breeder reactor (PFBR) in 2024. A Korean design, the Prototype Generation IV sodium-cooled fast reactor (PGSFR) was also under consideration (Yoo, 2016), but was initially abandoned when Korea decided to progressively phase out nuclear power – a decision that has since been rescinded. Supported by the United States in connection with GNEP/IFNEC and a Korean prototype Generation IV sodium-cooled fast reactor (PGSFR) is now planned for 2028 (World Nuclear Association, 2024).

Figure 1: **BN-1200 Reactor**

Source: Maroval, 2017.

To evaluate the safety of SFR concepts, simulation codes have to be able to model specific phenomena:

- Regarding power control, it should be noted that in large cores with oxide fuel there is a risk of significant mechanical energy release due to sodium voiding in different states, including an accidental transition phase. This phenomenon, known as the "sodium void effect" can also occur in the event of gas bubbles being entrained into and through the core (Couturier, 2012). Design studies have been performed to investigate options for reducing sodium void worth (Khalil and Hill, 1991; CEA, 2012). However, the potential for an SFR design to have a positive coefficient remains a safety concern and needs to be assessed as part of a safety review.
- Numerical models have been developed to simulate fuel behaviour, effects of fast neutron flux on cladding, thermo-mechanical loadings and interaction between fuel and cladding (chemical and mechanical) in normal operation and during accidents. These models address several phenomena specific to SFRs like clad and fuel swelling under irradiation,

gaseous products' behaviour and migration of fissile elements in the fuel, and the thermo-mechanical behaviour of the pellets (ceramics) or slugs (metallic fuel). For some fuels, chemical interaction with sodium has to be taken into account in case of clad rupture.

- In SFR cores, heat is essentially evacuated through vertical sodium flows cooling each assembly separately. Radial heat exchange between assemblies through the hexagonal wrapper tubes is almost insignificant in normal operation. SFRs are thus sensitive to coolant flow local blockage in assemblies. It is important to model local assembly faults and the potential propagation to neighbouring assemblies and even an escalation into large core meltdown (multi-physics phenomenon).
- Severe accidents in SFRs may result in various scenarios depending on reactor design. A specific issue is the risk of neutronic criticality due to the potential core compaction before or during the accident phase. Then, typical to SFRs is the core disruptive accident in which high nuclear energy release (prompt criticality) and possible thermo-chemical reaction (sodium-fuel interaction) lead to an explosion-like scenario inside the reactor vessel. The preliminary phase of severe accident and core disruptive accident are multi-physics-driven sequences with three-dimensional aspects.
- In case of a core meltdown accident (with or without disruption phase), the safety option is the in-vessel confinement of the corium. Modelling of corium motion and interactions with sodium and structural materials is requested. The efficiency of the core catcher has to be proven based on computational models.
- Radioactive source term in case of severe accident in SFRs requires specific models due to fission products' interaction with sodium (a high level of retention of iodine and caesium are often claimed) and the potential for contaminated sodium fire.
- With regards to external hazards, the possible consequences of earthquakes and flooding must be assessed.
- Risks specific to SFRs are mainly due to sodium chemical reactivity with air and water, which have to be modelled. The risk of release of toxic chemical aerosols produced in the event of sodium fire must be taken into account. Furthermore, sodium reacts with concrete, leading to its ablation associated with hydrogen production (desorption of water from the concrete and sodium-water reaction).

3.1.2 *(Very) High-temperature reactor ([V]HTR)*

The VHTR is the next generation in the development of high-temperature reactors with TRISO-coated fuel particles, using graphite as a moderator and helium gas as a coolant. The VHTR is of interest for high efficiency power generation and for co-generation of electricity and process heat, e.g. for hydrogen production or providing process heat for industry.

The original target for the GIF VHTR outlet temperature was set at 900-1 000°C. The main driver for this technology was initially large-scale bulk hydrogen production with the iodine-sulphur (I-S) process. In the meantime, market studies in several GIF signatory countries have confirmed the existence and development potential of a significant near-term market for lower-temperature applications using process steam below 600°C. This would need less stringent reactor outlet temperatures of approximately 750°C.

The HTR technology is based on the TRISO-coated fuel particle. For this type of fuel UO_2 or UCO fuel kernels are coated by a buffer layer of porous carbon, an inner layer of high-density pyrolytic carbon (IPyC), a layer of silicon carbide (SiC) and an outer layer of high-density pyrolytic carbon (OPyC). These fuel particles are then inserted in a graphite matrix of a fuel pebble or a cylindrical fuel compact.

VHTRs have demonstrated inherent safety features in a number of past and current prototypes and engineering-scale tests, and all modern concepts are designed to remove decay heat passively. Modular construction and the elimination of active safety systems are expected to result in low equipment, operation and maintenance costs.

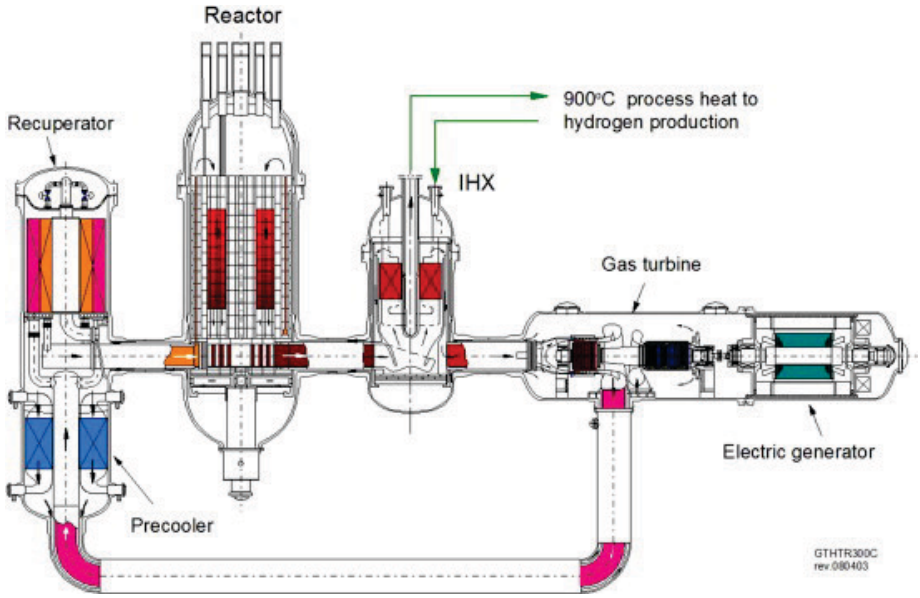
[V]HTR designs can be categorised into two types:

- the pebble bed type; and
- the prismatic block type.

The basic technology for the VHTR was established in former high-temperature gas reactors such as the US Peach Bottom and Fort St Vrain plants, as well as the German Arbeitsgemeinschaft Versuchsreaktor (AVR) and THTR prototypes. The technology has benefitted from many R&D projects internationally, as well as from former and ongoing projects, such as the high-temperature reactor-pebble bed module which began commercial operation in 2023 (HTR-PM, China), the gas turbine high-temperature reactor (GTHTR300C, Japan), the A-HTR-100 SMR based on the Pebble Bed Modular Reactor (PBMR) research initiative (South Africa), ANTARES (France, United States), the nuclear hydrogen development and demonstration (NHDD, Korea) project, the gas turbine modular helium reactor (GT-MHR, Russia, United States) and the next-generation nuclear plant (NGNP, United States), led by several plant vendors and national laboratories.

Experimental reactors, such as the high-temperature engineering test reactor (HTTR) (Japan, 30 MW_{th}) and HTR-10 (China, 10 MW_{th}), support this advanced reactor concept development, together with the co-generation of electricity and hydrogen or a range of other nuclear heat applications.

Figure 2: **GTHTR300C reactor concept**



Source: GIF, 2018.

Specific phenomena related to (V)HTR concepts are mainly associated with the presence of graphite in the reactor:

- Air ingress into the reactor could therefore lead to oxidation of graphite, or even a graphite fire.
- Water ingress could lead to corrosion of graphite and the production of flammable gases, together with increased core reactivity.
- It should be noted that, in the case of (V)HTRs, the favourable physical characteristics of the fuel and the reactor in general play an essential role in preventing the aggravation of incidents or accidents. Thus, in the event of loss of the cooling systems, the strong thermal

feedback due to graphite virtually stops the nuclear reaction, even without the intervention of a scram system. The heat removal via radiative and convective heat transfer, including a possible existing reactor pit heat removal system, should be modelled.

- As helium is non-condensable, it would seem unrealistic to confine it in the reactor "containment"[1] when the primary system depressurises. Regarding this last point, the short-term radioactive content released into the "containment" in the event of a LOCA would mainly contain fission and activation products associated with carbon dust present in the primary system under normal operation, which will be partly resuspended (Couturier, 2012).

3.1.3 *Gas-cooled fast reactor (GFR)*

The GFR concept aims to combine the advantages of a fast neutron reactor, with regard to the objectives of uranium conservation and minimising final waste, with high-temperature operation for efficient power-generation and hydrogen production. The power of GFR designs ranges from 200 MW_e (modularity) up to a larger 1 500 MW_e.

A typical 2 400 MW_{th} GFR plant has a break-even core, using helium as coolant and operating with a core outlet temperature of 850°C. The core consists of an assembly of hexagonal fuel elements, each consisting of ceramic clad, mixed carbide-fuelled pins contained within a ceramic hex tube. The favoured material at the moment for the pin clad and hex tubes is silicon carbide fibre-reinforced silicon carbide (SiCf/SiC). The whole primary circuit with three loops is contained within a secondary pressure boundary, the guard containment. The produced heat is converted into electricity in the indirect combined cycle with three gas turbines and one steam turbine. The cycle efficiency is approximately 48%.

A heat exchanger transfers the heat from the primary helium coolant to a secondary gas cycle containing a helium-nitrogen mixture, which in turn drives a closed-cycle gas turbine. The waste heat from the gas turbine exhaust is used to raise steam in a steam generator, which is then used to drive a steam turbine. Such a combined cycle is common practice in natural gas-fired power plants and so represents an established technology, with the only difference in the GFR case being the use of a closed-cycle gas turbine.

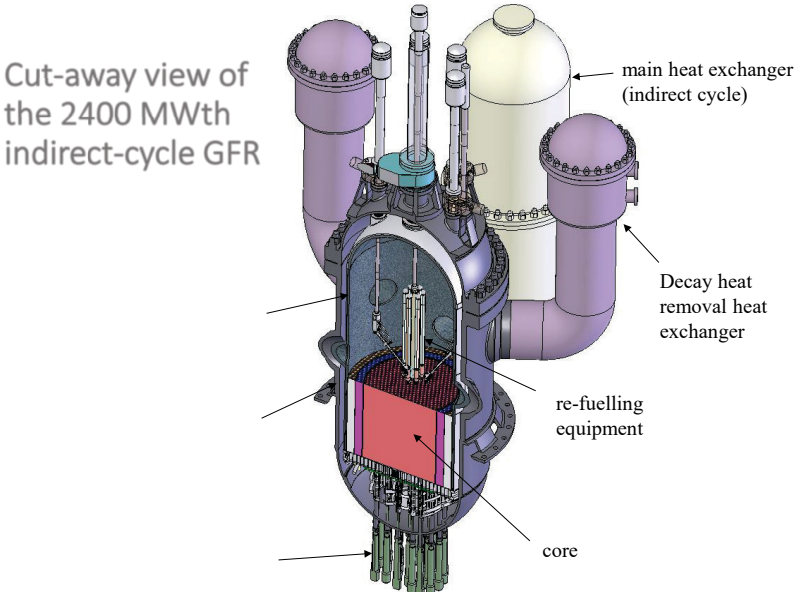
GFRs have some specific issues regarding safety:

- Prevention of accidents, particularly core damage, requires special attention in GFRs due to the use of a gas as coolant and the low thermal inertia of the system. This concerns helium-leakage (LOCA) scenarios in particular. It is necessary to maintain coolant circulation to remove the residual heat, at least during the first few hours following reactor scram. The quantity of residual heat removal available via natural convection alone is dependent on the fallback pressure of outer containment (Couturier, 2012).
- Special attention should be paid to the prevention of significant water ingress, an accident which would lead to reactivity insertion and oxidation of the fuel cladding.
- It seems that total or partial core meltdown in a GFR needs to be considered (Couturier, 2012). However, the materials that comprise the core have varied high-temperature behaviours; they may melt, break down or sublimate.

  The topic of re-criticality has to be considered. In case of core melt the thermal radiation from the high-temperature core could degrade the surrounding structures without them being in contact with molten materials.

1. The expression "containment" used here is a conventional containment that is defined as a (high) pressure retaining structural envelope like the LWR reactor building.

Figure 3: **Example of a GFR design**



Source: GIF, 2018.

3.1.4 *Molten salt reactor (MSR)*

An MSR is a reactor that, to a significant degree, uses molten salt within the core as a fuel carrier or coolant. There are two main MSR subclasses:

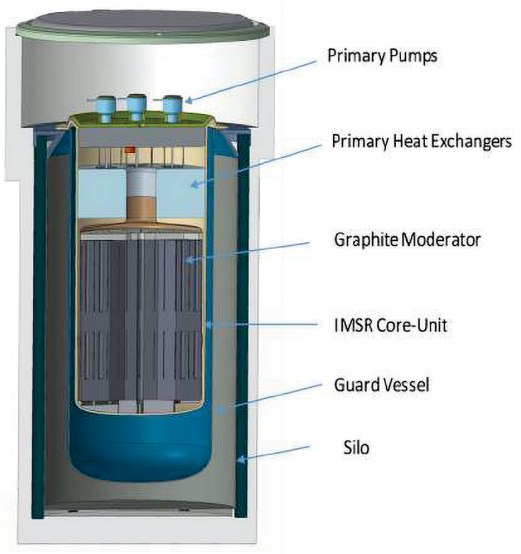
- In the first, fissile material is dissolved in the molten salt, which serves both as fuel carrier and coolant in the primary circuit.
- In the second, molten salt is the coolant in a graphite-moderated core fuelled with ceramic fuel, similar to that employed in VHTRs. This solid-fuel variant is typically referred to as an FHR to distinguish it from the previous one.

In liquid fuel MSRs, the fissile material is part of the liquid coolant. Significant elements of the MSR technology were developed and demonstrated in the 1950s and 1960s in the United States. While early MSR development focused on thermal neutron spectrum concepts, the liquid fuel MSR concepts under development by GIF members, following GIF's sustainability objectives, have been fast neutron spectrum reactors with circulating fluoride-based fuel in a closed fuel cycle.

The 1 400 MW_e thorium-uranium MSFR consists of a 2.25-metre equal diameter and height cylinder made of a nickel-based alloy, filled with the liquid fuel salt under ambient pressure conditions, operating at temperatures up to 750°C. The fuel salt in the primary circuit is pumped around in upward direction through the central core zone and in a downward direction through the heat exchangers located circumferentially around the core. In between is a thorium-fluoride salt-filled container to increase the breeding gain.

The fast spectrum reactor can be operated in the full range from breeder to burner mode. This flexibility is facilitated by the fact that the fuel salt composition can easily be adapted during reactor operation, since there is no need to manufacture solid fuel elements. In addition to the large MSFR considered so far and in view of identifying the potential limits of such concepts, small and medium-sized reactor (SMR) designs might mitigate some of the safety constraints identified for large power concepts, with their high fuel salt power density. Because of this, these SMRs may be the first type of MSFR to be developed, and related R&D studies are due to be launched in the near future to evaluate industrialisation viability and define the industrialisation process.

Figure 4: **Containment and reactor of IMSR concept**



Source: IAEA, 2018a.

To evaluate the safety of MSR type reactor concepts, the simulation codes should be able to model specific phenomena:

- The corrosive effects of molten-salt fuels are a major topic that must be considered, especially regarding the design of the containment barriers. The procedure for managing redox potential used in the molten salt reactor experiment (MSRE, ONRL) seems to provide an answer to the problem and was validated by MSRE operations (Couturier, 2012). However, the Hastelloy-N, developed for the MSRE, was found to have acceptable chemical compatibility in fuel-containing fluoride salt for short operating lifetimes, but was susceptible to degradation at longer lifetimes in more aggressive salt environments (Raiman, 2020).
- Related to simulations, further investigation related to salt properties are needed.
- Regarding the "delayed" neutrons, the boundary conditions related to neutron kinetics are different in an MSR type reactor with dissolved fissile material.

3.1.5 *Lead-cooled fast reactor (LFR)*

The LFR features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. It can also be used to manage actinides from used fuel and as a burner/breeder with thorium matrices. An important feature of the LFR is the enhanced safety that results from the choice of a relatively inert coolant, provided that the effects of weight and the corrosive and erosive nature of lead can be overcome.

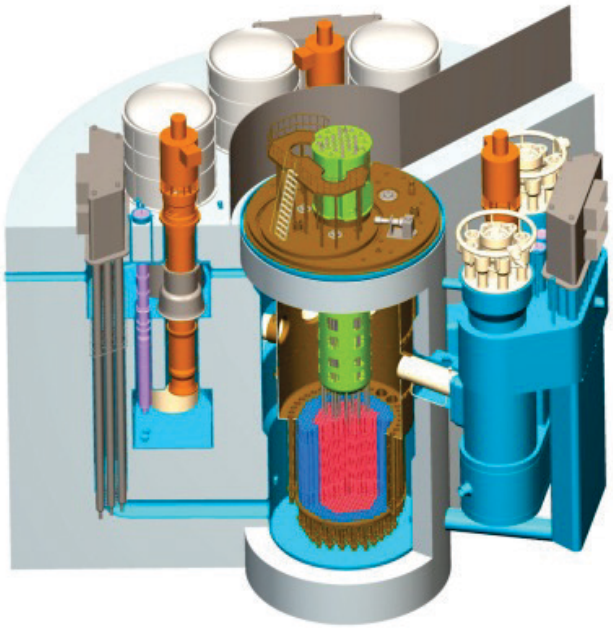
The designs that are currently proposed as reference concepts in the frame of Generation IV activities are three pool-type reactors:

- a reference design of 600 MWe based on the European lead fast reactor (ELFR), representing a large central electrical power generation unit;
- a 300 MWe option developed by Russia, which is in an advanced stage of development (BREST-OD-300). The detailed design of BREST-OD-300 (Figure 5) was submitted to the safety authority in 2016, and received a construction licence and began construction in 2021;

- a small modular design of 10-100 MWe, based on the small secure transportable autonomous reactor (SSTAR), representative of the small size initiatives related to LFRs (GIF, 2024).

All these three reference designs are cooled by pure lead, although lead-bismuth eutectic remains a coolant option. The present LFR reference designs feature simple primary circuits with an objective of maximising removability of internal components to promote competitive electric power generation and long-term investment protection.

Figure 5: **BREST-OD-300 concept**



Source: GIF, 2018.

Below are some points related to LFR-specific safety phenomena to be modelled:

- There is a risk of structural erosion and corrosion.
- The management of chemistry of lead coolant can be quite complex.
- In the event of air ingress into a lead or LBE system, or in the event of a small water leak from a steam-generator tube, compounds such as oxides, hydroxides and hydrides would form in the lead or LBE (Couturier, 2012).
- The topic of coolant freezing is more relevant compared to SFRs. Pure lead freezes at 327°C, which is only 73°C lower than the core inlet temperature during LFR operation.
- While lead and LBE do not react violently with air or water, contact between these molten metals and liquid water could lead to "vapour explosion" thermodynamic interactions, shockwaves and the displacement of structures and fluids.
- The behaviour of molten fuel and its relocation in an LFR is different to SFR-type concepts because the fuel is lighter than the coolant. As a result, molten MOX fuel should float on the primary lead. These physical characteristics of fuel and coolant require a specific strategy to be implemented for mitigation of core degradation (solutions envisaged for SFRs are not applicable). Questions related to re-criticality must be analysed.
- Due to the high density of lead, reactor resistance to earthquakes requires design precautions.

3.1.6 *Supercritical water-cooled reactor (SCWR)*

The SCWR is a high-temperature, high-pressure water-cooled reactor that operates above the thermodynamic critical point of water (374°C, 22.1 MPa).

Two types of SCWR core configuration are being pursued: pressure vessel and pressure tube. These core designs are based on thermal neutron, fast neutron or mixed (thermal and fast neutron) spectra. Uranium-based fuel has been adopted for the pressure-vessel type, with thorium-based fuel for the pressure-tube-type thermal-spectrum SCWR concepts. Mixed oxide-based fuel is selected for the fast-spectrum SCWR concepts. Most plant concepts are being developed for power generation higher than 1 000 MW_e at operating pressures of about 25 MPa and reactor outlet temperatures up to 625 C.

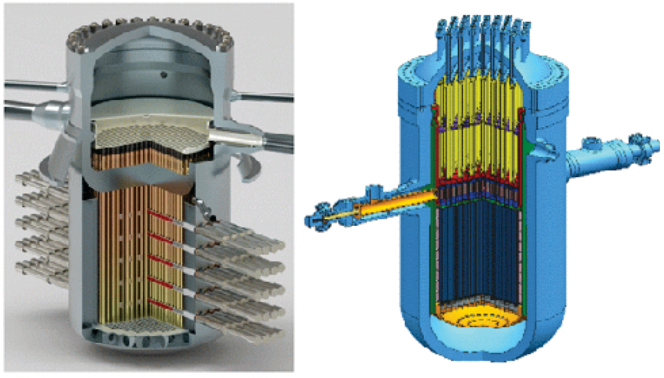
The SCWR balance-of-plant is considerably simplified because the coolant does not change phase in the reactor, which eliminates the need for the moisture separator reheaters and recirculation pumps found in boiling water reactors or the steam generators found in pressurised light water and heavy water reactors. The configuration of the main high-pressure circuits is comparable to that of fossil fuel-fired power plants thanks to many years of operation experience. The safety systems of the SCWR are similar to those of advanced boiling water reactors. Additional passive safety features have been incorporated to enhance the SCWR safety characteristics.

SCWRs are developed to generate high-temperature steam, ranging from 500°C to 625°C at the outlet of the core, which facilitates the use of a direct cycle as in the boiling water reactors, but without the need for the moisture separator, recirculation pumps or steam generators. The simplification of system configurations in the SCWR reduces considerably the size of the containment and the reactor building, as well as the footprint of the plant. The high outlet temperature of the SCWR core is applicable for hydrogen production, industrial process heat for chemical processing facilities such as petroleum refineries, and steam generation for oil production. Waste heat from the plant is ideal for potable water production using the desalination process.

The use of supercritical water in a nuclear reactor raises many questions, especially concerning materials (cladding materials and materials for reactor structures), which are frequently considered to constitute the major difficulty for this design regarding the characteristics of supercritical water (chemistry, high temperatures and high pressures):

- The very particular behaviour of water in the “pseudo-critical” region, with significant variations in its thermodynamic properties depending on the thermal flux in the fuel and its mass flowrate, would require much research, including under conditions during the normal transients of reactor start-up and shutdown (Couturier, 2012). The specific properties lead to specific problems to simulate heat transfer and simulation of flow instabilities. The reactor’s normal start-up and shutdown phases must be studied in detail.
- Study of these phases is not easy due to the physical complexity of an SCWR core: several zones with different levels of uranium-235 enrichment, strong coupling between neutronics and thermo-hydraulics, water flowing up the fuel assemblies and down the “water rods” and multiple passes through the core.
- Simulation of a loss of coolant accident (LOCA) scenario must be performed, during which the water would change from the supercritical state to a “normal” state, with separation of the steam and water phases and large variations in heat exchange depending on the composition of the mixture.
- Assessment of chemistry inside coolant (especially radiolysis) is necessary.
- Assessment of materials under these specific conditions including irradiation, must be conducted.

Figure 6: **Pressure-tube type and pressure-vessel type SCWR concepts**



Canada's Pressure-Tube Type SCWR Core Concept

China's Pressure-Vessel Type SCWR Core Concept

Source: GIF, 2018.

3.2 Overview of relevant phenomena

While a comprehensive discussion of safety aspects is beyond the scope of this report, for each reactor concept a brief overview of relevant phenomena is provided with the purpose of justifying computational or analytical modelling needs.

So-called safety-relevant physical phenomena are those involved in the achievement of the main safety functions and in the accidental transients assessed within the reactor safety demonstration. The following sections extend the discussion on relevant phenomena identified in Section 3.1 to highlight important phenomena and identified knowledge gaps associated with the evaluation of Generation IV reactor concepts.

3.2.1 *Specific safety-relevant phenomena in an SFR*

The identification of the physical phenomena that can challenge the safety of an SFR is grounded in decades of operational feedback of experimental facilities and prototype reactors. The design specifics of SFRs that have an impact on safety and deserve physical modelling are mainly the high-temperature operation, the high fast neutron flux density in the core, the potential for chemical interactions of sodium with water, air and concrete and the risk of prompt critical events in the case of severe accidents. These issues are developed hereafter, based on the results of the NEA TAREF project (NEA, 2011).

3.2.1.1 *Fuel and materials*

The wide variety of fissile materials and cladding materials that can be irradiated in SFRs implies a corresponding variety of models to describe their behaviour. In particular, phenomena involved in metal fuel/cladding chemical interaction need to be investigated further, especially when a significant amount of plutonium is introduced in the alloys. For all the types of fuels, modelling of the behaviour at high burn-up is safety-relevant and has to be improved, especially in accidental transients with regard to the margins to fuel melting and the fuel/cladding mechanical interaction. This later phenomenon is important considering the low swelling rate of novel clad materials. Codes must also be able to take into account the impact of very strong changes in the structure of the fuel on the fuel performances.

Finally, it should be noted that very few data are available on the irradiation of fuel bearing minor actinides.

Although the structural materials of the internal core structures, second barrier and sodium circuits are well characterised, a fine description of the variations of thermo-mechanical stresses and the resulting damages is needed. The phenomena of creep/fatigue interaction and thermal ageing are prominent in the demonstration of the integrity of the second barrier surrounding the primary sodium and cover gas. On this point, the modelling of long-term fatigue damages under low amplitude stress variations remains a challenge (for 60 years of operation).

3.2.1.2 *Reactor kinetics and thermal hydraulics*

Models describing reactor kinetics are essentially limited by uncertainties on nuclear data. This may be especially challenging for the computation of the safety coefficients of heterogeneous cores with integrated sodium plena, fertile material layers, moderated subassemblies (for transmutation), etc. As an example, end of life tests in Phénix revealed high uncertainties in the computation of fissile subassembly power close to moderator-bearing experimental assemblies. The computation of core power in case of accidental transients which depends on the combination of positive and negative feedback effects may require more precise nuclear data than those which were utilised up to now. The impact of the cross sections of the minor actinides is also to be taken into account in accidental transients.

Concerning reactor thermal hydraulics, the analysis of the risk of sodium boiling in subassemblies is essential for the demonstration of the margins in terms of clad integrity and during loss of sodium flow events at global and local scales. Such a phenomenon may require a coupled approach due to the positive reactivity effects induced by the voiding of the core (effects intensity depending on core design).

Detailed description of the subassembly internal flow in case of partial blockage in the pin bundle would allow for a more comprehensive and graded safety demonstration (definition of design basis accidents of flow blockage). Such a description must rely on fine turbulence models.

More globally, a description of the field of primary sodium temperatures with relevant accuracy is needed together with the heat transfer to the core supporting structures and second barrier. Such results are boundary conditions for the computation of the various thermal-induced damages on these safety important structures. Turbulent flow regimes, mixing zones and cavitation are phenomena of importance in this field.

Specific flow regimes associated with natural convection in primary and decay heat removal circuits need to be numerically investigated. Of particular importance are the thermal-hydraulic conditions which trigger the natural convection and the phenomena of thermal stratification in the fluid.

3.2.1.3 *Main accident sequences*

Typical of SFRs is the risk of reactivity insertion caused by core distortion, coolant density variations and moderator ingress (oil, carbonaceous substances, etc.).

However, accidents which may involve local or extended core melt are the main concern for SFRs because their management requires not only restoring sufficient core cooling but also preventing criticality and the potential for high mechanical energy release.

In MOX-fuelled reactors, many complex physical phenomena are involved in the envisaged scenarios of severe accidents. Firstly, an important phenomenon is the thermo-chemical reaction between the hot fuel and the sodium. Safety demonstrations should also take into account 3D effects associated with local subassembly faults or reactivity insertion in which coupling between neutronics and thermal-hydraulics is significant. For a severe accident with core meltdown, besides the sodium-fuel reaction, the main physical concern is the description of heat exchanges and motions in multiphasic media involving gases, viscous liquids and solids. The neutronic power of these media has to be taken into account as well as the risk of criticality. The scenarios with mechanical energy release require the evaluation of the conversion between nuclear power, thermal power and mechanical energy. The consequences of mechanical energy release depend on the dynamic response of the primary structures that surround the core.

Accidents with core degradation in metallic fuelled SFRs may involve different phenomena or, at least, the relative importance of the phenomena is different when compared to MOX fuelled reactors. However, thermo-chemical reaction phenomena are not expected with metallic fuels based on UPuZr alloys. The same questions occur with respect to nitride and carbide fuels, which can also undergo thermal decomposition.

3.2.1.4 *Sodium chemical interactions and fires*

The primary consequence of sodium leaks and fires is the loss of coolant leading to the degradation of the core cooling function. Usually, sodium fires involve non contaminated sodium because of inertisation of double envelopes isolating the primary sodium circuits, but contaminated sodium fire may be considered depending on the reactor design. Contaminated sodium fire could also be associated with a severe accident when mechanical energy release is plausible. More generally, sodium fire should be studied as an internal hazard (i.e. with the potential to damage safety related SSCs) and as a source of toxic release outside the plant. For that purpose, fire typologies, combustion rates, heat exchange correlations and aerosol behaviour are the important phenomena. Simulation of sodium-water interaction in stream generators (SG) is requested in particular to design the overpressure protections of the intermediate circuit (part of the second barrier). On that issue, it must be noted that potential interactions with coolants other than water could be studied as relevant. Finally, sodium-concrete interaction may be of concern in case the structures are not protected by metallic liners.

Regarding fire typology, combustion in spray and pool fires is modelled. Spray fires are the major concern and require model improvement in the field of low-medium temperature spray (delayed ignition) and configuration with 3D issues. In fact, specific circuit geometrical configurations often require representative tests to tune the numerical models which remain semi-empirical. Characterisation of heat exchanges between fire atmosphere and surrounding structures is essential for the safety demonstration.

Regarding the aerosols, the main relevant phenomena are their production, deposition rate in ventilation circuits and rooms, and chemical reactions with CO_2 and moisture.

The computer codes used to model sodium-water reactions take into account the dynamics of the gases, steam and water in the SG circuits and the wastage phenomenon, which depends on the materials (fast erosion of water tubes close to the reaction area).

3.2.2 ***Safety-relevant phenomena in a VHTR***

The section below is informed by the results of a phenomena identification and ranking table activity that was completed in 2007 (NRC, 2008). Several phenomena are not specific to VHTRs only, but also valid for other high-temperature concepts.

3.2.2.1 *Physics and chemistry of materials*

High-temperature materials

The major aspects of material degradation phenomena that may give rise to regulatory safety concerns have been evaluated for major structural components and their associated materials (NRC, 2008). These material phenomena were evaluated with regard to their potential for contributing to fission product release at the site boundary under a variety of event scenarios covering normal operation, anticipated transients and accidents, and the currently available state of knowledge with which to assess them. Key aspects identified for the VHTR in the area of high temperature materials are:

- High-temperature stability and a component's ability to withstand service conditions.
- Issues associated with fabrication and heavy-section properties of the reactor pressure vessel.

- Long-term thermal ageing and possible compromise of reactor pressure vessel surface emissivity as well as the reactor cavity coolant system.
- High-temperature performance, ageing fatigue and environmental degradation of insulation.

Graphite

Much has been learnt about the behaviour of graphite in nuclear reactor environments since the first graphite reactors went into service. It is expected that the behaviour of these graphites will conform to the recognised trends for near-isotropic nuclear graphite (NRC, 2008). However, the theoretical models still need to be tested against experimental data for the new graphites and extended to higher neutron doses and temperatures typical of Generation IV reactor designs. Significant phenomena associated with the use of graphite in the VHTR have been identified as:

- material properties (creep, strength, toughness, etc.) and the respective changes caused by neutron irradiation;
- fuel element coolant channel blockage due to graphite failures;
- consistency in graphite quality (includes replacement graphite over the service life); and
- dust generation and abrasion (especially noteworthy for pebbles, but of concern as well for the prismatic design).

3.2.2.2 *Reactor kinetics and thermal-fluids*

The most important phenomena identified by a panel of experts associated with the thermal-fluids (including neutronics) aspects of the VHTR (NRC, 2008) include:

- Primary system heat transport phenomena (conduction, convection and radiation), including the reactor cavity cooling system (RCCS) performance, that impact fuel and component temperatures.
- Reactor physics phenomena (feedback coefficients, power distribution for normal and shutdown conditions) as well as core thermal and flow aspects. These often relate to the power-to-flow ration and thus impact peak fuel temperatures in many events.

Specific phenomena that have been identified as being of high importance with a low knowledge level include the following (NRC, 2008):

- Outlet plenum flow distribution during normal operation – Localised hot spots and excessive thermal gradients may lead to structural problems. Thermal streaking may lead to problems with downstream components such as a turbine or intermediate heat exchanger.
- Core coolant bypass flow – Bypass flow varies with shifts in block gaps (prismatic VHTR). The effect of the pebble-bed core wall interface on bypass flow has also been recognised as an issue. Bypass flow impacts fuel time at temperature and fuel failure fraction.
- Reactivity-temperature feedback coefficients – There are limited available experimental data for the validation of reactivity temperature effects, particularly through direct measurements of reactivity coefficients rather than using tests of the overall transient response of the system. Additionally, there are limited data for high-burnup fuels. Physical phenomena may not be adequately captured by current spectral codes (e.g. resonance capture at high temperatures) and may be the cause of miscalculation of power coefficients. The high temperature of the VHTR magnifies errors in differential feedback coefficients compared with relatively well-known systems.
- Fuel performance modelling for tristructural isotropic (TRISO) fuel is a major consideration for determining fuel failure fraction and contribution to accident source term. Unknowns such as kernel migration and silicon carbide morphology modelling and the impact on radionuclide release have been identified.

Additional considerations affecting VHTR reactor physics and thermal-fluid modelling include the double-heterogeneity of the fuel (the first heterogeneity being the TRISO particle, and the second being the fuel compact). A technique for capturing the double-heterogeneity of the fuel is required in the development of cross-section data for core wide reactor physics analyses. Additionally, the thermal-fluids modelling techniques need to consider the multi-scale heat transfer aspects introduced by TRISO fuel (i.e. micro-scale within the TRISO particle, meso-scale within the fuel compacts, and macro-scale within the bulk material of the reactor core).

3.2.2.3 *Initiating hazards and accident scenarios*

A panel of experts defined the following scenarios when identifying and ranking important phenomena associated with thermal-fluids and neutronics analyses (NRC, 2008):

- normal operations;
- pressurised loss of forced cooling;
- depressurised loss of forced cooling;
- air ingress;
- reactivity insertion events;
- phenomena associated with the process heat loop; and
- phenomena associated with the intermediate heat exchanger (e.g. overcooling/undercooling).

3.2.3 ***Specific safety-relevant phenomena in a GFR***

This chapter describes the specific safety-related phenomena relevant to a GFR-type reactor concept. The applied analytical codes and methods must cover these types of phenomena. The content of this chapter is mainly based on the results of the NEA TAREF group (NEA, 2009).

There are certain characteristic differences between a (V)HTR and a GFR concept. For GFR systems, a specific characteristic is the lack of thermal inertia of the primary system, combined with the pressurised gas and the associated risk of leakage. Unlike a (V)HTR, the safety of the reactor mainly relies on active systems (e.g. helium blower) and the progression of a transient depends on the characteristics of several critical components. In consideration of the fast neutron environment, the GFR core physics is sensitive to several reactivity effects.

3.2.3.1 *Safety issues in the accidents and thermal fluids area*

A GFR specificity is the lack of thermal inertia of the primary system, combined with the pressurised gas and the associated risk of leak. The safety of the reactor relies on active systems and the progression of a transient depends on the characteristics of several critical components that are stimulated under conditions that are far from normal in terms of pressure and temperature.

Accident scenarios and natural convection

Decay heat removal emergency systems are based on forced gas circulation through the core. Under high pressure or in a later phase of an accident, natural circulation through several decay heat removal loops might be possible to remove the decay heat. However, the onset of natural circulation, possible inner circulation or direct circulation between different loops must be simulated. Here the estimation of pressure drops through the different components is an essential point.

In GFRs, modifications of the core geometry, for instance due to a seismic event or the pressure wave caused by a depressurisation transient, may increase the reactivity and hence the core power.

Behaviour of critical components

Technology using helium gas needs specific components compared to heavier gas like air. For instance, a compressor in helium has a special design due to the properties of helium. In case of emergency operation, the properties of helium changes due to new thermal hydraulic conditions. This should be considered in simulations. A possible impact on structures (e.g. the guard containment) must be considered.

Nuclear data on specific GFR materials and spectrum conditions

Nuclear data for core physics on specific GFR materials (carbide, helium, neutron absorber metals, and high content of minor actinides) and spectrum conditions must be simulated.

3.2.3.2 *Safety issues related to materials and fuel*

Ceramic material properties

In GFRs, during some anticipated severe conditions, the gas at core outlet can reach more than 1 200°C and heat shielding using ceramic materials must be used to withstand this temperature for several minutes.

Behaviour of GFR ceramic fuel element including degradation modes

The reference GFR fuel element is a carbide fuel cladded with a composite matrix ceramic. Up to 1 600°C the fuel must keep its operating properties and mainly its capability to retain fission products. The second temperature threshold is examined in the "severe accident" technical area. The fuel must keep its geometry. However, simulation codes should be able to simulate a total or partial core degradation. Pin and plate type fuel element geometry should be considered. One major difficulty is that the degradation process could be very different from the one studied in a PWR or in a sodium fast reactor. For instance, no clad melting is expected, but presently one cannot predict the conditions and the way a SiC/SiC clad might collapse.

Mechanical behaviour of fuel element and assembly in a GFR

To exploit the high-temperature potential of GFRs, the use of advanced ceramic materials (e.g. SiC fibres and/or composites) is sought in some concepts for the core components, and notably for the cladding of the fuel elements, usually in combination with advanced fuels (e.g. carbides). This introduces specific elements of novelty that need to be captured for an exhaustive safety analysis.

3.2.4 ***Specific safety-relevant phenomena in an MSR***

This section extends the discussion on relevant phenomena identified in Section 3.1.4 to highlight important phenomena and identified knowledge gaps associated with the evaluation of molten salt reactors. The content provided below is informed by the results of a phenomena identification and ranking table activity that was completed in 2018 (BNL, 2018).

3.2.4.1 *Physics and chemistry of materials*

Fuel salt/coolant

As discussed in Section 3.1.4, molten salt serves as the fuel salt in most MSR designs and as a coolant to a solid fuel in FHR designs. Salt performance as described by thermophysical properties has been identified as important to capture during normal operation and accident scenarios (BNL, 2018; ORNL, 2018). Due to the experience with MSRE, some salt properties are sufficiently well known to support licensing activity and material qualification (Kairos, 2019).

Phenomena associated with the high temperatures reached by molten salts have been identified as significant (BNL, 2018). Specifically, radiative heat transfer is significant at the elevated temperatures, which increases the importance of salt's optical properties. Phase change in the form of freezing is an issue that could occur due to localised cool spots (e.g. in a heat exchanger tube) or during over-cooling events.

Salt chemistry changes associated with fission product building and the resulting impact on thermophysical properties has been identified as an important phenomenon that is unique to reactors utilising fuel salts (ORNL, 2018). Active research is ongoing to develop a process for qualifying fuel salts with a recognition that salt chemistry is expected to be impacted by the addition of fission products during reactor operation.

There are potential challenges associated with large tritium generation rates in MSRs utilising salts containing lithium and/or beryllium, either as a fuel-slat or coolant-salt. The modelling and simulation of tritium behaviour is central to developing appropriate control strategies and demonstrating safe operation as part of the reactor licensing process. The modelling and simulation of MSRs should address (1) tritium production, (2) tritium behaviour within the reactor system, and (3) tritium behaviour outside the reactor system (ANL, 2020).

Because of the limited feedback experience, the understanding of the phenomenology of accidental processes in MSRs would be also limited. For example, one could mention some ontological uncertainties such as:

- the viscosity and pseudo-viscosity of the fuelled-salt and mushy zones under irradiation;
- the intervals of the salts' stability for more than ternary-component solutions, e.g. contaminated with fission and corrosion products;
- uncertainties over the major nuclear data in the epithermal spectra needed for criticality and fuel inventory studies;
- the quantified power and kinetic characteristics of the bed-type system in case of delay neutron shift and/or turbulent intermittency;
- the practical limits of the controllability depending on many factors, including power.

The first three points merely refer to the simulation inputs, while others correspond to deep systemic studies. Of course, the list is far from complete and further in-depth analysis would reveal more uncertainties.

Structural materials

Plate-out and corrosion of structures is an important phenomenon to be considered in the evaluation of an MSR (BNL, 2018). Precipitation and collection of contaminants on heat exchanger surfaces can result in fouling of heat exchanger surfaces.

Many MSR designs utilise carbon as a moderator or structural material, and FHR designs utilise carbon in the fuel structure. Potential change in carbon density and dimensions due to irradiation is an important phenomenon to be considered. Additionally, graphite is known to experience fluorination when exposed to fluoride salts (Wu, 2018).

3.2.4.2 *Reactor kinetics and thermal-hydraulics*

Currently, there are no cross-section data that represent the effect of molecular bonds (i.e. thermal scattering kernel) on thermal neutron scattering in molten salts. The absence of a thermal scattering kernel for molten salts leads to a computational bias of unknown magnitude in all reactor design calculations (Bernstein, 2019). A benchmark experiment evaluation of the MSRE demonstrated a reactivity balance bias of 1 000 pcm from the experimental value with other presented results showing biases of over 3 000 pcm (Shen, 2018). Reactor designs that utilise molten salts must address this source of uncertainty in their reactor design and performance analyses through adequate design margins and by having processes in place to ensure that key figures of merit (e.g. reactivity balance, power distribution, shutdown margin) are maintained within the bounds of the safety analyses.

Maintaining a negative temperature feedback coefficient for the salt is essential for MSRs. A strong negative temperature feedback is expected for designs that utilise a fuel salt because heat-up and expansion of the fuel salt displaces the fissile material. However, for FHR designs, where the salt is used as a coolant to solid fuel, there is concern that parasitic neutron absorption in the salt coolant can result in a positive coolant temperature coefficient. Care must be taken to ensure that sufficient margin is included in the reactor design to accommodate sources of uncertainty such that a negative temperature coefficient can be assured.

MSR designs based on the use of a fuel salt experience delayed neutron drift. This is the phenomenon associated with having a moving fuel that results in delayed neutron precursors emitting delayed neutrons outside of the core region. The effect of delayed neutron drift is a lower delayed neutron fraction, β.

3.2.4.3 *Initiating hazards and accident scenarios*

Because liquid-fuel MSRs are fundamentally different from solid-fuel reactors, a variety of new accident scenarios will have to be considered, such as reactivity insertions associated with a failure of material control in a fuel processing system. Many MSR designs based on the use of fuel salts also include a helium bubbler (sparer) to assist in the removal of fission products from the fuel salt. The presence of this compressible gas within the fuel salt, combined with the strong feedback coefficient associated with a compression of the fuel-salt, results in reactivity insertions being associated with events that cause an increase in system pressure.

3.2.5 ***Specific safety-relevant phenomena in an LFR***

LFRs share many features with SFRs in terms of design approach and layout, and therefore also share several safety-relevant phenomena. Since extensive work has been done developing analytical codes and methods tailored for SFRs, in this section the differences between LFRs and SFRs are clearly drawn as a starting point before any LFR-specific aspects and related phenomena are discussed.

3.2.5.1 *Physics and chemistry of materials*

Coolant

Besides the different physical and chemical properties of lead (relative to sodium), the following aspects should be considered in the development or adaptation of any code or method for application to an LFR.

- The high boiling point of lead in practice excludes coolant boiling scenarios, allowing for single-phase analysis in the whole operational and accidental domain.
- Lead vapours are toxic, and their release mechanisms and pathways must be included in appropriate analytical codes and methods.
- Polonium is generated in lead, and particularly in lead-bismuth eutectic (approximately four orders of magnitude more than in commercially pure lead), and largely retained as lead polonide; the retention/release mechanisms must be included in appropriate codes and methods, accounting for all nominal and accidental scenarios.

Fuel

An LFR shares many fuel-related safety aspects with other fast spectrum systems, such as SFRs. The key differences, which must be reflected in the analytical codes and methods for their correct representation, are:

- the lack of exothermic chemical reactions between the fuel and the coolant;
- the very high fission products' retention in lead in case of release from the fuel elements;

- the similar densities of oxide fuels and coolant, making relocation of the fuel, in case of its dispersal from the core, not straightforward but dependent on the specific thermal and hydraulic conditions.

Additionally, some LFR designs consider non-oxide fuels (e.g. nitrides), which introduce elements of novelty that must be captured by codes used for safety analysis.

Structures

The key differences that must be reflected in the analytical codes and methods for their correct representation are listed below.

- If exposed to high temperature, structures immersed in lead undergo corrosion. The mechanisms of corrosion (e.g. dissolution of alloying elements) and of protection (e.g. formation of oxide scales by control of the oxygen concentration and their removal under the action of flowing lead as a function of the flow velocity) must be implemented in analytical codes and methods to properly evaluate the effective resisting thickness of structures for their mechanical integrity. For even higher temperatures, corrosion protection by coatings must be investigated wherever the temperatures exceed the limit of effectiveness of the oxide scale. Codes must be available that can consider the mechanisms of corrosion and of formation/removal of protective means.
- Ferritic-martensitic steels exposed to lead exhibit liquid metal embrittlement, whose modelling must be considered to properly evaluate the mechanical resistance of structures and as such are not typically proposed for use in LFR cores (IAEA, 2020).
- Buoyancy forces due to the lower specific gravity of structures respective to lead must be allowed in analytical codes and methods whenever required.
- The high density of lead makes erosion rates rise significantly with increasing coolant velocity. This introduces the need to account for such aspects in the evaluation of the performance and long-term operation of the high-speed rotating parts of the circulation pumps.

3.2.5.2 *Reactor kinetics and thermal hydraulics*

Reactor kinetics

The typical LFR parameters ruling reactor kinetics are similar to those of other fast spectrum systems like SFRs (i.e. reactivity feedback that is lower than in LWRs, regions of positive lead density coefficient in the active core, fast kinetics). The most significant differences, which must be reflected in the analytical codes and methods for their correct representation, are:

- the practical elimination of core voiding scenarios, because of the high boiling temperature of lead; and
- the mechanisms triggering the reactivity effects (e.g. relative direction of expanding structures, temperatures driving thermal expansions) due to the reactor layout (e.g. bottom-insertion of buoyancy-driven control and shutdown elements).

Core thermal hydraulics

The core thermal hydraulics of an LFR are very similar to those of an SFR except for the following aspects with possible safety implications:

- large coolant sub-channels are typically employed in the assemblies also because of the low flow velocity; both aspects require adaption of appropriate models for pressure drops (including for the impact of spacer grids, used instead of wire spacers) and bundle heat exchange in the codes and methods;

- low flow velocity and high coolant density modify the forcing terms of flow-induced vibrations, requiring adaption of appropriate models in the interested analytical codes and methods.

System thermal hydraulics

The overall system thermal hydraulics of an LFR are very similar to those of an SFR, so that codes and methods developed for the latter can easily be applied to the former. The following aspects, however, should be ensured, as specific and characteristic of LFRs:

- the chemical compatibility between lead and water/air allows for the steam generators and heat exchangers of decay heat removal systems to be installed in the primary system, with elimination of the intermediate circuit; such possibility should be allowed by any system code or method;
- the high density of lead enhances thermal stratification and striping effects, so that any model implemented in existing codes and methods should be verified to remain within its validity range, or new models should be developed and validated instead.

3.2.5.3 *Initiating events and accident transients*

Initiating events

Many initiating events for SFRs also apply to LFRs, with some specific considerations for technology differences:

- all initiators pertaining to structures, systems or components of the intermediate circuit do not apply since there is no intermediate circuit;
- where the lack of phenomena determining large energy releases in the core (e.g. sodium-fuel interaction) also eliminate the risk of significant core damage;
- sloshing effects in case of an earthquake, as they might be magnified by the high density of lead, requiring its modelling to be included in codes and methods when appropriate;
- the rupture of a tube in the steam generator, which must be treated considering that the latter is included within the reactor coolant boundary; to properly address such an initiating event, analytical codes and methods require modelling capabilities of the pressure wave propagation in the coolant, of the water/steam-lead chemical interaction (including with any species present as impurity or activation product, notably polonium), and of the water/steam transport through the primary circuit (up to, possibly, the core). For those systems considering alternative secondary cooling fluids (e.g. supercritical CO_2), diverse phenomena of chemical and physical interaction with lead must be represented in the codes for simulation of this type of initiating event;
- when the elements resulting from corrosion and erosion of structures deposit on structures, possibly making the plugging of flow paths an additional mechanism for a blockage-type initiating event.

Accident transients

For LFRs, like other FRs, threats for design extension conditions are combinations of events, one (mandatory) of which is loss of scram. Typical accident transients include unprotected scenarios such as unprotected loss of flow (ULOF), unprotected transient overpower (UTOP) or unprotected loss of heat sink (ULOHS). For each there are typical behaviours and challenges, as described below.

- Upon unprotected loss of flow events, the balance between heat generation and removal from the core is quickly lost, with the temperature of coolant and core structures increasing. Thanks to the specific gravity of lead and the negative reactivity feedbacks, a new balance is found under natural circulation at reduced power. However, the

increased temperature poses a challenge to the long-term integrity of the cladding due to creep rupture.

- In unprotected transient overpower events, a reactivity insertion makes power quickly rise, due to the fast kinetics, before the temperature of fuel and other elements increases, thus before fuel Doppler and other reactivity effects contrast the reactivity insertion. The main challenges, both of which are in the short term, are to avoid fuel melting and to preserve the integrity of the fuel cladding.
- Unprotected losses of heat sink pose long-term challenges to the integrity of both the fuel claddings and the reactor coolant boundary. The significant reduction of heat removal from the system makes all temperatures steadily increase until the feedback coefficients reduce the core power to a level equal to that of the residual heat removal capacity.

Simulation codes called to analyse these types of transients must address due modelling of all the involved phenomena (singularly listed in all previous subsections), and be validated in the specific domains pertaining to the LFR, which might differ from those of other FRs.

3.2.6 *Specific safety-relevant phenomena in a SCWR*

This section describes the specific safety-relevant phenomena related to a SCWR type reactor concept. The applied analytical codes and methods should cover these types of phenomena.

3.2.6.1 *Reactor kinetics and thermal hydraulics*

Beside the typical safety-relevant phenomena in a light water reactor (LWR) several additional SCWR specific phenomena have been identified (Aksan, 2019; IAEA, 2019a). The thermal hydraulics of supercritical pressure fluids is one of the most important areas to establish and analyse the SCWR concepts. Information on heat transfer and pressure drop is required for analyses of normal operating conditions and postulated accident scenarios. Heat transfer calculations are applied to establish the fuel and cladding temperatures. Pressure drop analyses are needed to establish the mass flowrate in the core and the pump capacity/requirement. In addition, these analyses help to establish the subchannel flow in fuel assembly. Stability calculations are essential to identify the operating and start-up conditions for SCWRs in view of the significant variation in density in the core, which could lead to dynamic instability. Another interesting SCWR characteristic is the strong coupling between thermal hydraulics and neutronics. Instability could lead to power fluctuations within the core. The ability to depressurise rapidly is a requirement for SCWR in mitigating any upset conditions. A pressure relief valve is one of the major components in the ADS. Design of this valve requires critical flow behaviour at supercritical pressures. In addition, information on this behaviour is also needed for the analysis of postulated large break loss of coolant accidents (LOCAs).

Reactor kinetics

Under SCWR conditions the pronounced water density reduction with the heat-up, together with the multi-pass core design, results in a pronounced coupling between reactor physics and thermal-hydraulics core analyses that cannot be neglected. The water density distribution within the core changes appreciably the reactivity and the leakage probability of the different core regions while the fuel temperature variations, and the associated actinides resonance broadening, result in 3D feedback distribution that modifies the power generation within the core. The steady state operative condition can be predicted only with coupled reactor physics/thermal-hydraulics analyses requiring the development of corresponding computational systems (Monti, 2009). The temperatures of the various materials within the core depend on the in-core power distribution, which is actually modified by the change in the material properties requiring a coupled analysis also for the determination of the steady state operative condition. In addition, large temperature gradients may cause mechanical deformations, and hence core geometry changes, adding further complexity to the considered analyses.

Thermal-physical properties near and at supercritical conditions

Heat transfer at supercritical pressure is mainly characterised by the thermal physical properties, which vary strongly, especially near the pseudo-critical line. The simulation codes should implement these properties and manage the property changes along the pseudo-critical line, without creating numerical instabilities.

Critical flow during depressurisation from super-critical conditions

During a postulated loss of coolant accident (LOCA) of a pressurised water reactor (PWR), the break flow will be subject to this condition. An accurate amount of the critical flow rate is important for the evaluation of the reactor safety, because this flow rate controls the loss of coolant inventory and energy from the system, and thus has a significant effect on the accident consequence. A depressurisation event or LOCA is particularly of concern for the SCWR safety due to the lower coolant inventory compared to a typical PWR for the same power output. This lower coolant inventory would result in a faster transient response of the reactor dominated by the discharge flow rate. Critical flow is coupled with drastic depressurisation and vaporisation, and the flow rate is heavily dominated by the vapour content or quality of the vapour, which is closely related with the onset of vaporisation and the interfacial interaction between phases. This presents a major challenge for the estimation of the flow rate due to the lack of knowledge on those processes, especially under the conditions of interest for the SCWR.

Pressure loss characteristics of supercritical water

Pressure losses in the heat-transport system have a direct impact on the pump capacity and the power output, affecting the capital and operating costs. In addition, these losses affect the local flow conditions along the fuel assembly, impacting the heat-transfer characteristics. Depending on the design, the pressure loss components consist of frictional loss, form losses, head loss due to gravity, and loss due to acceleration. A significant portion of the overall pressure loss occurs in the core region (particularly over the bundle assembly). Therefore, it is important to evaluate the core pressure loss accurately.

Heat transfer characteristics of supercritical water

The fuel design criterion for the SCWR is based on the cladding temperature limit because phase change is not present in the core during normal operation. The lack of qualified experimental data on heat transfer for supercritical water flow in rod bundles has been identified as a significant challenge to the SCWR design.

In supercritical fluids, the properties change rapidly with the temperature, especially in the pseudo-critical region. Here, the local heat transfer coefficient varies significantly, depending on flow conditions and the value of the heat flux. At a high heat flux a similar behaviour of the wall temperature is expected, except for the bulk temperature approaching the pseudo-critical value. In this case a sharp increase in the wall temperature can occur. The wall temperature decreases again when the bulk temperature exceeds the pseudo-critical value. This large increase in the wall temperature is referred to “heat transfer deterioration”.

The large deviations between test data related to heat transfer lead to the assumption that this deviation obeys a physical phenomenon, the so-called multisolutions. This means that under the same experimental conditions, i.e. test channel geometry, pressure, mass flux, heat flux, and bulk temperature, the wall temperature has different values (Baindur, 2008). Efforts should be made to further clarify the possibility of a multisolution and the conditions where a multisolution may occur. This work is high priority, with an impact not only on future experiments, but also on CFD validation and on the development of HTC prediction methods. Another possibility leading to the large deviation between the test data could be the effect of the upstream flow condition on HTC (Baindur, 2008).

Reflooding for tight lattice geometry at low pressure

The geometry and flow paths inside a SCWR core differ from conventional LWRs. In general, quenching and rewetting phenomena processes known from LWRs are valid also for SCWR core geometries. However, it must be considered that specific reflooding modelling and experiments are needed for selected geometrical consideration.

Boil-off for tight lattice geometry at low pressure

Like reflooding phenomena, the principal boil-off phenomena are similar to those in LWR-type cores. However, it must be considered that specific CHF and post-CHF heat transfer models and experiments are needed for selected geometrical consideration.

Instability and natural circulation in supercritical pressure systems

The design of supercritical water-cooled reactors poses new challenges in relation to the prediction of some basic phenomena whose characteristics are relatively well-known at subcritical pressure. In particular, stability and natural circulation phenomena observed in single-phase conditions at subcritical pressures become more complex at supercritical pressures because of the strong variability of thermodynamic and thermophysical properties. Therefore, stability has been addressed as an issue of great importance for SCWR design. Though relevant experiments and modelling tools are already available in this field, there is the need for further understanding and specific assessments.

Loss of coolant accident (LOCA) scenario

In a LOCA scenario, the water changes from the supercritical state to a "normal" state, with separation of the steam and water phases and large variations in heat exchange, depending on the composition of the mixture.

Start-up systems and procedures

Starting from cold conditions, the first reactor power will be needed to warm up the steam cycle; either to start with constant supercritical pressure by depressurising some coolant into a flash tank, or to start with a sliding, subcritical pressure by separating water and steam from the reactor core in external cyclone separators. In either case, the separated liquid is taken to preheat the feed water and the remaining steam warms up the turbines. As dryout will be unavoidable in the reactor core during subcritical operation, the maximum cladding surface temperature of the fuel rods needs to be checked to avoid damage.

Safety systems

Safety systems for SCWRs have been investigated specifically per reactor concept. The basic idea of SCWR safety systems is the same as that of conventional WCRs. The emergency core cooling system of SCWRs consists of high-pressure and low-pressure coolant injection systems and automatic depressurisation systems, and some of the concepts adopt or try to adopt passive cooling systems. Most of the SCWR concepts have a pressure suppression type containment vessel like BWRs because SCWRs can have a very small containment vessel since steam generators are not used. In case of loss of coolant accidents, special attention may be paid to two-pass and three-pass cores because they delay refilling and reflooding of the core. Loss of feedwater flow accident should also be considered as it represents a specific difference compared to conventional WCRs. In a direct cycle SCWR plant, the feedwater works as a reactor coolant. The loss of feedwater flow means that reactor coolant stops to flow. The effectiveness of safety systems of the SCWR has been evaluated with system code analyses as usually this is the approach for conventional WCRs, and the safety of the concepts has been assessed.

3.2.6.2 *Materials and chemistry*

In-core and out of core components for SCWRs are required to have sufficient strength to avoid intolerable degradation or deformation at high temperatures. Major degradation mechanisms

of SCWR materials are general corrosion, environmentally assisted cracking such as stress corrosion cracking (SCC) and corrosion fatigue, and radiation embrittlement (IAEA, 2019a). Deformation can be induced mainly by void swelling and/or creep. From this point of view, zirconium alloys cannot be used as fuel cladding in SCWRs because these alloys do not have enough strength at high temperatures (i.e. the yield strength rapidly decreases at temperatures above 500°C) and also because they have low resistance to general corrosion (i.e. they corrode easily at temperatures above 400°C). In order to select SCWR materials, the effects of environment, stress and manufacturing processes need to be assessed.

It should be mentioned that the existing R&D methodologies used for current WCRs can be easily applied to the evaluation of SCWR materials. In addition, the experiences of water chemistry strategies in current WCRs and fossil fuel power plants can be used as a starting point for safe and reliable operation of SCWRs.

High temperature and pressures

The high operating temperature and pressure in SCWRs can induce a heavy stress on the structures. There is usually a maximum pressure that structures can handle at a given temperature without failure, defined by the bulk modulus of the structural material. But interestingly, this pressure threshold rises with temperature, and most structures in the power plant are designed to handle the operating pressure only at higher temperatures. Although structures can be designed in this way to withstand stresses without breaking, long-term phenomena like creep pose significant challenges (Baindur, 2008).

Irradiation

When a material is irradiated, all its physical properties can change. The specific conditions at the time of irradiation, such as temperature, local material composition, dose and dose rate, ultimately determine the property changes that will result. The irradiation-induced changes of greatest concern are (i) swelling, (ii) irradiation creep, and (iii) embrittlement.

Radiolysis and water chemistry

Two key chemistry issues have been highlighted for the SCWR:

- The first is the transport of corrosion products and impurities such as chloride from the feedtrain to the core. This is still ta major gap in supercritical water technology i.e. the lack of information on the magnitude of the problems of deposition of radioactivity in the external system and of the build-up of internal crud under irradiation. Moreover, it is well known that chloride deposition can lead to failure by SCC.
- The second issue is the fact that SCWR coolant will be subjected to an intense radiation field as it passes through the reactor core. Water radiolysis reactions resulting from the ionisation of water by the passage of gamma and fast neutron radiation (and b-radiolysis close to fuel cladding surfaces and fission fragment radiolysis from tramp uranium) leads to the formation of hydrogen, oxygen and hydrogen peroxide. Although subject to significant uncertainties, some simulations showed that very high concentrations of oxidants are possible in an SCWR core if water radiolysis is not controlled. Of particular concern is the behaviour of Cr under oxidising conditions; the passive films formed on almost all alloys being considered for in-core use in an SCWR are Cr oxides.

Chapter 4. Survey results

The survey consists of two main thematic areas of interest to analytical codes for the advanced reactors being considered: regulatory requirements and confirmatory analysis. In the following chapter an overview of existing codes will be provided. The capabilities of these codes compared to the relevant phenomena discussed in Chapter 3 and regulatory expectations will be discussed. Answers to the questionnaire were provided by Canada, France, Germany, Italy, Russia, the United Kingdom and the United States.

Regarding the answers from Germany and Italy, it should be pointed out that compared to other countries there will be no new application of nuclear power plants, including advanced concepts. Activities there are restricted to research reactors and waste management.

4.1 Regulatory requirements

Analytical codes and methods are commonly used to analyse the response of a complex engineering system to postulated events with potentially severe health, financial, and environmental implications. This part of the answers describes the requirements and expectations on the nuclear power plant designer for the development and use of analytical codes and methods in order to ensure the quality, credibility and confidence in the analyses produced by the analytical codes and methods. Starting with general aspects and oversight, codes assessment and user qualification are also discussed.

4.1.1 *General*

Question 1.1.1 – What regulations/laws are applicable to the development and use of analytical codes and methods? If applicable, what guidance documents are available to address these regulations/laws?

In general, requirements and expectations on the development and use of analytical codes and methods are derived from high-level principles associated with the applications of the codes to support the safety case and safety analysis. Most of these requirements are based on light water reactors (LWRs). These are for example REGDOC 2.4.1 "Deterministic Safety Analysis" and REGDOC 2.5.2 "Design of Reactor Facilities: Nuclear Power Plants" in Canada, the German SiAnf, the regulatory documents of the top level NP-001-15 and NP-082-07 in Russia, the ONR Safety Assessment Principles for Nuclear Facilities – AV.1 to AV.8 in the United Kingdom and several requirements from title 10 of the Code of Federal Regulations (CFR) in the United States (e.g. 10 CFR 50.34, 10 CFR 50.43(e), 10 CFR 50.46, Appendix B to 10 CFR 50, Appendix K to 10 CFR 50, 10 CFR 52.47, 10 CFR 52.79). In all these documents there is a general requirement to verify and validate that the analytical codes and methods used in the safety analysis and in systems important to safety. For reactor cores, more detailed requirements exist. Examples are the German KTA 3101.1 and 3101.2.

In addition, guidance (typically non-binding) may be provided such as in France the "ASN Guide No. 28 Qualification of scientific computing tools used in the nuclear safety case – 1st barrier", and in the United Kingdom the NS-TAST-GD-042 "Validation of Computer Codes and Calculation Methods". In the United States, requirements on QA procedures are described in 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Pre-processing Plants". These quality assurance programmes include, among other items, independent verification of the analyses, planned periodic audits of the quality assurance programme, and a corrective action programme to ensure that any deficiencies are promptly identified and corrected.

Question 1.1.2 – If applicable, what enforcement mechanisms are used to apply these regulations/laws?

The answers of Canada, France and Russia focus on the codes themselves. In Canada the validation status of codes is part of the licence application. In France, if the qualification is not fully demonstrated, a conservative approach can be required. In Russia all codes used in the design, safety analysis and support of the reactor operation must be certified. The answers of Germany and the United Kingdom describe the enforcement procedures. The United States underlines the routine oversight inspections by the regulator.

4.1.2 ***Oversight***

Question 1.2.1 – Are there regulatory efforts in progress to develop guidance for modelling and simulation of advanced reactors?

In general, the participating countries are not developing specific guidance for modelling and simulation of advanced reactors, although some reviews of the applicability of the existing guidance have been completed or are planned. Where such reviews have been completed, it is considered that existing guidance is sufficiently generic to be suitable for application to advanced reactors.

Question 1.2.2 – Are nuclear plant operators and designers required to obtain regulatory approval for their analytical codes and methods?

The regulators expect the licensee to provide evidence of sufficient validation and verification for codes and methods used to support a safety submission. This demonstration is greatly simplified for codes and methods already approved (e.g. in France and the United States).

Question 1.2.3 – If applicable, what process is typically used to obtain approval for analytical codes and methods? What strengths/weaknesses have been observed in this process?

For those countries that approve analytical codes and methods, this involves the submission of a report to the regulator, with the approval being linked to a particular usage case or range of applications. The approval processes can involve some degree of iteration, either by a staged process or regulatory feedback and requests for additional justification. The process also typically results in limitations and conditions (e.g. limited range) on the use of the code or method.

One of the key strengths of this approach is the efficiency, as the code can be approved in advance of a licensing application and approved codes and methods can be reused in the future, and can potentially be applied across a fleet of plants.

One of the key weaknesses of this approach is the difficulty in updating or modifying approved methodologies. Licensees may be reluctant to implement updates and improvements to approved codes due to the perceived risks associated with re-opening a regulatory review.

Question 1.2.4 – How do you cope with the potential bugs/errors in analytical codes and methods?

There is a general consensus that bugs and errors in analytical codes and methods are expected to be managed using a combination of code quality assurance procedures, validation and verification, and in some cases, sensitivity analyses and/or confirmatory analyses. Comparison of code results against experimental data is seen as an important tool in code validation and elimination of defects in the code. The United States has mandatory reporting requirements for code defects that could affect safety.

Question 1.2.5 – Do you require that nuclear plant operators and designers make their analytical codes and methods available for use by the regulator during licensing review?

France and Germany do not require the plant operators and designers to make their analytical codes and methods available for regulatory use. In all other participating countries access to the codes can be requested by the regulator if needed to support regulatory decision making. This is common practice in the United States, unless the applicant is using established technology, but uncommon in the United Kingdom and Canada, which instead typically use independent confirmatory analysis with alternative codes.

Question 1.2.6 – What requirements are set with regard to the quality assurance programme for analytical codes and methods?

All regulators have requirements or regulatory expectations related to quality assurance for analytical codes and methods. Some common features include the need for validation against experimental data, verification of the implementation of numerical methods and adequate code lifecycle management.

Question 1.2.7 – Analytical codes and methods used in the design and safety analysis of advanced reactors to date may not have been developed/maintained in accordance with a quality assurance programme appropriate for the intended application. What process, if any, is used to reconcile any quality assurance gaps?

In general, regulators expect that observance of existing good practices be demonstrated to support the validation, verification and quality assurance of legacy analytical codes and methods. Only the United States has a specific process for licensees wishing to adopt a legacy code into their QA arrangements and reconciling quality assurance gaps, known as "commercial-grade dedication".

4.1.3 *Code assessment*

Question 1.3.1 – Does the regulator have guidance on the sufficiency and adequacy of the data (e.g. scaling, range of conditions, number of cases) used to support the verification and validation of analytical codes and methods?

All regulatory authorities that responded to the question discussed some type of guidance available to support the verification and validation of analytical codes and methods. The specificity and depth of the guidance appear to vary by country.

Question 1.3.2 – What quality assurance requirements are imposed on the experiments and associated data used in model development and validation of analytical codes and methods?

All regulatory authorities except those of Germany stated that quality assurance requirements exist. The level of quality assurance required for these experiments appears to be consistent with the quality assurance used in the design and analysis of the plant. Only Germany reported no quality assurance requirements associated with experiments and the associated data.

Question 1.3.3 – What are accepted methods to assess uncertainty estimations (e.g. comparison with experimental data and operational measurements, comparison with exact analytical solutions, with precision codes, cross-verification with similar codes with estimated uncertainties, expert judgement)?

There is a general preference for comparisons with experimental data and operational measurements. However, there is some disparity in the answers regarding the acceptability of code-to-code comparisons for assessing uncertainty. Canada and the United States stated that code-to-code comparisons are not used to assess uncertainty. France and Russia stated that expert judgement is used in some instances. Germany and the United Kingdom responded that specific methods are not provided and that they are addressed on a case-by-case basis.

Question 1.3.4 – What value characterises the estimated uncertainty of the analytical code or method (e.g. the maximum deviation of the calculations from the results of measurements or reference calculations, the standard deviation)?

There is a general consensus that statistical analyses to obtain predictions at high confidence levels are acceptable (e.g. Wilks, 2-sigma) and preferred when sufficient data is available. There is also a general recognition that sufficient data may not exist to obtain statistical values with high confidence levels and a conservative approach (e.g. maximum deviation) can be used.

Question 1.3.5 – How is the uncertainty of the experimental data or the uncertainty of the reference code used for comparison taken into account when assessing the uncertainty of the code being tested?

There is a general consensus that the experimental uncertainty needs to be addressed when assessing the uncertainty in the code being tested. However, there is some disparity regarding

the method used to incorporate this uncertainty. Canada and Russia stated that the uncertainty is added to the code prediction in the square norm. The United Kingdom and the United States stated that the uncertainty treatment was dependent on the application. Like the United Kingdom and the United States, France stated that there is no unique or generic method.

Question 1.3.6 – How is the estimated code uncertainty taken into account in the safety analysis along with other sources of error (e.g. uncertainty of uranium enrichment, technological uncertainties in the manufacture of fuel rods and fuel assemblies, the uncertainty of determining and maintaining reactor power)?

There is a general consensus that uncertainty needs to be accounted for with a conservative bias for codes used in support of design basis analysis. France and the United States discussed uncertainty propagation approaches using sampling techniques to determine a suitably conservative value (e.g. 95% confidence at the 95th percentile). The responses also indicated that conservative biasing of the inputs is also an acceptable practice.

Question 1.3.7 – Are there specific aspects and methods used for uncertainty assessment in relation to passive functions and associated systems?

There is a general consensus that uncertainty assessment for passive systems is not conceptually different than for active systems although the range of parameters can be very different. France provided additional clarification that the application of the uncertainty methodology needs to be appropriately justified for passive systems and there are ongoing activities within the NEA and IAEA to addresses this topic. The responses of the United Kingdom and the United States indicate that differences in the phenomena or scale of the phenomena for passive systems may lead to different treatment of those systems in the safety analyses (e.g. chaotic behaviour and single-failure criteria).

Question 1.3.8 – Are there specific requirements on documentation of analytical codes and methods?

There is a general consensus that there are no specific requirements on the documentation of analytical codes and methods for advanced reactors. However, Russia clarified that there are specific requirements for the verification report, and the United States clarified that there are specific requirements for emergency core cooling system evaluation models for light water reactors that are generally applied for other analytical codes and methods. Both the United Kingdom and the United States stated that guidance on the documentation for analytical codes and methods is available.

Question 1.3.9 – What information should be included in the code approval/certificate documentation (e.g. range of applicability, estimated errors, requirements to users)?

There is a general consensus that a code approval/certificate needs to specify purpose and scope of the approval, range of applicability, and any applicable restrictions on the use of the code.

Question 1.3.10 – What are the accepted methods used to identify knowledge gaps and phenomenological modelling requirements in analytical codes and methods (i.e. what evidence can the regulator use to determine that the analytical codes and methods are adequate)?

No general consensus was identified in the responses. Canada, the United Kingdom and the United States indicated the use of expert judgement with the United Kingdom and the United States discussing a phenomenon identification and ranking process to identify potential gaps. Russia said that the main evidence for determining the adequacy of the code is contained in the code certification, while France explained that the methods to determine knowledge gaps and modelling requirements are handled on a case-by-case basis.

4.1.4 *User qualification*

Question 1.4.1 – Are there requirements for organisation (developer and user) qualification?

In general, it is expected, that code users are qualified. However, in Germany there is no formal request on the user itself. In Canada the personnel must be qualified and in France there are requirements inside the QA procedure. In Russia the user qualification is more on the

organisation level. The qualification is part of the certification process. In the United States personnel qualification records must be maintained as part of the quality assurance programme. The US NRC does not impose specific criteria regarding the necessary credentials/training for the user of an analytical code/method, but reasonable expectations include demonstrated competence with the analytical code or method through experience or training.

Question 1.4.2 – How are user choices factored in the consideration of uncertainties (e.g. selection of input deck parameters, models)?

The answers to this question are somewhat country specific. In Canada, France and the United States the user choices are expected to be covered by the V&V process. Furthermore, certain sensitivity analyses are expected. Russia takes only the final result of the safety case, with the selected choices. In case of doubts, the regulator can initiate confirmatory calculations or require additional calculations from the licensee. In the United Kingdom the development and qualification of reference input decks is expected. Variance from these should be controlled.

4.2 Confirmatory analyses

Confirmatory analyses are typically studies performed by the reviewer repeating the calculations presented by the applicant. Regulatory authorities or their technical support organisations may also use confirmatory analyses to investigate sensitivities to analysis inputs or to examine accident sequences not presented in a license application. This is somewhat different to independent verification of deterministic safety analysis by the licensee (cf. Requirement 21 of GSR Part 4 [IAEA, 2016]) that is provided by the applicant. The independent verification should consist of two main parts: an overall (qualitative) review focused on the quality and comprehensiveness of the safety analysis; and specific detailed reviews of important aspects of the analysis.

Confirmatory analyses also support understanding the results of the applicant's analyses and provide additional confidence. Confirmatory analysis helps to understand the proposed design and assess its compliance with regulations. Additionally, confirmatory calculations are considered when performing sensitivity analyses and can also provide evaluations of the impact of uncertainties.

4.2.1 *Reasons for performing confirmatory analysis*

Question 2.1.1 – To what extent is confirmatory analysis performed and what factors into the decision to conduct confirmatory analyses?

Confirmatory analyses are not always performed during the regulatory assessment; this is decided on a case-by-case basis. The strategies or reasons for deciding to carry out confirmatory analysis include:

- to confirm key design values;
- to check the available margins when they are limited;
- to check the results when safety issues are important;
- to maintain competences in capabilities to perform calculations;
- to follow a graded approach and focus on important issues;
- to help understand the design**,** integrated system performance, and analytical methods;
- to perform additional sensitivity studies;
- to assess the approach when a first-of-a kind design is proposed (or new for the reviewing staff).

Question 2.1.2 – What alternatives to confirmatory analyses are available to support regulatory decisions?

Regulatory findings are based on the analyses and data submitted by the applicant and not upon confirmatory analyses. The process of requesting additional information and answers from an applicant generally involves supplementary information. It is also suggested to compare the results with similar cases previously reviewed and to check for consistency using engineering judgement.

4.2.2 *Analytical tools and methods for confirmatory analysis*

Question 2.2.1 – To what extent are confirmatory analyses required to be independent (e.g. is it acceptable for the regulator to perform sensitivity analyses using the same code as the nuclear plant operator or designer)?

The use of different analytical codes and methods is sometimes required (depending on the country's regulation) or necessary because the tools used by the applicant are not available for the reviewer. In the latter case, some countries' regulations require that the applicant make available the analytical methods to the reviewer. There are drawbacks and benefits to both approaches. Using the same tool provides confidence in the correct use of the tool (when results from the applicant and confirmatory analysis are similar) and therefore can help to detect any flaws in the preparation or execution of the calculation (when discrepancies exist between applicant and reviewer). A side-by-side comparison is also possible when the same tools are used. However, this implies that the reviewer has a sufficient knowledge of the applicant's tool. On the contrary, when the tools are different, similar results can provide a higher degree of confidence. When results are different, it may be challenging to identify the sources of discrepancies.

4.2.3 *Quality assurance for confirmatory analysis*

Question 2.2.2 – What levels of quality assurance and confidence are imposed on the analytical codes and methods used in confirmatory analysis by the regulator in comparison to the applicant?

The approach to quality assurance (and by extension qualification of analytical codes and methods) is also different depending on a country's regulations. On the one hand, for both the applicant and reviewer, the qualification of analytical tools is required. On the other hand, some countries require only qualification of the applicant's tools because they are the bases for safety demonstration, and there are no specific requirements for the confirmatory analyses' tools.

4.3 Phenomenology and associated analytical needs

Based on the discussion of safety-relevant phenomena in Section 3.2 for the different Generation IV concepts, the specific demands made of analytical codes and methods can be preliminarily derived. The phenomena identified below are intended to be applicable to the different Generation IV concepts. However, the identified phenomena should not be considered comprehensive or applicable to all reactor designs associated with the referenced concept.

In the following, a high-level categorisation of the relevant phenomena is proposed in an attempt to exploit commonalities to highlight the potential transferability of approaches, methods or even codes among the different Generation IV concepts.

4.3.1 *Reactor physics*

All the Generation IV concepts share the need for reactor physics analysis, with the purpose of assessing the neutronic behaviour of the core not only in normal operation but also in accidental conditions. Besides the methods requested for general purpose analyses, it is worth stressing that all the concepts share the need for a level of detail compatible with capturing localised effects that might show up and introduce additional specific challenges to the system during transient events.

All concepts but the MSR also share similar phenomena – provided for the general differences that can be anticipated, e.g. the differing neutron lifetimes between thermal and fast spectrum systems. For the MSR, the circulating nature of the fuel introduces specific phenomena, like the transport of delayed neutrons precursors, which makes neutronic analysis for these systems substantially different from that of any other solid-fuelled concept.

Differences may arise also in terms of numerical methods requested because of the reactor core configuration: fuel bundles (i.e. for SFR, GFR, LFR and SCWR), dispersed fuel microparticles (i.e. TRISO in [V]HTR), or homogeneous fuels (i.e. for MSR) pose different grades of priority, hence needs of accuracy of solution, to self-shielding and (where applicable) moderation phenomena.

The use of non-standard fuels, like minor-actinide-doped ones for transmutation purposes in fast reactors, or the molten salts themselves, notably for thermal-spectrum systems, adds specific needs in terms of nuclear data (ND), e.g. cross-section data that represent the effect of molecular bonds in molten salts, among others. Specific ND might be needed also in case of transition phase studies (Ivanova, 2017).

High enriched fuel is anticipated in SFR, LFR, GFR and VHTR. That would require enhanced capabilities to simulate criticality events with irradiated and fresh fuel.

4.3.2 *Core thermal-hydraulics and thermo-mechanics*

Among the most critical safety limits characterising the design of any reactor, thus including all Generation IV concepts, are those that apply to the components and structures of the reactor core, which are typically of a thermal-hydraulic and thermo-mechanical nature.

The thermal-hydraulic behaviour of the core components must be understood and simulated to establish the critical safety limits and safety margins that characterise the design of a reactor. All of the Generation IV concepts rely upon the core thermal-hydraulics to transfer heat during normal operation and to remove decay heat during shutdown and hypothetical accident scenarios.

In some of the Generation IV designs the thermal-hydraulics have important effects on the neutronic behaviour of the core and in the thermo-mechanical response of core and vessel components. In a fast reactor the thermal expansion can result in neutron leakage and thus affect reactivity and core power. For some scenarios, the thermal-hydraulics, neutronics, and thermo-mechanics of the core may require a coupled simulation.

While it is well understood that the thermal-hydraulics (and to some extent the thermo-mechanics and neutronics) cannot be resolve apart from the rest of the system, there are some configurations and safety issues that may require dedicated methods and code capabilities, such as simulation of fuel assembly blockage or subchannel flow distributions.

While the methods for thermo-mechanical analysis of core structures and components are rather standard and common among the Generation IV concepts, for thermal-hydraulics different mechanisms – notably for decay heat removal – may have different importance, e.g. radiation heat transfer in VHTRs, which must be reflected in the validation of the codes and methods used for the analysis.

4.3.3 *Fuel performance*

Analytical methods are generally needed to assess fuel performance under conditions of normal operation, including anticipated operational occurrences, and accident conditions. The analytical methods need sufficient modelling capabilities to capture fuel geometry, material properties and physics sufficient to capture degradation mechanisms and failure modes. The specific degradation mechanisms and failure modes are generally specific to each fuel type and should be identified during fuel qualification. Some identified phenomena associated with Generation IV fuel designs are discussed below.

Fuel swelling, fission gas release, creep, stress/strain, chemical interactions with the reactor environment (e.g. coolant) and neighbouring materials (e.g. fuel/clad chemical interaction), and changes in thermal-physical properties with exposure (e.g. thermal-conductivity degradation) are generally expected to be addressed by fuel performance codes for solid fuel designs. Higher burnups anticipated for the Generation IV concepts are expected to introduce specific modelling needs that must be reflected in fuel performance codes. This might pose challenges to the validation of these models. Metal fuel is susceptible to the formation of low melting-point eutectics, which could contribute to cladding failure under accident conditions (Bauer, 1987). Fuel performance modelling for TRISO fuel is a major consideration for determining fuel failure fraction and contribution to accident source term. Unknowns such as kernel migration and silicon carbide morphology modelling and the impact on radionuclide release have been identified. For MSRs, the change in salt chemistry associated with fission product building and the resulting impact on thermophysical properties has been identified as an important phenomenon that is unique to reactors utilising fuel salts (ORNL, 2018). Wherever advanced fuels are considered (e.g. nitrides or carbides) ad hoc models need to be developed and validated.

4.3.4 *System thermal-hydraulics and thermo-mechanics*

System thermal-hydraulics modelling in the context of innovative systems has similar objectives as in the context of safety studies and assessment for the reactors of the 2nd and 3rd generations. The only difference might be in the enhanced safety requirements imposed for the innovative systems of the 4th generation as well as for small modular ones. According to modern tendencies and requirements, and in the light of the lessons learnt from the Fukushima Daiichi accident, a broader use of passive heat removal capabilities in all incidental and accidental conditions is considered. This is why the advanced system thermal-hydraulics tools could be required to be able to perform in a credible manner the best estimate simulations of passive heat removal processes, such as long-term and natural convection processes.

Some Generation IV designs involve “pool type” configurations in which natural circulation through components internal to the reactor vessel transfer heat from the core. Simulation of flow in this type of configuration can be complex and should be validated using well-scaled integral test facilities. For designs that include large volumes of coolant, simulation of thermal stratification, thermal striping, gas entrainment and turbulent mixing can be difficult for thermal-hydraulic codes. Capabilities similar to those in computational fluid dynamics (CFD) codes may need to be built into these codes to accurately simulate thermal stratification and mixing.

System thermal-hydraulics are important to a wide range of design and safety considerations in all of the Generation IV designs. Several of the designs utilise secondary or tertiary loops to remove heat from the system, and the performance of these loops can have significant effects on system performance. Decay heat removal may be accomplished by several means: reactor cavity cooling systems, decay heat removal systems, or cooling by heat loss to the surrounding containment or confinement. System thermal-hydraulic codes must be capable of simulating these heat removal mechanisms and the components (pumps and heat exchangers) that are part of the secondary loops. In addition, when heat removal is accomplished by “passive systems”, the thermal-hydraulic code must be able to simulate the balance between small driving heads and the resistance through a flow path.

It seems rational to expect the system simulations to provide essential inputs that could help, inter alia, in DEC-A and -B assessments.

The analytical codes and methods for the thermo-mechanical analysis of the reactor system structures and components should provide for models capturing the phenomena expected in the specific environmental conditions determined by the investigated Generation IV technology, such as temperature, radiation level and spectrum, or chemistry. High-temperature creep, high-radiation embrittlement and corrosion are some examples of phenomena that introduce specific modelling needs and might pose challenges to the validation of the associated models.

4.3.5 *Containment/confinement analysis*

Whatever the Generation IV concept considered, the need exists for an analysis of its containment and/or confinement capabilities. This, in turn, involves the simulation of all those failure and degradation mechanisms, and of all the consequent release paths – including the associated retention provisions – that are pretty much specific to each concept, or even design.

Like for LWRs, most of the simulation codes applied for the simulation of phenomena inside containments/confinements are based on a so-called lumped-parameter (LP) concept (e.g. as used in MELCOR, ASTEC, SOKRAT). This concept is suitable for the simulation of temperature and gas distribution. A second major task is the simulation of aerosol and radionuclides behaviour. The main processes are agglomeration of aerosols, considering a possible hygroscopy, and deposition of aerosols. The specific characteristics of dusts (e.g. graphite shape factor) should be considered. Possible toxicity and the radioactive decay should be considered, e.g. the aerosol ageing (reaction of sodium aerosols). The reaction of the coolant with the containment surface requires specific modelling, similarly to the core concrete interaction, in case of ex-vessel scenarios. Furthermore, a release of fission products out of the coolant has to be modelled. The existing thermal boundary conditions inside the containment have to be considered. Regarding SFR concepts, possible sodium fires (pool and spray fires) require specific models. Like in typical fire scenarios, a hot gas layer formation might be possible. In case of pressure transients, a possible resuspension of aerosols (e.g. graphite dust in VHTRs) must be modelled.

In containment/confinement analysis, a few systems must be simulated. In the case of VHTRs using a confinement, these are venting filters. Other systems include passive reactor cavity cooling systems (RCCS). Several concepts rely on passive heat removal via the cavity.

4.3.6 *Material specificities*

Be it for verification of normal operations or accident conditions, all analytical codes and methods have to properly represent the characteristic phenomena introduced by the specific materials in use, and by their anticipated operative conditions (normal and abnormal). Generation IV reactor concepts use materials in environmental and radiation exposures that introduce modelling changes. Analytical codes and methods for all reactor types need to be capable of accommodating uncertainty associated with the thermal-physical properties (methods for treating uncertainty are discussed in Section 4.1.3). Of the material modelling issues discussed in Chapter 3 of this report, chemical interactions and dimensional changes are highlighted as areas presenting challenges to modelling and simulation. Some concept specific phenomena are highlighted below:

- The phenomena of creep/fatigue interaction and thermal ageing have been identified as prominent issues for SFRs.
- Age-related phenomena for high-temperature materials are a significant source of uncertainty is establishing safety limits for structural components in VHTRs.
- Thermal-physical property and dimensional changes are associated with the irradiation of graphite impacts in VHTRs and MSRs.
- Corrosion, fission product plate-out and condensation can be a source of uncertainty impacting the structural integrity and source term analyses for MSRs.
- Corrosion, embrittlement and erosion contribute to uncertainty in safety-limits for LFR structural components.

Additionally, uncertainty in thermal-physical properties may have a significant impact on predictions for passive heat removal system performance (e.g. natural circulation flow). Analytical codes and methods need to be capable of incorporating this uncertainty in their predictions of system performance.

4.3.7 *Coupled and multi-physics simulations*

Following the ideas of a graded safety assessment approach (IAEA, 2016) and taking into account fundamental limits in available feedback experience for innovative nuclear designs, it may be relevant to underline the heavy reliance on fully mechanistic modelling in nuclear technological science as well as in design and safety assessment. Furthermore, the regulatory environment in several countries presumes using risk informed methodology, including the Best Estimate Plus Uncertainty (Bruna, 2018). This methodology requires best-estimate simulations, i.e. high-fidelity modelling of the complex processes and phenomena both in nuclear power systems, as such, and in relevant fuel cycle facilities.

Achievements in the practice of multi-physics simulations indicate the possibility that coupled modelling lies closer to physical reality than separated stand-alone modelling. In such a context, multi-physics tools look promising with respect to the predictive capabilities of simulations covering a larger range of applications and states.

However, their maturity seems to be questionable because of scarce relevant experimental databases and weakness in the available validation methodology (Avramova and Ivanov, 2020). This is the case if the need to provide evidence-based validation is taken into account.

At the same time, as discussed above, the most frequently used simulation platforms are based on an explicit coupling of the various single-physics and single-domain modules that could be elaborated and validated within stand-alone applications. Accordingly, the validation of an entire coupled model could be facilitated, although this does not solve the problem of how to validate the effect of coupling itself.

Whenever actual multi-physics simulation is called for in the analysis of Generation IV systems, the need for relevant experiment-based benchmarks will be stressed, adding significant complexity to the common practice of securing the required data quality and consistency.

Chapter 5. Conclusions

Analytical codes and methods are used extensively in the design and safety analysis of nuclear reactors. These codes and methods are commonly used to analyse the response of a complex engineering system to postulated events with potentially severe health, financial and environmental implications. This technical report developed by the NEA Working Group on the Safety of Advanced Reactors (WGSAR) discusses the regulatory requirements imposed by regulatory authorities upon reactor designers and operators that use analytical codes and methods for design and analysis. Furthermore, it discusses the use of analytical codes and methods by regulatory authorities and/or their technical support organisations as a confirmatory analysis tool.

The results are based on a survey that was answered by participants in Canada, France, Germany, Italy, Russia, the United Kingdom and the United States.

The WGSAR has defined common positions underlining the need for verification and validation, the estimation of uncertainties and configuration management (Section 2.1). However, differences between countries exist. Examples include the certification of codes and expectations of user qualification. No general consensus was identified in the responses regarding methods used to identify knowledge gaps and phenomenological modelling requirements.

Furthermore, common positions related to confirmatory analysis are drawn (Section 2.2). The expectations of codes and methods are independent of their application on Generation III or Generation IV type plants. However, within the licensing process of Generation IV reactors, simulation codes and methods must be available and must be able to simulate the regarded safety-relevant phenomena, which are partly specific to Generation IV concepts.

An overview of safety-relevant phenomena is provided for each of the six reactor concepts covered by GIF (Section 3.2). Regarding the main topics (such as reactor physics, core thermal hydraulics and so on) the demands made on simulation codes are described briefly (Section 4.3). The development and qualification of codes and methods should be a point of focus in Generation IV reactor development projects.

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Glossary

This report uses these terms as defined by the *IAEA Safety Glossary* (IAEA, 2018b).

Accuracy

In this context, the known bias between the prediction of a *system*[1] *code* and the actual performance in transients of a *facility*.

System code validation

Assessment of the accuracy of values predicted by the *system code* against relevant experimental data for the important phenomena expected to occur.

Model verification

The *process* of determining whether a *computational model* correctly implements the intended *conceptual model* or *mathematical model*.

System code verification

Review of source coding in relation to its description in the *system code* documentation.

1. The concepts introduced here and in all definitions in the glossary are valid also for codes beyond system codes.

Appendix A. Country responses to survey for analytical codes and methods for advanced reactors

This chapter contains the responses of each country. Answers to the questionnaire were provided by Canada, France, Germany, Italy, Russia, United Kingdom and United States. The answers to the first part of the questionnaire related to regulatory aspects are combined in tabular form to facilitate comparisons (Table A.1).

The answers related to simulation codes are combined for each listed code and method to avoid repetition.

Table A.1: **Country answers related to regulatory aspects**

Canada	France	Germany	Italy	Russia	United Kingdom	United States
1. Regulatory Requirements						
1.1 General						
Question: What regulations/laws are applicable to the development and use of analytical codes and methods? If applicable, what guidance documents are available to address these regulations/laws?						
REGDOC 2.4.1 "Deterministic Safety Analysis", REGDOC. 2.5.2 "Design of Reactor Facilities: Nuclear Power Plants"; guidance included within the REGDOCs.	Any scientific calculational tool must be qualified for its use in a safety analysis. For the evaluation of the first barrier integrity, a guidance document (non- binding) is available "ASN Guide No. 28 Qualification of scientific computing tools used in the nuclear safety case – 1st barrier".	In Germany the general regulations are formulated in the "Safety Requirements for Nuclear Power Plants" (Edition 3/2015) (referred below as "German SiAnf"). There are few general requests on the application of analytical codes and requirements on the quality (validation) of these codes. Further generic requirements are given in KTA 1401. For the reactor core requirements are provided in KTA 3101.1 and 3101.2. Further requirement related to criticality are given in DIN 25478:2014.	As a result of the Chernobyl accident and the referendum vote that followed in 1987, Italy decided for the closure of its nuclear power plants. The only remaining nuclear activities are related to the small research reactors still in operation (e.g. TRIGA, TAPIRO) and the decommissioning of the power plants. In the 1980s, the use of computer codes to support the design and licensing was very limited and not regulated with laws or even guidelines. After Chernobyl, the adoption of regulations/laws by the regulatory body became not relevant.	Article 26 of the Federal Law "On the Use of Atomic Energy" No. 170-FZ requires that the analytical codes and methods used to justify the safety and actual state of NPPs must be examined in the scientific and technical support organisation of the Russian regulator. The procedure for this examination is established by the authorised Russian body of state safety regulation («Rostekhnadzor»). Currently, the scientific and technical support organisation of «Rostekhnadzor» is SEC NRS. In the regulatory documents of the top level NP-001-15 and NP-082-07 there is a general requirement that the analytical codes and methods used in the safety analysis and in systems important to safety must be verified and certified according to established procedures. The guidance RD-03-34-2000 sets out the requirements for the content of the verification report of analytical codes used to justify the safety of nuclear facilities (under revision now). In addition, there is a special guidance RB-061-11 for the verification and examination of neutronic analytical codes and methods.	The law governing licensing of nuclear facilities requires license conditions to be imposed. These cover: the requirements for a suitable safety case, for training and qualification of staff and for the keeping of records. ONR expectations related to assurance of validity of data and models are captured in the ONR Safety Assessment Principles for Nuclear Facilities – AV.1 to AV.8 and further guidance for ONR Inspectors is available in NS-TAST-GD-042 Validation of Computer Codes and Calculation Methods.	There are specific requirements for LWR LOCA analysis identified in 10 CFR 50.46 and 10 CFR 50, Appendix K. Although these specific requirements are not directly applicable to non-light water reactors, most of the requirements identified for LWR LOCA are reflected in the generic guidance document, RG 1.203, "Transient and Accident Analysis Methods." Additionally, the development and use of analytical codes in the design of a nuclear power plant have quality assurance requirements. Further discussion of these requirements is below. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," contains specific requirements for LOCA. All applicants for a construction permit, operating license, early site permit, design approval, design certification, or manufacturing license are required to have a quality assurance programme. 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Pre-processing Plants," establishes the quality assurance requirements for the design, manufacture, construction, and operation of those structure, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. These quality assurance programmes include, among other items, independent verification of the analyses, a corrective action programme to ensure that any deficiencies are promptly identified and corrected, and planned periodic audits of the quality assurance programme.

Table A.1: **Country answers related to regulatory aspects** *(cont'd)*

Canada	France	Germany	Italy	Russia	United Kingdom	United States
Question: If applicable, what enforcement mechanisms are used to apply these regulations/laws?						
All codes used in safety analysis and design must be validated – this is a requirement subject to verification at license application.	When qualification is not fully demonstrated, conservatisms can be required.	The latest version of the "Safety Requirements for Nuclear Power Plants" and the accompanying interpretations have been published in the Federal gazette on 3 March 2015 (BAnz AT 30.03.2015 B2 and BAnz AT 30.03.2015 B2). Following consultation with the Länder, the federal authority publishes official notices (in the form of requirements, rules, guidelines, criteria and recommendations). These are usually provisions agreed in consensus with the competent licensing and supervisory authorities of the Länder regarding uniform application of the AtG. The federal announcements describe the view of the federal nuclear regulatory authority on general questions of nuclear safety and administrative practice and serve the Land authorities as orientation in the execution of the AtG. They are implemented by the competent Land authorities in licensing procedures or in connection with supervisory activities performed on their own authority. This also ensures that the AtG is enforced in the various Länder as far as possible to similar standards. For the licensees, these sub statutory regulations become binding by their inclusion in licenses or supervisory decrees.	n/a	All codes used in the design, safety analysis and to support of the reactor operation must be certified – this is a requirement subject to verification at licensing process.	The enforcement is proportionate to the magnitude of the legal and safety shortfall. In the extreme, this would take the form of refusal of a permission or the issuing of an enforceable improvement notice. The licensee's codes and methods may be assessed against the aforementioned regulatory expectations during a permissioning activity (e.g. construction of a new facility, a high category safety case modification), or during a periodic review of safety. Early engagement, persuasion and influencing are used in the first instance, with other enforcement actions such as withholding consent to proceed with construction available if necessary.	Operating reactors and reactors under construction undergo routine oversight inspections by the regulator to ensure that the plant is built and operated consistent with the licensing basis. Additionally, quality-assurance inspections are typically performed as part of a design review to ensure that supporting analyses are performed consistent with the approved methodologies.

Table A.1: **Country answers related to regulatory aspects** *(cont'd)*

Canada	France	Germany	Italy	Russia	United Kingdom	United States
1.2 Oversight						
Question: Are there regulatory efforts in progress to develop guidance for modelling and simulation of advanced reactors?						
No. This is part of the safety analysis and V&V requirements, which are generic.	In France, there are no activities on development of guidance for simulation of advanced reactors.	No	No	No. The licensee himself determines the scope and methods of modelling and simulation of advanced reactors. In the licensing process, the regulator assesses the adequacy of these methods and the completeness of the studies.	We issue guidance to assessors in the form of published Technical Assessment Guides. We also publish papers from time to time (for example at the Top Fuel conference). A review of existing regulatory guidance to identify any necessary revisions or gaps which need to be addressed, for the potential future assessment of advanced reactors, is being considered for Phase 2 of the UK Government's advanced modular reactor (AMR) initiative, subject to financial approval.	The NRC is not developing a guidance document for modelling and simulation of advanced reactors. Based on recent engagement with potential applicants, the NRC believes that the process outlined in existing guidance documents (i.e. RG 1.203) is generic and adequate.
Question: Are nuclear plant operators and designers required to obtain regulatory approval for their analytical codes and methods?						
They are required to provide verifiable evidence of V&V.	Yes	No	As any design being developed is meant for foreign countries, obligations apply for designers as set by the respective authorities.	Yes	No, ONR does not approve analytical codes and methods. ONR does not approve codes. ONR permissions activities, which will have safety cases that may be supported by codes. The level of assessment the analysis receives is based on a graded approach, dependent on the magnitude of the safety margin (compared to the uncertainty), the novelty and the risk of the activity. We permission on a targeted sampling basis, so the use of a code to support previous activities does not set a precedent.	Acceptable methods are required to be used to establish the design basis and in the safety analyses. These methods can be approved within a licensing document (e.g. final safety analysis report (FSAR)) or through a separate document that is incorporated by reference into the FSAR. Changing a method used in establishing the design basis or in the safety analyses would require a license amendment request.

Table A.1: **Country answers related to regulatory aspects** *(cont'd)*

Canada	France	Germany	Italy	Russia	United Kingdom	United States
Question: If applicable, what process is typically used to obtain approval for analytical codes and methods? What strengths/weaknesses have been observed in this process?						
V&V process is described in the quoted REGDOCs	Authorisation to use new (i.e. not previously approved) analytical codes and methods should be transmitted to the safety authority. File prepared by designer/operator is reviewed and at the end of the process, additional justifications or demonstrations can be required, and/or use can be limited or penalisations imposed. Weaknesses: in general experimental data base is insufficient and conservatisms do not fully cover the identified uncertainties and deficiencies in models and calculations.	n/a	For national activities, no regulator approval is required, thus qualified analytical codes and methods are used in their own application domain. For foreign activities, where regulator approval is required for analytical codes and methods, new codes and methods are developed in Software Quality Assurance (SQA), and both new and existing codes are validated for use in the anticipated application domain through testing against Separate and Integral Effect Test experimental data, and/or benchmarked with other codes, and the validation process and results (including uncertainty quantification) documented in dedicated Validation Dossiers.	On the basis of the SEC NRS, an Expert Council for the certification of analytical codes was created. Delegates of more than 50 scientific, design and operating organisations participate in this council. The Expert Council includes the Presidium and 7 topical sections in the following areas: - physics of nuclear reactors and nuclear safety; - thermos-hydrodynamics and multiphysical processes; - transfer of ionising radiation, radiation protection, migration of radioactive substances; - strength, resource of elements, equipment and systems; - strength and reliability of building structures; - probabilistic safety analysis, reliability of systems and facilities; - physical chemistry, geochemistry and hydrogeology. Depending on the application area and complexity of the analytical code, it is accepted for consideration by one or more sections. At a section meeting, experts (section members and invited experts) from different organisations independent of the code developer are appointed. The task of experts is to analyse the verification report. The results of this analysis (expert conclusions) are considered at a section meeting and a decision is made on the possibility or impossibility of code certification, a certificate draft is being prepared. The conclusion of the section and the certificate draft are submitted to the Presidium of the Expert Council for a final decision. A strength of this process is that as a result we get a more realistic assessment of the application area and the code uncertainties than the assessment of the code developers. Weaknesses: Lack of operation experience and insufficient volume of experimental data for new generation reactors, such as lead-cooled reactor including the behaviour of nitride fuel in emergency conditions.	Not applicable, ONR does not approve analytical codes and methods.	The typical process to obtain approval is the topical report process. This process involves submitting a stand-alone report that addresses a topic generically (e.g. the use of a specific analytical code for safety analyses). The strengths of the topical report process include: - Efficiency and a staged review process – topical reports can be submitted prior to a plant licensing application. Once approved, the methodology can be referenced in a plant licensing application. - Generic application across several plants – the topical report process is commonly used by plant vendors who wish to apply a generic process to several operating plants/customers. - Reduces regulatory risk on specific topics – the topical report process can provide a regulatory decision on topics involving regulatory uncertainty for first-of-a-kind approaches to safety issues. The weaknesses of the topical report process include: - Difficult to update/modify approved methodologies – applicants and licensees with existing approved methodologies have been hesitant to submit updates over concerns of "reopening" a safety review due to the associated costs and potential for the regulator to identify concerns in the already approved methodology. - Generation of additional requirements – the review of a topical report typically results in the generation of limitations and conditions on the use of that methodology. Compliance with these limitations and conditions requires additional tracking.

Table A.1: **Country answers related to regulatory aspects** *(cont'd)*

Canada	France	Germany	Italy	Russia	United Kingdom	United States
Question: How do you cope with the potential bugs/errors in analytical codes and methods?						
This is handled via mandatory code QA procedures.	During the review, sensitivity analyses and comparisons with experiments (or other analytical methods) are required. If sensitivity analyses and comparisons are covering the whole range of analytical code, probability for an "error free" code is improved. A progressive validation (on separated effects tests then on integral tests) is also required to minimise bugs/errors in analytical codes and methods.	No procedures	In case the SQA is enforced for the analytical code or method, the bugs/errors are reported and analysed, and the required corrective action estimated. Depending on the severity of the bug/error, the effort requested by the correction and the availability of the source code: 1) the nonconformance is reported and all users warned (minor bugs/ errors not affecting the integrity of the results); 2) the nonconformance is communicated to the developer requesting for correction (major bugs/ errors and i. no source code available, or ii. source code available); 3) the nonconformance is corrected in-house, the changes are reported and verified, a new code/method baseline is developed, validated and deployed (major bugs/errors and source code available).	As indicated in RD-03-34-2000, a verification matrix should be provided in the verification report. This is the table in which should be indicated: - operational modes, transients and states or types of impact on the object, which is modelled by the considered analytical code; - physical phenomena or processes occurring under these transients or states; - experimental or calculation tests that were used to verify the code in the above modes and conditions. A justification should be provided for the sufficiency of the information included in the verification matrix to confirm the adequacy of the code in the stated scope. A sufficient amount of test calculations allows us to realistically evaluate the uncertainty of the calculated results and take this uncertainty into account in the safety analysis.	We expect the licensee to have code life cycle management processes and we expect them to operate a safety case anomalies process. ONR would expect the licensee to demonstrate how potential bugs/errors in analytical codes and methods are handled to ensure they do not negatively affect the safety justification. An ONR inspector may choose to instruct a Technical Support Contractor (TSC) to perform independent confirmatory analysis to support their assessment of a licensee's safety submission, on a sampling basis.	Potential bugs/errors are addressed primarily through the quality assurance programme. This programme tracks the identification and resolution of any deficiencies. The quality assurance programme is monitored by NRC inspectors. Additionally, there are reporting requirements associated with any defect (including bugs/errors) that could result in a potential safety hazard. However, the approval process for analytical codes and methods includes a code assessment of a frozen code version, where the code (including any unknown bugs/errors) is compared against experimental data. Due to the robustness of the code assessment process, the reporting of defects associated with code bugs/errors is rate. It is noted that 10 CFR 50.46 has specific reporting requirements for bugs/errors identified in LWR LOCA evaluation models. These include: (1) reporting within 30 days, a change to or error in the evaluation model that results in a calculated peak clad temperature different by more than 50°F (27.8°C) from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F (27.8°C) , and (2) annual reporting of each change to or error that affects the temperature calculation.
Question: Do you require that nuclear plant operators and designers make their analytical codes and methods available for use by the regulator during licensing review?						
We can make such request should we decide to do so. However, it is considered more efficient to conduct confirmatory analyses using alternative codes.	No	No	n/a in Italy. It could be required by the regulators of other countries in those cases where ENEA is part of the design pool.	No. We don't have such a requirement. Regulator uses own independent codes domestic or foreign.	If this is required, but it would be the exception. ONR do not run codes ourselves, and when commissioning a TSC to perform confirmatory analysis, we would typically expect a different analysis code to be used. ONR would expect the licensee/requesting party to share the relevant data (or input deck) to enable the confirmatory analysis to adequately represent the plant under assessment.	The NRC typically requires that applicants make their analytical codes and methods available during a review. This request may not be necessary for applicants using well understood technology. However, for advanced reactor work, the NRC anticipates requesting that the applicants make their analytical codes and methods available to the NRC during the review.

Table A.1: **Country answers related to regulatory aspects** *(cont'd)*

Canada	France	Germany	Italy	Russia	United Kingdom	United States
Question: What requirements are set with regard to the quality assurance programme for analytical codes and methods?						
The requirement is for the vendor to have a proper QA for the codes. CNSC will conduct assessment of such QA process at the license application (this may include an audit).	Usual QA is required, in particular verification is required to determine whether the equations are solved correctly from both a numerical and data processing viewpoint. Verification concerns numerical methods and algorithms, their implementation, the data flow diagrams, the architecture of the IT programmes and compliance with any programming rules defined by the licensee.	KTA 1401 (2017-11) General Requirements for the Quality Assurance.	No specific requirements by national regulation. Foreign regulations apply whenever requested. In general, ISO 9001, ISO/IEC/IEEE 90003 and all other ISO, IEC and IEEE standards related to software quality, are used to enforce the guidelines set out by IAEA in the Specific Safety Guide SSG-2.	We have a general regulatory document NP-090-11 "Requirements for quality assurance programmes of nuclear facilities" which contains inter alia item 22 "Quality assurance of analytical codes and methods". According to this item, a description of the current procedures for ensuring the quality of analytical codes and methods, including their verification and certification, should be provided.	These would depend on the safety class. Lifecycle management would follow a model appropriate to a graded approach to safety. There are standards on software quality and we would expect the relevant standard to be followed. ONR SAP AV.4 states: "Computer models and datasets used in support of the safety analysis should be developed, maintained and applied in accordance with quality management procedures. These procedures should identify measures and controls to provide confidence that calculations are undertaken without error, to a level commensurate with the importance to safety of the analysis being performed. The procedures should, where appropriate, address code and dataset verification, version control, testing, documentation, user qualification requirements and training, peer review and endorsement. The procedures should specify independent verification of computer codes and datasets to confirm consistency with the supporting documentation. The process of inputting data into a model should be independently verified."	All applicants for a construction permit, operating license, early site permit, design approval, design certification, or manufacturing license are required to have a quality assurance programme. 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Pre-processing Plants," establishes the quality assurance requirements for the design, manufacture, construction, and operation of those structure, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. The development and use of analytical codes and methods falls under this requirement. These quality assurance programmes include, among other items, independent verification of the analyses, a corrective action programme to ensure that any deficiencies are promptly identified and corrected and planned periodic audits of the quality assurance programme. It is common for applicants and licensees to have a software and data quality assurance programme that includes development and modification to analytical codes and methods.

Table A.1: **Country answers related to regulatory aspects** *(cont'd)*

Canada	France	Germany	Italy	Russia	United Kingdom	United States
Question: Analytical codes and methods used in the design and safety analysis of advanced reactors to date may not have been developed/maintained in accordance with a quality assurance programme appropriate for the in-tended application. What process, if any, is used to reconcile any quality assurance gaps?						
Again, CNSC requires V&V and QA process to be established for the codes used in design and safety analysis. Lack of V&V is a barrier to licensing.	This is not considered. Designer/operator should provide QA meeting the current requirements, especially for new designs.	n/a	Independent validation is performed – as much as practicable – for the assessment of the applicability domain and the quantification of the credited uncertainties. Configuration control and use of analytical codes and methods in SQA is then enforced.	The main document for the Russian regulator is an analytical code certificate. During the certification process QA is usually not verified. This is the concern of the designer and operator.	Software quality-assurance procedures are only one aspect of building confidence. Where code is open source, the extent and period of its use also builds confidence. Provided that configuration management and regression testing is in place, we expect the number of bugs to fall with time. The focus should be on maintaining a validated and verified code. ONR would expect the licensee/requesting party to demonstrate that the verification and validation of the code was adequate to support the intended application. No specific process for reconciling legacy quality assurance gaps is provided in ONR guidance.	The applicant or licensee using the analytical codes and methods is responsible for the quality assurance of those tools. A non-QA code that is brought into an applicants' or licensees' QA programme would undergo a commercial grade dedication. Commercial-grade dedication is a process by which a commercial-grade item (CGI) is designated for use as a basic component. This acceptance process is undertaken to provide reasonable assurance that a CGI to be used as a basic component will perform its intended safety function and, in this respect, is deemed equivalent to an item designed and manufactured under a 10 CFR Part 50, Appendix B, quality assurance programme. This assurance is achieved by identifying the critical characteristics of the item and verifying their acceptability by inspections, tests, or analyses by the purchaser or third-party dedicating entity.
1.3 Code Assessment						
Question: Does the regulator have guidance on the sufficiency and adequacy of the data (e.g. scaling, range of conditions, number of cases) used to support the verification and validation of analytical codes and methods?						
Yes and No. CNSC has generic high level requirements and guidance on these topics but specifics are deferred to the vendor. CNSC will assess such details. It would appear that, e.g. imposing "number of cases" is not feasible.	Yes guide 28 addresses these questions.	There are no specific guidance provided. General requirements related to validation/verification are specified in the German SiAnf (see 1.1).	-	Yes. General recommendation on this issue are given by the guidance RD-03-34-2000.	A graded approach is required. See above. SAP AV.3 covers use of data in fault analysis: "The data used in the analysis of aspects of plant performance with safety significance should be shown to be valid for the circumstances by reference to established physical data, experiment or other appropriate means." NS-TAST-GD-042 provides further guidance on data used to support computer code validation.	RG 1.203, "Transient and Accident Analysis Methods," provides guidance on the entire process of evaluation model development, including code assessment (i.e. verification and validation). In general, the need for data and testing is driven by the identification, importance, and understanding of key phenomena and processes. The identification, importance, and understanding of these phenomena is assessed through the phenomena identification and ranking table (PIRT) process. Important phenomena require sufficient testing, in the appropriate ranges, to provide a high level of understating in the applicable scenarios.

Table A.1: **Country answers related to regulatory aspects** *(cont'd)*

Canada	France	Germany	Italy	Russia	United Kingdom	United States
Question: What quality assurance requirements are imposed on the experiments and associated data used in model development and validation of analytical codes and methods?						
Such experiments must be relevant to the application, data verifiable, uncertainties quantified.	Usual QA	No requirements	-	Requirements of RD-03-34-2000 and RB-061-11.	The experiments need to be reported and the data verified in accordance with appropriate procedure for control of work. This is a condition for all work which may affect a licensee's safety. Moreover, records need to be maintained in accordance with procedures for controlled document management.	Applicants and licensees are responsible for the quality assurance of experiments performed by or on behalf of themselves. 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," provides specific requirements that need to be addressed as part of a testing programme. Data not collected under a QA programme can be used, but needs to be assessed as part of a commercial grade dedication. High quality, independently developed data is also acceptable for use in and assessment of analytical codes and methods (e.g. ASME steam property tables, evaluated nuclear data files (ENDF), historical test data from national laboratories). Additional data may be acceptable and is considered on a case-by-case basis.
Question: What are accepted methods to assess uncertainty estimations (e.g. comparison with experimental data and operational measurements, comparison with exact analytical solutions, with precision codes, cross-verification with similar codes with estimated uncertainties, expert judgement, ...)?						
All of the above with exception of the code-to-code comparisons. The data must be of sufficient quality (i.e. measurement uncertainties must be quantified, data must cover the entire range of conditions for a given code application).	The code uncertainties (i.e. of its constitutive relationships) are assessed in different ways: either on the basis of the code/experiment differences or by expert judgement.	A specific set of methods is not provided. General requirements are given in the German SiAnf (see 1.1)	-	All of the above methods are used. Preference is given to comparison with experimental data, operational measurements and precision analytical codes. For codes simulating severe accidents, expert judgement are often used.	This needs to be addressed on a case-by-case basis and depends on what is reasonably practical for the application.	Uncertainty estimations require a comparison against experimental data and operational measurements. Comparison with exact analytical solutions, and code-to-code comparisons may be acceptable for code verification, but code validation and uncertainty quantification require the use of data obtained through physical experiments.
Question: What value characterises the estimated uncertainty of the analytical code or method (e.g. the maximum deviation of the calculations from the results of measurements or reference calculations, the standard deviation, ...)?						
Uncertainty is determined by two numbers: bias and 2 sigma random deviation (for consistency with BEAU method).	For conservative approach, the maximum deviation of the calculations from the results of measurements is considered.	n/a	-	If it possible uncertainty is determined by the standard deviation of the calculations from the results of measurements or reference calculations. When small statistics do not allow the use to estimate of uncertainty the standard deviation, the maximum observed deviation of the calculation from the reference result is used.	Various statistical approaches are acceptable, these can be distribution free (e.g. Wilkes), or with suitable justification can assume an appropriate statistical distribution.	This is treated on a case-by-case basis. A 95/95 statistical treatment is common when sufficient data exists. However, most integral effects tests do not produce an adequate number of samples to make a robust statistical argument, and a bounding treatment is used.

Table A.1: **Country answers related to regulatory aspects** *(cont'd)*

Canada	France	Germany	Italy	Russia	United Kingdom	United States
Question: How is the uncertainty of the experimental data or the uncertainty of the reference code used for comparison taken into account when assessing the uncertainty of the code being tested?						
Experimental uncertainty is added to the code prediction uncertainty in square norm.	There is no unique or generic method.	n/a	-	Experimental uncertainty, is added to the code prediction uncertainty in square norm.	This needs to be done appropriately for the application. SAP AV.5 regarding documentation for analysis codes and methods states that the documentation should include a description of the uncertainties in the model.	This is treated on a case-by-case basis. When a sufficient amount of data is collected, instrumentation error is relatively small, and there are no known biases, then the comparison is typically best-estimate code compared against unbiased data (e.g. critical heat flux correlation development). This situation can be different with integral effects tests with limited data points and scaling distortions. In this scenario, the evaluation model needs to accommodate the known biases in the estimation of the figure-of-merit. This is typically done using bounding/conservative arguments.
Question: How is the estimated code uncertainty taken into account in the safety analysis along with other sources of error (e.g. uncertainty of uranium enrichment, technological uncertainties in the manufacture of fuel rods and fuel assemblies, the uncertainty of determining and maintaining re-actor power,…)?						
Bounding values of parameters are considered in conservative analysis.	Methods combining and propagating all uncertainty sources are accepted on a case-by-case basis (e.g. based on a Monte-Carlo sampling technique plus order statistics or Wilks' formula in order to assess non-parametric tolerance limits). Uncertainty sources have to be treated following two types: aleatory and epistemic. The probabilistic modelling (based on probability distributions) is accepted for aleatory uncertainties but not for epistemic uncertainties (which have to be taken into account in a specific and justified way).	Requirements are defined in SiAnf Annex 5.	-	In most cases, conservative safety analyses take into account the code uncertainty and other uncertainties indicated above in the form ±2σ. In the justification of nuclear safety, ±3σ or maximum observed uncertainty are taken into account.	The qualification of a code needs to address the uncertainties associated with qualification and any bias quantified. Any additional uncertainty associated with application needs also to be accounted for.	Uncertainty is commonly treated by two means (1) conservative biasing of inputs, and (2) application of a penalty/bias to the code or safety limit. Key inputs to the safety analysis (e.g. reactor power, power distribution, control rod positions) are required to be surveilled during plant operation (10 CFR 50.36, "Technical Specifications"). Surveillance limits are set by the safety analysis. Violation of a surveillance limit requires prompt correct action of the condition or shutdown of the reactor. Additionally, the analysis prediction or safety limit is typically biased to ensure adequate margin to account for uncertainty. This treatment is not universal. The NRC has also accepted statistical arguments based on sampling of analysis inputs (e.g. 95/95 peak clad temperature limit in LOCA).

Table A.1: **Country answers related to regulatory aspects** *(cont'd)*

Canada	France	Germany	Italy	Russia	United Kingdom	United States
Question: Are there specific aspects and methods used for uncertainty assessment in relation to passive functions and associated systems?						
No	Methodology to be applied to passive system, although following the same requirements, have to be justified and in general cannot follow the same models as the ones related to active systems. There are several activities within OECD or IAEA that deals with this topics, however there is no common agreement achieved to date.	No	-	No.	In concept, these are no different and the same approach is needed. In practice, they can introduce particular phenomena (for example, their behaviour may be chaotic).	No special treatment of uncertainty. One possible exception may be the single failure criteria for passive valves. Systems need to be designed to adequately accommodate a single failure in the system. This design requirement is reflected in the safety analyses. Simple check valves are not considered susceptible to single failure when driving forces are high (such as under forced circulation or when the pressure difference is very large). Passive systems tend to have low driving forces and pressure differences such that the failure of a check value to open may need to be considered in the safety analysis.
Question: Are there specific requirements on documentation of analytical codes and methods?						
This is part of QA.	There is no specific requirement as long as, analytical codes and methods having to be approved by the safety authority, the documentation is sufficient in support for approval.	There are no specific requirements on the documentation of the analytical codes and methods itself.	-	There are specific requirements only for the verification report (RD-003-34-2000).	Further guidance for ONR Inspectors is available in technical assessment guides e.g. NS-TAST-GD-042 Validation of Computer Codes and Calculation Methods.	Yes. See answer to Question 1.1. NRC regulations are specific documentation requirements for LWR LOCA. These requirements are reflected in generic guidance (i.e. RG 1.203).
Question: What information should be included in the code approval / certificate documentation (e.g. range of applicability, estimated errors, requirements to users)?						
N/A	The qualification of the code is declared by the licensee for the intended scope of utilisation. Qualification is declared if the following conditions are met: - the code or method is capable of calculating the variables of interest for the intended scope of utilisation with uncertainty values adapted to the needs of the safety studies in which it will be used; - the utilisation range for which qualification is declared is within the validity range of the code.	n/a	-	In accordance with the RD-03-34-200, the code certificate should include the following information: Name code; Authors Names; Developer Organisation; Applicant Organisation; Purpose of code; Modelled objects; Modes; Application restrictions; Permissible parameters values; Accuracy provided in the range of permissible parameters; Information on calculation methods; Information about the databases (libraries of constants) used in the code; The list of organisations that are allowed to use the code.	Range of applicability, estimated errors and biases and requirements to users.	Approved methodologies typically have a set of limitation and conditions that specify (1) the range of applicability of the method, (2) penalties that need to be applied, (3) any limitations/restrictions on the use of the methodology (e.g. specific code options or correlations that must be used), and (4) any additional conditions that must be addressed by the user of the analytical code/method (e.g. reporting requirements).

Table A.1: **Country answers related to regulatory aspects** *(cont'd)*

Canada	France	Germany	Italy	Russia	United Kingdom	United States
Question: What are the accepted methods used to identify knowledge gaps and phenomenological modelling requirements in analytical codes and methods (i.e. what evidence can the regulator use to determine that the analytical codes and methods are adequate)?						
Expert judgement as well as V&V	The methods are accepted on a case by case basis.	n/a	-	The main evidence for the regulator in determining the adequacy of analytical codes and methods is contained in the code certificate. It is important that the code is used for the objects and modes specified in the certificate, and that the input and output parameters remain within the permissible range, also indicated in the code certificate.	ONR is not prescriptive, however NS-TAST-GD-042 identifies phenomenon identification and ranking is a potential starting point. We expect potential licensees to employ the precautionary principle when faced with uncertainty and to provide appropriate safety margins.	The PIRT process is typically used to identify, rank the importance of, and assess the understanding of key phenomena. See answer to question 1.3.
1.4 User Qualification						
Question: Are there requirements for organisation (developer and user) qualification?						
Yes, the personnel must be qualified	Not specifically. There are requirements within QA.	No	-	The qualifications of the developer and user organisations are evaluated in the code certification process. The code certificate contains a list of organisations that are allowed to use the code.	A suitable systematic approach to training and experience is required for qualification.	Yes. Personnel qualification records are required to be maintained as part of the quality assurance programme. The NRC does not impose specific criteria regarding the necessary credentials/training for the user of an analytical code/method, but reasonable expectations include demonstrated competence with the analytical code or method through experience or training.
Question: How are user choices factored in the consideration of uncertainties (e.g. selection of input deck parameters, models)?						
These choices should be covered by V&V	Sensitivity analyses (on meshing for example).	n/a	-	The regulator considers only the final result. If in doubt, we can make our own calculations or require additional calculations from the licensee.	We expect the development and qualification of reference input decks and variance from these to be controlled.	The approval of an evaluation models typically requires that the analytical code or method be used consistent with its validation (i.e. the same code options used to assess the uncertainty against experimental data need to be used when performing licensing calculations). Additionally, the uncertainty in initial plant parameters need to accommodate the uncertainties and ranges accounted for in the facilities technical specifications. The direction of conservatism can be applied using engineering judgement, when applicable, or demonstrated through a sensitivity analysis.

Table A.1: **Country answers related to regulatory aspects** *(cont'd)*

Canada	France	Germany	Italy	Russia	United Kingdom	United States
2. Confirmatory Analyses						
2.1 Need for Confirmatory Analyses						
Question: To what extent is confirmatory analysis performed and what factors into the decision to conduct confirmatory analyses?						
Confirmatory analysis is performed to confirm the key design values. The codes used must be different that those used by the vendor.	On a case-by-case basis, when the margin to criteria is narrow and/or safety issue is high.	Authorised experts in accordance with § 20 Atomic Energy Act (Atomgesetz (AtG)) are usually consulted as part of the reviews and assessments to provide expert support. These authorised experts are consulted by the way of contracts specifying the modalities and requirements, for example with regard to the qualifications required from them. In order to maintain the knowledge of the experts involved in the long run, the contracts are usually concluded for a longer period of time. On behalf of the regulatory body, the authorised experts carry out i.e. independent calculations, review design drawings, inspection plans and other documents, and prepare expert opinions and expert statements. The final assessment and decision, however, rests with the Regulatory Body.	-	-	This depends on the novelty of the application, and is determined by a graded approach to assessment. In particular, this is likely to be undertaken where there is a significant area of technical uncertainty.	Confirmatory analyses are typically performed to facilitate NRC staff's understanding of (1) plant performance, (2) margin to safety limits, and (3) sensitivity to input parameters. Confirmatory analyses are not required, but can be an effective tool to help staff efficiently and more thoroughly understand the analyses provided by the applicant/licensee. Typical scenarios where the staff will perform confirmatory analyses are (1) when there is a first-of-a-kind design or design feature that the staff is not familiar with, (2) the staff is uncertain regarding the robustness of an analysis performed by the applicant/licensee, or (3) the staff wants to assess the sensitivity of the analysis to uncertainties in plant parameters due to relatively low margin to safety limits. In addition, NRC staff have performed independent analyses to support rulemaking, assess the safety-significance of issues encountered in the operating fleet, and address generic safety issues.
Question: What alternatives to confirmatory analyses are available to support regulatory decisions?						
Assessment of vendor's calculations by inspection of the methods chosen, assumptions made, models, data used, etc. – this is complementary to confirmatory analysis (i.e. not a replacement).	Simple calculational tools, answers to sensitivity analyses, comparison with similar cases.	In practice, plausibility checks as well as engineering judgement can be used to support regulatory decision making.	-	Confirmatory analyses of separate transients or states is carried out in rare cases when the regulator has doubts about the confidence of the calculation performed by the licensee.	The more routine approach is to sample the qualification evidence in detail and to request further justification where required.	Regulatory findings are based the analyses and data submitted by the applicant/licensee, and not upon confirmatory analyses. However, confirmatory analyses are used to facilitate better understanding of system performance, and analysis uncertainties and sensitivities. As an alternative to confirmatory analyses, the NRC may request additional information be submitted by the applicant/licensee to facilitate better understanding of system behaviour and margin to safety limits.

Table A.1: **Country answers related to regulatory aspects** *(cont'd)*

Canada	France	Germany	Italy	Russia	United Kingdom	United States
2.2 Requirements on confirmatory analyses						
Question: To what extent are confirmatory analyses required to be independent (e.g. is it acceptable for the regulator to perform sensitivity analyses using the same code as the nuclear plant operator or designer)?						
A different code must be used. Sensitivity cases should be part of the vendor's submission.	In an ideal way, it recommended to make confirmatory analyses with an independent code and with the same code.	No requirements.	-	For the confirmatory analyses, the SEC NRS can use both its own codes (for example, RADUGA for VVER-type reactors) and the codes received in the framework of international co-operation such as DYN3D-ATHLET, ASTEC, AC2.	Independent methods are preferable, particularly where a higher-fidelity code is available. However, this depends on the objective of the study.	Because the purpose of confirmatory analyses is to facilitate understanding of the analysis submitted by the applicant/licensee, use of an independent code is not necessary. It is acceptable to use the same code as the applicant/licensee for confirmatory analyses.
Question: What level of quality assurance and confidence are imposed on the analytical codes and methods used in confirmatory analysis by the regulator in comparison to the applicant?						
Regulator's codes must have proper V&V.	No specific requirements as no compulsory and not used to do the safety demonstration (only used to ask questions to the utilities)	No requirements.	-	The quality assurance and confidence requirements are the same as for the applicant's codes, except that certification is not required for codes used by the regulator.	Normal control of work is required, but these calculations do not carry the burden of responsibility for safety and therefore a risk-proportionate approach is reasonable	The NRC does not have quality assurance requirements for confirmatory analyses. However, it is common practice at the NRC to perform code assessments on the NRC developed codes and to perform peer reviews on confirmatory analysis.

Appendix B. Synthesis of country responses to survey for analytical codes and methods for advanced reactors

In addition to the regulatory aspects regarding analytical codes and methods, information on existing codes and methods are provided in survey answers (see Table B.2 in this Appendix). It should be pointed out that the resulting list of analytical codes and methods is not complete.

The codes can be categorised into the type of code, the field of application regarding the plant state and scenario, and regarding the type of reactor concept. Regarding neutron kinetics, the methods are point kinetics, the 2D/3D neutron kinetics based on multi-group diffusion or transport approximation, supported by some precise Monte Carlo tools, and so on. In thermal-hydraulics, distinctions are made between Lumped-Parameter (LP), sub-channel methods and CFD. Further criteria are related to mechanical codes and structure analysis and fuel performance. Additional categories are the capability to simulate fission product behaviour and transport and to simulate severe accident phenomena.

The tendency worldwide for both areas of expertise – neutronics and thermal-hydraulics – can be traced in the wide use of the Hi2Lo (High fidelity to inform Low fidelity modelling) methodology.

One of the complicating factors for advanced non-LWRs is that it may be necessary to have a tight coupling between several analysis codes because of the feedback between physical phenomena. Fast reactors, for example, require a tight coupling between neutronics, thermal-hydraulics and in some cases thermal-mechanics to account for rapid changes in power due to reactivity feedback that can occur with changes in temperature. Thermomechanical expansion of the core is an important phenomenon that must be accounted for because of its impact on neutron leakage. As mentioned earlier, coupled methodologies require specific qualification.

Most modern liquid metal and molten salt reactors are “pool” as opposed to “loop” designs. That is, flow circulation and heat transfer take place within the reactor vessel rather than through an external flow loop. The reactor vessels often contain large regions in the plena where thermal stratification and internal recirculation can be important. LP type codes generally have difficulty simulating these processes because they lack the models and experimental support, in addition to the detailed nodalisation required for fluid mixing. Regarding simulations inside the reactor vessel, codes can be distinguished between LP or CFD type codes. The LP type codes (e.g. ASTEC, ATHLET, CATHARE2, MELCOR, RELAP5) use often point-kinetic models. Most of the used CFD codes do not consider neutron kinetics, one exception being the SIMMER code. Examples of neutron kinetic codes are DYN3D, PARCS, SERPENT, TORT-TD and ERANOS with the kinetic module KIN3D. Partly these are coupled to thermal-hydraulic codes.

Most of thermal-hydraulic codes can handle steady-state conditions, transients and accident conditions. However, regarding fast transients such as 3D modelling of SFR core mechanics in transient conditions, including very fast ones (e.g. driven by the pressure pulse caused by sodium vapour bubble collapse), knowledge gaps remain. Another example is the 3D modelling of transition from forced convection to natural convection for the pool-type SFR, including fluid dynamics and thermal stratification and the simulation of systems such as the dynamic modelling of thermal electromagnetic pumps.

As long as scenarios (e.g. leading to core degradation) cannot be practically eliminated, the regulators consider that these kinds of scenarios must be investigated. Regarding severe accident simulation capability, few codes are available. These include, for example, ASTEC (for SFRs) and MELCOR (for SFRs and VHTRs).

Table B.1: **List of codes according to answers in questionnaires**

Canada	France	Germany	Italy	Russia	United Kingdom	United States
Canada did not provide any information about code, to avoid any impression about preferences.	CATHARE2	AC2 / ATHLET 3.2 Open FOAM V7 ANSYS CFD PARCS SERPENT DYN3D ATTICA3D TORT-TD/ATHLET TORT-TD/ATTICA3D	ERANOS MCNP SERPENT ANTEO+ TRANSURANUS CATHARE2 RELAP-5/mod3 RELAP-5-3D SIMMER MELCOR ASTEC	SOKRAT-BN CONSYST JAR-FR TRIGEX MCU-BR MMKFK RADAR DIN800 BURAN GRIF-SM TRANMS TWOCOM RGT BOX200 AERO SLEAK LEAK RELAP5/MOD3 KORAT DRACON KONDOR PINCOD ACME ASTRA-AES DPIPE5 ANSYS UZOR 1.0 Zenit-95 DINAMIKA-3 COBEF CRISS 5.3 BARS COREMELT TRAMS-TWEL MOTE SUBMELT MELTROP TRANS-DUCT CRUST TRELM FAZAN NTAMS-FDOWN, INTERACT CSLOY STO-BED BRUT FIRECON LIMITS CUPUL-M VIBROS-3.1 RISKZONE SIBILLA	WIMS MONK MELCOR RELAP-5/mod3 RELAP-5-3D HTGR customised ANSYS CFX WIMSTER PRIMSTER	SCALE PARCS TRACE SERPENT BISON FAST AGREE PRONGHORN

Table B.2: **List of codes related to advanced reactor concepts**

Code	Code type						Technical area				Reactor type						Availability	
	Neut[1]	THY[2]	Mech[3]	Fuel[4]	FP	SA	ST	Tr	Ac	SA	SFR	VHTR	GFR	MSR	LFR	SCWR	Status[5]	Validated[6]
AC[2] 2019/ ATHLET	☒ PK	☒ LP	☐	☐	☐	☐	☒	☒	☒	☐	✓	✓	✓	✓	✓	✓	A, D	☒
ACME	☐	☐	☒ FEM	☐	☐	☐	☒	☒	☒	☐	✓						A	☒
AERO	☐	☐	☐	☐	☒ Transport of FPs and aerosols	☐	☐	☐	☒	☒	✓						A	☒
ANSYS CFD	☐	☒ CFD	☒ FEM	☐	☒	☐	☒	☒	☒	☐	✓	✓	✓	✓	✓	✓		☐
ANTEO+	☐	☒ SC	☐	☐	☐	☐	☒	☐	☐	☐	✓				✓		A	☐
ASTEC	☒ PK	☒ LP	☒ AN	☒ EC	☒	☒	☒	☒	☒	☒	✓						A, D	☒
ASTRA-AES	☐	☐	☒ FEM	☐	☐	☐	☒	☒	☒	☐							A	☒
ATTICA3D	☐	☒	☐	☐	☐	☐	☒	☒	☒	☐		✓						☐
BARS	☐	☐	☐	☐	☐	☒ PSA	☐	☐	☒	☒	✓				✓		A	☒
BISON	☐	☐	☐	☒ BP	☐	☐	☒	☒	☐	☐	✓		✓		✓	✓	D	☐
BOX200	☐	☐	☐	☐	☐	☒ Na fire	☐	☐	☐	☐	✓						A	☒

1. Neutronics (PK: point kinetics; 2D/3D: two- or three-dimensional modelling; MC: Monte Carlo; DI: neutron diffusion; TR: neutron transport {PN: Pn approximation; SPN: SPn approximation; BN: Bn approximation; SN: Sn approximation; VN: variational-nodal method}; CE: continuous energy; MG: multi-group energy discretisation.
2. Thermal-hydraulics (LP: lumped parameter; SC: sub-channel method; CFD: computational fluid-dynamics.
3. Mechanics (AN: analytical model; EC: engineering correlations; FEM: finite element model).
4. Fuel (EC: engineering correlations; BP: basic principles constitutive equations).
5. Status (D: under development; A: actively used; U: being upgraded).
6. Validation: in the domain of application to any advanced reactor type (specified in brackets).

Table B.2: **List of codes related to advanced reactor concepts** *(cont'd)*

Code	Code type						Technical area				Reactor type						Availability	
	Neut	THY	Mech	Fuel	FP	SA	ST	Tr	Ac	SA	SFR	VHTR	GFR	MSR	LFR	SCWR	Status	Validated
BRUT	☐	☒ Movement and retention of molten materials in the RV EC	☐	☐	☐	☐	☐	☐	☐	☒	✓						A	☒
BURAN	☒ PK	☒ SC	☐	☐	☐	☐	☐	☒	☒	☐	✓						A	☒
CATHARE 2	☒ PK	☒ LP	☐	☒ EC	☐	☐	☒	☒	☒	☐	✓				✓		A	☐
COBEF	☐	☐	☒ Strength of reinforced concrete containment FEM	☐	☐	☐	☒	☒	☒	☒	✓						A	☒
CONSYST	☒ Preparing of cross sections 1D, MG, SPN	☐	☐	☐	☐	☐	☒	☒	☒	☒	✓				✓		A	☒
COREMELT	☒ 3D, DI, MG	☒ SC	☒ EC	☒ EC	☒	☐	☐	☐	☐	☒	✓						A, D	☐
CRISS 5.3	☐	☐	☐	☐	☐	☒ PSA	☐	☐	☒	☒	✓				✓		A	☒
CRUST	☐	☐	☐	☒ Movement and solidifying of molten material EC	☐	☐	☐	☐	☐	☒	✓						A	☒
CSLOY	☐	☒Cooling of the core fragments in the finale stage of a SA SC	☐	☐	☐	☐	☐	☐	☐	☒	✓						A	☒

Table B.2: **List of codes related to advanced reactor concepts** *(cont'd)*

Code	Code type						Technical area				Reactor type						Availability	
	Neut	THY	Mech	Fuel	FP	SA	ST	Tr	Ac	SA	SFR	VHTR	GFR	MSR	LFR	SCWR	Status	Validated
CUPUL-M	☐	☐	☐	☐	☐	☒ Calc. parameters in the volume of containments	☐	☐	☒	☒	✓				✓			☒
DIN800	☒ PK	☒ SC	☐	☐	☐	☐	☐	☒	☒	☐	✓						A	☒
DINAMIKA-3	☐	☐	☒ FN, EC FEM	☐	☐	☐	☐	☒	☒	☐	✓				✓		A	☒
DPIPE5	☐	☐	☒ FEM	☐	☐	☐	☒	☒	☒	☐	✓						A	☒
DRACON	☐	☐	☐	☒ AN, EC	☐	☐	☒	☒	☒	☐	✓						A, (D for LFR)	☒
DYN3D	☒ 3D, MG, DI/TR (SPN)	☒	☐	☒	☐	☐	☒	☒	☒	☐	✓				✓			☐
ERANOS 2	☒ 3D, MG, DI/TR (SN, VN)	☐	☐	☐	☐	☐	☒	☐	☐	☐	✓		✓		✓		A	☒ (SFR)
FAST	☐	☐	☐	☒ EC	☐	☐	☒	☒	☐	☐	✓		✓		✓	✓	A	☐
FAZAN	☐	☐	☐	☒ Motion of the fuel corium EC	☐	☐	☐	☐	☐	☒	✓						A	☒
FIRECON	☐	☐	☐	☐	☐	☒ Deflagration and detonation of the air-H_2 mixture in the containment EC	☐	☐	☐	☒	✓						A	☒
GRIF-SM	☐	☒ SC	☐	☐	☐	☒	☐	☐	☒	☒	✓						A	☒
INTERACT	☐	☐	☐	☒ Motion of the fuel corium EC	☐	☐	☐	☐	☐	☒	✓						A	☒
JAR-FR	☒ 3D DI	☐	☐	☐	☐	☐		☐	☐	☐	✓				✓		A (SFR)	☒ (SFR)

Table B.2: **List of codes related to advanced reactor concepts** *(cont'd)*

Code	Code type						Technical area				Reactor type						Availability	Code
	Neut	THY	Mech	Fuel	FP	SA	ST	Tr	Ac	SA	SFR	VHTR	GFR	MSR	LFR	SCWR	Status	Validated
KONDOR	☐	☐	☐	☒ AN, EC	☐	☐	☒	☒	☒	☐	✓						A	☒
KORAT	☐	☐	☐	☒ AN, EC	☐	☐	☒	☒	☒	☐	✓						A	☒
LEAK	☐	☐	☐	☐	☐	☒ (Large SG leak)	☐	☐	☐	☒	✓						A	☒
LIMITS	☐	☐	☐	☐	☐	☒ Deflagration and detonation of the air-H_2 mixture in the containment EC	☐	☐	☐	☒	✓						A	☒
MCNP 6	☒ 3D, CE, MC	☐	☐	☐	☐	☐	☒	☐	☐	☐	✓	✓	✓	✓	✓	✓	A	☐
MCU-BR	☒ 3D, CE, MG, MC	☐	☐	☐	☐	☐	☒	☐	☐	☐	✓	✓	✓	✓	✓	✓	A	☒ (SFR, LFR)
MELCOR	☒ PK	☒ LP	☒ EC	☒ EC	☒	☒	☒	☒	☒	☒								☐
MELTROP	☐	☐	☐	☒ FA melting EC	☐	☐	☐	☐	☐	☒	✓						A	☒
MMKFK	☒ 3D, MG, MC	☐	☐	☐	☐	☐	☒	☐	☐	☐	✓						A (SFR)	☒ (SFR)
MOTE	☐	☐	☐	☒ Movement of molten cladding EC	☐	☐	☐	☐	☐	☒	✓						A	☒
NTAMS-FDOWN	☐	☐	☐	☒ Motion of the fuel corium EC	☐	☐	☐	☐	☐	☒	✓						A	☒
Open FOAM V7	☐	☒ CFD	☒ FEM	☐	☒	☐	☒	☒	☒	☐	✓	✓	✓	✓	✓	✓	U	☐
PARCS	☒ 3D, MG, DI/TR(SPN)	☐	☐	☐	☐	☐	☒	☒	☒	☐	✓						A	☐
PINCOD	☐	☐	☐	☒ EC	☐	☐	☒	☒	☒	☐	✓						A	☒
RADAR	☒ 3D, MG, DI	☐	☐	☐	☐	☐	☐	☒	☒	☐	✓						A	☒

Table B.2: **List of codes related to advanced reactor concepts** *(cont'd)*

Code	Code type						Technical area				Reactor type						Availability	
	Neut	THY	Mech	Fuel	FP	SA	ST	Tr	Ac	SA	SFR	VHTR	GFR	MSR	LFR	SCWR	Status	Validated
RELAP 5/mod 3	☒ PK	☒ LP	☐	☒ EC	☐	☐	☒	☒	☒	☐	✓		✓		✓		A	☐
RELAP 5-3D	☒ PK	☒ LP	☐	☒ EC	☐	☐	☒	☒	☒	☐	✓		✓		✓		A	☐
RGT	☐	☐	☐	☐	☒	☐	☐	☐	☒	☒	✓						A	☒
RISKZONE	☐	☐	☐	☐	☒ Evaluation of the radiological consequences AN, EC	☐	☐	☐	☒	☒	✓						A	☒
Serpent	☒ 3D, CE, MC	☐	☐	☐	☐	☐	☒	☐	☐	☐	✓				✓		U	☐
SIBILLA	☐	☐	☐	☐	☒ Evaluation of the radiological consequences AN, EC	☐	☐	☐	☒	☒	✓						A	☒
SIMMER	☒ 3D, MG, DI	☒ CFD	☒ AN	☒ EC	☐	☐	☒	☒	☒	☐	✓				✓		A	☐
SLEAK	☐	☐	☐	☐	☐	☒ (Small SG leak)	☐	☐	☒	☐	✓						A	☒
SOKRAT-BN	☒ PK	☒ SC	☒ EC	☒ EC	☐	☐	☐	☒	☒	☐	✓						A	☒
STO-BED	☐	☒ Cooling of the core fragments in the finale stage of a SA SC	☐	☐	☐	☐	☐	☐	☐	☒	✓						A	☒
SUBMELT	☐	☐	☐	☒ Fuel melting EC	☐	☐	☐	☐	☐	☒	✓						A	☒
TORT-TD/ATTICA3D	☒ 3D, MG, DI/TR(SN)	☐	☐	☐	☐	☐	☒	☒	☒	☐		✓						☐

Table B.2: **List of codes related to advanced reactor concepts** *(cont'd)*

Code	Code type						Technical area				Reactor type						Availability	
	Neut	THY	Mech	Fuel	FP	SA	ST	Tr	Ac	SA	SFR	VHTR	GFR	MSR	LFR	SCWR	Status	Validated
TRAMS-TWEL	☐	☐	☐	☒ Pin meltdown EC	☐	☐	☐	☐	☐	☒	✓						A	☒
TRANMS	☐	☒ SC	☐	☐	☐	☒	☐	☐	☒	☒	✓						A	☒
TRANS-DUCT	☐	☐	☐	☒ FA duct meltdown EC	☐	☐	☐	☐	☐	☒	✓						A	☒
TRANSURANUS	☐	☐	☐	☒ EC/BP	☐	☐	☒	☒	☐	☐	✓		✓		✓	✓	U	☐
TRELM	☐	☐	☐	☒ Movement and solidifying of molten material EC	☐	☐	☐	☐	☐	☒	✓						A	☒
TRIGEX	☒ 3D DI	☐	☐	☐	☐	☐	☒	☐	☐	☐	✓				✓		A (SFR)	☒ (SFR)
TWOCOM	☐	☐	☐	☐	☒	☐	☐	☐	☒	☒	✓						A	☒
UZOR 1.0	☐	☐	☒ FEM	☐	☐	☐	☒	☒	☒	☐	✓						A	☒
VIBROS 3.1	☐	☐	☐	☐	☒ Evaluation of the radiological consequences AN, EC	☐	☐	☐	☒	☒	✓						A	☒
ZENIT-95	☐	☐	☒ FEM	☐	☐	☐	☒	☒	☒	☐	✓						A	☒

In the following, the survey answers related to simulation codes are listed.

Table B.3.1: **AC²**

Name:	**AC² 2019/ATHLET 3.2**	
Brief description of code/method	The thermal-hydraulic computer code ATHLET (Analysis of thermal-hydraulics of leaks and transients) is being developed by the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) for the analysis of operational conditions, abnormal transients and all kinds of leaks and breaks in nuclear power plants. The aim of the code development is to cover the whole spectrum of design basis and beyond design basis accidents (without core degradation) for PWRs, BWRs, SMRs and future Generation IV reactors with one single code. The main code features are: • advanced thermal-hydraulic modelling (compressible fluids, mechanical and thermal nonequilibrium of vapour and liquid phase) • availability of diverse working fluids: light or heavy water, helium, sodium, lead or lead-bismuth eutectic, supercritical carbon dioxide, molten salts as well as user-provided single-phase (non-boiling) working fluids • heat generation, heat conduction and heat transfer to single- or two-phase fluid considering structures of different geometry, e.g. rod or pebble bed • interfaces to specialised numerical models such as 3D neutron kinetic codes or 3D CFD codes for coupled multiphysical or multiscale simulations • control of ATHLET calculation by call-backs to programming language independent user code enabling the coupling of external models • plug-in technique for user provided code extensions • modular code architecture • separation between physical models and numerical methods • numerous pre- and post-processing tools • portability • continuous and comprehensive code validation	
Publication	Lerchl, G. et al., (2019), ATHLET 3.2 – User's Manual, distributed with ATHLET, GRS, March 2019	
Developer/maintained organisation	GRS	
Type (*select all applicable boxes*)	☒ Neutronics, type: Point kinetics	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☒ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	Single and two-phase flow states in the reactor. AOO and DBA and DEC without core degradation. Limited validation basis.	
Usage by	GRS, others	
Ongoing research and validation activities	Yes	

Table B.3.2: **AGREE**

Name:	**AGREE**	
Brief description of code/method	Thermal-fluids solver for pebble bed and prismatic cores. Originally developed to model gas reactors. Coupled to PARCS. It is being modified and used by a MSR developer.	
Publication	The US NRC Advanced gas reactor evaluator (AGREE)	
Developer/maintained organisation	NRC/University of Michigan	
Type (*select all applicable boxes*)	☒ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	Limited to single phase. Modeling is focused on core thermal-fluids.	
Key phenomena	Thermal-fluids, reactor physics	
Usage by	Regulators, plant designers	
Knowledge gaps	Thermal-physical properties of molten salt	
Ongoing research and validation activities		

Table B.3.3: **ANSYS CFD**

Name:	**ANSYS CFD**	
Brief description of code/method	Computational fluid dynamics (CFD, flow simulation) is a tool with remarkable flexibility, accuracy, and a wide range of applications. Reliable numerical flow simulations (CFD), which provide the insights necessary for design concepts, are only possible if the right software is used. For reliable and meaningful CFD results, powerful software is needed. ANSYS CFD offers more than qualitative results. The software enables precise quantitative predictions of flow interactions and trade-offs. These insights open up unprecedented possibilities for your product – possibilities that even experienced engineers might otherwise miss.	
Publication	www.ansys.com/de-de/products/fluids	
Developer/maintained organisation	ANSYS	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☒ Mechanical codes and structure analysis	☐ Fuel performance
	☒ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☒ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Usage by	GRS, others	
Ongoing research and validation activities	Yes	

Table B.3.4: **ANSYS CFX (HTGR-Customised)**

Name:	**HTGR-Customised ANSYS CFX**	
Brief description of code/method	A customised version of ANSYS CFX is under development to provide detailed multiscale modelling of pebble bed and prismatic high-temperature gas reactors (including fuel element and TRISO fuel particle temperatures). The model will couple with the WIMS reactor physics code suite to facilitate coupled thermal and neutronic analysis for both accident analysis and fuel cycle simulation.	
Publication	To be presented at PHYSOR-202 conference – development ongoing	
Developer/maintained organisation	Wood	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☒ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	Whole-core thermal modelling of pebble bed and prismatic high-temperature gas-cooled reactors. Multiscale (TRISO particle, pebble/fuel element, whole core)	
Key phenomena	Single-phase flow in porous bed, detailed heat transfer	
Usage by	Industry	
Ongoing research and validation activities	Currently under initial development. Validation anticipated to include analysis of HTR-10 experiments.	

Table B.3.5: **ANTEO+**

Name:	**ANTEO+**	
Brief description of code/method	Thermal hydraulic sub-channel code for liquid-metal-cooled fuel assembly steady-state simulations	
Publication	Lodi, F. et al. (2016), "A subchannel code for thermal-hydraulic analysis of liquid-metal-cooled systems." *Nucl. Eng. Des.,* 301, pp. 128-152, https://doi.org/10.1016/j.nucengdes.2016.03.001. Lodi, F. et al. (2017), "Extension of the sub-channel code ANTEO+ to the mixed convection regime." *Nucl. Eng. Des.,* 322, pp. 368-378, https://doi.org/10.1016/j.nucengdes.2017.07.018.	
Developer/maintained organisation	Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA)	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☐ Transient (e.g. AOO)
	☐ Accidents	☐ Severe accidents

Table B.3.5: **ANTEO+** *(cont'd)*

Range of applicability	Steady-state, single-phase operation in mixed and force convections of liquid metal-cooled reactors. The code handles both hexagonal and square arrangements of pins in a closed sub-assembly. Various modifications to the geometry such as pins with different diameters inside the bundle, guide tubes, internal ducts for hosting finger-type control elements are possible or the inner discharge duct proposed for the JSFR (Japan Sodium-cooled Fast Reactor).
Key phenomena	Thermal-hydraulic modelling of momentum and energy exchange in-between pins; working fluids: sodium, lead and lead-bismuth eutectic as well as user-provided single-phase (non-boiling) working fluids; heat transfer with the inter-wrapper region and between the coolant and the pin clad. Effect of uncertainties on main output parameters considered via the Hot Channel factors.
Usage by	Several organisations in Europe and America
Knowledge gaps	Circumferential variations of pin clad temperature, especially in the presence of a wire. Heat exchange between interior and edge sub-channels in tight bundles with a wire spacer for heavy liquid metals.
Ongoing research and validation activities	Adjoint-based uncertainty propagation.

Table B.3.6: **ATHLET**

Name:	**ATHLET**	
Brief description of code/method	Thermal hydraulic system code for core and whole plant simulations	
Publication	Wielenberg, A. et al. (2019), "Recent improvements in the system code package AC^2 2019 for the safety analysis of nuclear reactors", *Nuclear Engineering and Design*, accepted for publication.	
Developer/maintained organisation	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☒ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	Analysis of whole spectrum of leaks and transients in PWRs and BWRs. The code is applicable to western reactor designs as well as Russian VVER and RBMK reactors and also to VHTR, SFR, LFR, MSR, SMR. Increase of power and enrichment, probabilistic safety analyses, core damage analyses, periodic safety analyses, boron dilution events, reactivity-initiated events (coupling with 3D neutron kinetics), research reactors (e.g. FRM-2 in Garching).	
Key phenomena	Advanced thermal-hydraulic modeling (compressible fluids, mechanical and thermal nonequilibrium of vapour and liquid phase); working fluids: light or heavy water, helium, sodium, lead or lead-bismuth eutectic, supercritical carbon dioxide, molten salts as well as user-provided single phase (non-boiling) working fluids; heat conduction and heat transfer to single- or two-phase fluid considering structures of different geometry, e.g. rod or pebble bed	
Usage by	Many organisations in Europe, America, Asia	
Knowledge gaps	SFR specific requirements, e.g. sodium boiling and two-phase flow simulation (fundamentals already implemented), thermal hydraulic impact of thermal expansion of structures.	
Ongoing research and validation activities	Addressing the knowledge gaps (see above); validation.	

Table B.3.7: **ATTICA3D**

Name:	**ATTICA3D**	
Brief description of code/method	3D system analysis Advanced Thermal hydraulics Tool for In-vessel and Core Analysis in 3D	
Publication	Hossain, K. et al. (2008), "Development of a fast 3D thermal-hydraulic tool for design and safety studies for HTRS", *Nuclear Engineering and Design*, 238, pp. 2976-2984.	
Developer/maintained organisation	University of Stuttgart, Institute of Nuclear Technology and Energy Systems (IKE)	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☒ System code	☒ High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	Safety related issues in high-temperature reactors; operational conditions with forced cooling as well as accident situations with heat removal by conduction and natural circulation	
Key phenomena	Porous media approach. Time dependent mass and energy conservation equations and simplified steady-state momentum conservation equations (dominance of friction) solved for cooling gas along with time dependent energy conservation equation for solids. Appropriate set of constitutive equations (e.g. effective heat conductivity of solid, pressure drop, heat transfer coefficient, etc.) is applied. Finite-volume method for the spatial discretisation. Fully implicit method with adaptive time step selection for the temporal integration in transient problems.	
Usage by	E.g. by GRS, coupling with TORT-TD in co-operation with U Stuttgart/IKE	

Table B.3.8: **CATHARE 2**

Name:	**BISON**	
Brief description of code/method	Finite element fuel performance code	
Publication	Bison	
Developer/maintained organisation	Idaho National Laboratory	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☐ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☒ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	Currently contains models for LWR fuel, SFR fuel, and TRISO fuel	
Key phenomena	Fuel mechanical	
Usage by	Regulators, TSOs, plant designers	
Knowledge gaps	Does not include all constitutive models needed for some advanced reactor fuel	

Table B.3.9: **CATHARE 2**

Name:	**CATHARE 2**	
Brief description of code/method	Thermal-hydraulic system code	
Publication	- "Status of CATHARE code for sodium cooled fast reactors" – *Nuclear Engineering and Design Volume 245*, April 2012, Pages 140-152 - *Development and Validation of Multi-Scale Thermal-Hydraulics Calculation Schemes for SFR Applications at CEA* – IAEA-CN-245-455 Geffraye, G. et al. (2011), "CATHARE 2 V2.5_2: a Single version for various applications", *Nuclear Engineering and Design,* Volume 241, Issue 11, November 2011, pp. 4456-4463.	
Developer/maintained organisation	CEA	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Key phenomena	Two-phase flow	
Usage by	ENEA, others	

Table B.3.10: **DYN3D**

Name	**DYN3D**	
Brief description of code/method	3D few-group neutron kinetics code for the calculation of steady states and transients in light water reactors with hexagonal or quadratic fuel assemblies. The neutron kinetics is connected with a two-phase thermohydraulic model and a fuel rod model.	
Publication	Rohde, U. et al. (2016), "The reactor dynamics code DYN3D – models, validation and applications", *Progress in Nuclear Energy,* 89, pp. 170-190.	
Developer/maintained organisation	Helmholtz-Zentrum Dresden-Rossendorf (HZDR)	
Type (*select all applicable boxes*)	☒ Neutronics, type: 3D steady-state and time-dependent, multi-group neutron diffusion approximation and SP3 transport approximation, Cartesian and hexagonal (trigonal) geometry	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☒ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	DYN3D or DYN3D/ATHLET analyses of transients and accident scenarios for different reactor types – VVER, PWR and BWR	
Key phenomena	3D steady state and transient neutron kinetics with hexagonal or quadratic fuel assemblies; linked with a thermohydraulic model and a fuel rod model; capable of calculating the burn-up prior to transient simulations;	
Usage by	NPP, vendors, nuclear regulatory authorities and technical support organisations	
Ongoing research and validation activities	Validation of physical models and system codes; modifying the codes or implementation of special models on requirement, e.g. SFR	

Table B.3.11: **FAST**

Name:	**FAST**	
Brief description of code/method	Thermal-mechanical fuel performance code	
Publication	FAST: The Merger of NRC's Fuel Performance Codes FRAPCON and FRAPTRAN for Scoping and Regulatory Decision Making	
Developer/maintained organisation	NRC	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☐ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☒ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	Steady-state and transient fuel performance	
Key phenomena	Fuel mechanical	
Usage by	Regulators and TSOs	
Knowledge gaps	Need to compare against integral test data (e.g. fuel centreline temperature predictions)	
Ongoing research and validation activities	EBR-II benchmarking	

Table B.3.12: **MELCOR**

Name:	**MELCOR**	
Brief description of code/method	MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactors (PWR and BWR), CANDU, prismatic fuel and pebble bed HTRs and sodium-cooled fast reactors. A broad spectrum of severe accident phenomena is treated in a unified lumped-parameter framework, including: thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heatup, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport behaviour.	
Publication	https://energy.sandia.gov/energy/nuclear-energy/nuclear-energy-safety-technologies/melcor	
Developer/maintained organisation	Sandia National Laboratories	
Type (select all applicable boxes)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☒ Fission product behaviour and transport	☒ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☐ Steady-state	☐ Transient (e.g. AOO)
	☐ Accidents	☒ Severe accidents
Range of applicability	Severe accident analysis	
Key phenomena	Fission product inventory and decay heat calculation, multi-phase flows, Zr oxidation and nitriding, eutectic mixtures, Fuel-coolant interaction (FCI), steam explosion, molten pool stratification, fission product transport (aerosol physics, iodine chemistry, bubble scrubbing), hydrogen combustion, molten core-concrete interaction (MCCI).	
Usage by	Industry, TSOs, universities, regulators	
Ongoing research and validation activities	Continuing extensive V&V against experimental test programmes, international benchmarks, plant data	

Table B.3.13: **MONK**

Name:	**MONK**	
Brief description of code/method	Monte Carlo software tool for nuclear criticality safety and reactor physics analysis. Advanced geometry modelling. Detailed continuous energy collision treatment, enables realistic 3D models for an accurate simulation of neutronic behaviour. Utilises various international data libraries JEFF, ENDF, JENDL New feature – applicable for extension to SMR variants	
Publication	www.answerssoftwareservice.com	
Developer/maintained organisation	Wood	
Type (select all applicable boxes)	☒ Neutronics, type:	
	☐ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☐ Transient (e.g. AOO)
	☐ Accidents	☐ Severe accidents
Range of applicability	Nuclear criticality safety, Monte Carlo. Advanced geometry. Reactor physics. Example applications include: • Uranium enrichment covering diffusion and centrifuge plant • Fuel fabrication for thermal, fast and experimental reactors • New and spent fuel transportation both within countries and between countries • Design studies for fissile material transport containers • Spent fuel handling and long-term pool storage • Spent fuel dry storage • Fuel consolidation and dry cell handling • Fuel dissolution • Chemical separation involving mixer-settlers and pulsed-columns • Reactor core loading assessment • Product finishing and storage • Waste treatment and handling • Waste storage including evaporation, vitrification, encapsulation and consolidation • Plutonium metal production and handling • Reactor analysis for all thermal reactor types • Burnup credit analyses	
Key phenomena	3D neutron transport phenomena, including parallelisation, whole core depletion	
Usage by	Industry, TSOs, universities, regulators	
Knowledge gaps	Need for thermal feedback, coupling with TH codes.	
Ongoing research and validation activities	Continuing extensive V&V against experimental test programmes, international benchmarks, plant data	

Table B.3.14: **OpenFOAM V7**

Name:	**OpenFOAM V7**	
Brief description of code/method	OpenFOAM is a framework for developing application executables that use packaged functionality contained within a collection of approximately 100 C+ libraries. OpenFOAM is shipped with approximately 250 pre-built applications that fall into two categories: solvers, that are each designed to solve a specific problem in fluid (or continuum) mechanics; and utilities, that are designed to perform tasks that involve data manipulation. The solvers in OpenFOAM cover a wide range of problems in fluid dynamics, as described in Chapter 3. Users can extend the collection of solvers, utilities and libraries in OpenFOAM, using some pre-requisite knowledge of the underlying method, physics and programming techniques involved.	
Publication	https://cfd.direct/openfoam/user-guide Herb, J. (2019), "Coupled OpenFOAM-ATHLET simulations of the primary circuit of a liquid sodium cooled reactor", OpenFOAM Workshop 2019, Duisburg, Germany, 23.-26.07.2019	
Developer/maintained organisation	OpenFOAM Foundation	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☒ Mechanical codes and structure analysis	☐ Fuel performance
	☒ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☒ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	Dependent on the solver	
Usage by	GRS, others	
Ongoing research and validation activities	Coupling with ATHLET Use within EU ESFR-SMART project (http://esfr-smart.eu)	

Table B.3.15: **PARCS**

Name:	**PARCS**	
Brief description of code/method	3D reactor core simulator which solves the steady-state and time-dependent, multi-group neutron diffusion and SP3 transport equations in orthogonal and non-orthogonal geometries.	
Publication	Joo, H. G., D. Barber, G. Jiang, T.J. Downar (1998), "PARCS, A Multi-Dimensional Two-Group Reactor Kinetics Code Based on the Nonlinear Analytic Nodal Method," PU/NE-98-26, Sept. 1998 PARCS, NRC – v3.3.1 Release	
Developer/maintained organisation	University of Michigan; the core simulator is supported by the USNRC	
Type (*select all applicable boxes*)	☒ Neutronics, type: 3D steady-state and time-dependent, multi-group neutron diffusion approximation and SP3 transport approximation	
	☐ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:

Table B.3.15: **PARCS** *(cont'd)*

Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	PWR, BWR, VVER, HTR, SFR, ADS (last two with GRS extensions) Limited by nuclear data	
Key phenomena	3D neutron kinetics, more than two energy groups. With GRS extensions: radial (diagrid) and axial (fuel/cladding) thermal expansion of SFR; time-dependent external distributed neutron source for ADS simulation; coupling with ATHLET	
Usage by	Original/official PARCS: E.g. by CAMP members through USNRC, universities, research centres. Improved PARCS: by GRS in frame of BMWi research projects, EU projects, IAEA CRP.	
Knowledge gaps	SFR specific 3D neutronics requirements, e.g. energy group structure, void effect modelling accuracy (neutron streaming), thermal expansion modelling. Concerns over gaps and uncertainties in nuclear data as it applies to advanced reactor designs.	
Ongoing research and validation activities	USNRC: new MG CMFD (Coarse-Mesh Finite Difference accelerator) NEM method; implementation of Source Expansion Nodal Method (SENM); general development activities for SFR. At GRS: addressing the knowledge gaps (see above) and including high-fidelity pin-by-pin/subchannel thermal hydraulic SFR safety assessment developments. Assessment against EBR-II benchmarks. Assessment of nuclear data (gaps and uncertainties as it applies to advanced reactors).	

Table B3.16: **PRIMSTER**

Name:	**PRIMSTER**	
Brief description of code/method	A custom multi-channel flow and heat transfer model for the analysis of prismatic VHTR cores. Designed to provide accurate temperature feedback to neutronic analysis. Incorporates multiscale (whole-core, fuel element and TRISO-particle scale) temperature models.	
Developer/maintained organisation	Wood	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☐ Transient (e.g. AOO)
	☐ Accidents	☐ Severe accidents
Range of applicability	Whole-core thermal modelling of prismatic VHTR reactors. Multiscale (TRISO particle, fuel element, whole core).	
Key phenomena	Single-phase flow in channels, detailed 3-D heat transfer.	
Usage by	Industry in support of Generation IV projects.	

Table B3.17: **PRONGHORN**

Name:	**PRONGHORN**	
Brief description of code/method	Porous fluid flow and solid-state heat conduction coupled with multi-group diffusion	
Publication	Pronghorn Theory Manual	
Developer/maintained organisation	Idaho National Laboratory	
Type (*select all applicable boxes*)	☒ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	Limited to single phase. Modelling is focused on core thermal-fluids.	
Key phenomena	Thermal-fluids, reactor physics.	
Usage by	Regulators, TSOs, plant designers.	

Table B.3.18: **RELAP-5/mod3 and RELAP5-3d**

Name:	**RELAP-5/mod3 and RELAP5-3d**	
Brief description of code/method	Transient and accident analysis code. Whilst developed principally for light water reactor application, RELAP5 provides a reliable single-phase gas modelling capability which has been successfully applied to a range of helium-cooled systems, including helium-cooled fast reactor designs.	
Publication	https://relap53d.inl.gov www.nrc.gov/about-nrc/regulatory/research/safetycodes.html	
Developer/maintained organisation	Idaho National Laboratory (RELAP5-3D) US NRC (RELAP5/mod3)	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☒ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	Whole-plant modelling for normal operation and design-basis accidents.	
Key phenomena	1-dimensional and simplified 3-dimensional two-phase fluid flow, heat transfer,	
Usage by	Industry, TSOs, universities, regulators	

Table B.3.19: **SAM**

Name:	**SAM**	
Brief description of code/method	System analysis tool similar to SAS4A/SASSYS-1. Aims to provide fast running, whole-plant transient analyses capability with improved-fidelity for SFR, LFR, and MSR/FHR.	
Developer/maintained organisation	Argonne National Laboratory	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☒ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☒ Severe accidents
Key phenomena	Integrated system performance	
Usage by	Regulators, TSOs	

Table B.3.20: **SCALE**

Name:	**SCALE**	
Brief description of code/method	Performs calculations in the areas of reactor physics, criticality safety, radiation shielding, and spent fuel characterisation for nuclear facilities and transportation/storage package designs. Also being used to generate source term for advanced reactors.	
Publication	SCALE code system	
Developer/maintained organisation	Oak Ridge National Laboratory	
Type (*select all applicable boxes*)	☒ Neutronics, type:	
	☐ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☒ Fission product behaviour and transport	☒ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☒ Severe accidents
Range of applicability	Limited by nuclear data	
Key phenomena	Reactor physics and source term.	
Usage by	Regulators, TSOs, plant designers	
Knowledge gaps	Concerns over gaps and uncertainties in nuclear data as it applies to advanced reactor designs	
Ongoing research and validation activities	Development of accident source term for advanced reactors. Assessment of nuclear data (gaps and uncertainties as it applies to advanced reactors)	

Table B.3.21: **SERPENT**

Name:	**Serpent**	
Brief description of code/method	Multi-purpose three-dimensional continuous-energy Monte Carlo particle transport code	
Publication	Leppänen, J. et al. (2015), "The Serpent Monte Carlo code: Status, development and applications in 2013", *Annual Nuclear Energy*, 82, pp. 142-150.	
Developer/maintained organisation	VTT Technical Research Centre of Finland, Ltd.	
Type (*select all applicable boxes*)	☒ Neutronics, type: 3D continuous energy Monte Carlo method, depletion	
	☐ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☐ Transient (e.g. AOO)
	☐ Accidents	☐ Severe accidents
Range of applicability	Traditional reactor physics applications, including spatial homogenisation, criticality calculations, fuel cycle studies, research reactor modeling, validation of deterministic transport codes, etc. Multi-physics simulations, i.e. coupled calculations with thermal hydraulics, CFD and fuel performance codes Neutron and photon transport simulations for radiation dose rate calculations, shielding, fusion research and medical physics Limited by nuclear data	
Key phenomena	General purpose, steady state reference criticality and depletion neutronics modelling, generation of few-group macroscopic cross sections.	
Usage by	Many organisations around the world.	
Knowledge gaps	Concerns over gaps and uncertainties in nuclear data as it applies to advanced reactor designs	
Ongoing research and validation activities	Coupled multi-physics simulations, transient extension Assessment of nuclear data (gaps and uncertainties as it applies to advanced reactors)	

Table B.3.22: **SOKRAT-BN**

Name:	**SOKRAT-BN**	
Brief description of code/method	Integrated code is intended for the simulation of neutron physics, thermal-hydraulic, thermomechanical processes, the fission products transport in SFRs in abnormal operation, DBA and BDBA without core degradation. The code has a block architecture and includes: - models of elements of the hydraulic network (channel, chamber, pump, local and distributed hydraulic resistances, diaphragms, valves); - models of control and management systems (sensor, trigger, controller, commands, controller for opening the gate of air heat exchanger); - models of regulators of feedwater flow and power. The main modules are: SOFAR-TH, QUASIK, RTOP-BNO, TVEL-BN and TRANS-FP. SOFAR-TH is a channel (one-dimensional) module of thermal hydraulics with a two-phase, heterogeneous, two-speed model of the coolant flow. This module simulates the operation of equipment, including flow parts of heat exchangers, the core, pipelines, circulation channels for water, sodium coolants and gas volumes. To simulate the structural materials, r-z geometry is used. TRANS-FP is a module for calculating the transfer of radioactive fission products and activated corrosion products in the primary circuit and gas system of the SFR. Empirical correlations are used to describe the transport processes and the behaviour of activated corrosion products based on experimental data obtained on BN-600. QUASIK is the module of neutron point kinetics. RTOP-BNO is a module for determining the geometric and thermomechanical parameters of a fuel rod in the approximation of plane deformation and pressure under the cladding of a fuel rod in normal operation. TVEL-BN is a module for calculating the stresses and deformations of the claddings of fuel rods in conditions of violation of normal operation and emergency conditions.	
Publication	Kudashov, I.G. et al. (2014), "Models of heat exchange of the SOCRAT-BN code for calculation of sodium boiling in channels of different geometry", *Journ. Atomic Energy*, Volume 117, Issue 5.	
Developer/maintained organisation	IBRAE RAN Institute for the Safe Development of Atomic Energy Russian Academy of Sciences	
Type (*select all applicable boxes*)	☒ Neutronics, type: Point kinetics	
	☒ Thermal hydraulics	
	☒ Mechanical codes and structure analysis	☐ Fuel performance
	☒ Fission product behaviour and transport	☒ Severe accident phenomena
	☒ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☐ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☒ Severe accidents
Range of applicability	SFRs in abnormal operation, DBA and BDBA without core degradation.	
Usage by	IBRAE, OKBM, others.	
Ongoing research and validation activities	Yes. A second version of the code is being developed, including three-dimensional neutron dynamics, core degradation, and fuel melting.	

Table B.3.23: **TORT-TD/ATHLET, TORT-TD/ATTICA3D**

Name:	**TORT-TD/ATHLET, TORT-TD/ATTICA3D**	
Brief description of code/method	3D few-group steady state and transient neutron transport (SN) and fine-mesh diffusion, coupled with ATHLET and porous-medium 3D TH code ATTICA3D	
Publication	Seubert, A., K. Velkov, S. Langenbuch (2008), "The time-dependent 3D discrete ordinates code TORT-TD with thermal-hydraulic feedback by ATHLET models", *International Conference on the Physics of Reactors, Casino-Kursaal Conference Center, Interlaken, Switzerland, September 14-19, 2008.* Seubert, A., A. Sureda, J. Lapins, J. Bader, E. Laurien (2012), "The transient 3-D transport coupled code TORT-TD/ATTICA3D for high-fidelity pebble-bed HTGR analyses", *Transport Theory and Statistical Physics,* 41:1-2, pp. 133-152. www.grs.de/en/simulation-codes/tort-td	
Developer/maintained organisation	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH	
Type (*select all applicable boxes*)	☒ Neutronics, type: 3D few-group fine-mesh transient discrete ordinates (SN) and diffusion method	
	☐ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	VHTR of pebble bed type. Short (e.g. control rod ejection/withdrawal) and long-term transients (e.g. PLOFC, DLOFC with recriticality). Asymmetric spatial neutronics.	
Key phenomena	Xenon/Iodine dynamics, decay heat calculation, Leakage and buckling calculation over larger spatial regions (e.g. spectral zones)	
Usage by	GRS, TU Munich, TU Dresden, KIT (INR), U Stuttgart (IKE), Pennsylvania State University, BAPETEN (Badan Pengawas Tenaga Nuklir di Indonesia)	
Ongoing research and validation activities	Application and improvements for high-fidelity safety assessment of pebble-bed VHTR in combination with improved cross section generation strategies and coupling with 3D core thermal hydraulics; verification and validation	

Table B.3.24: **TRACE**

Name:	**TRACE**	
Brief description of code/method	System code. Thermal-hydraulics solver. Coupled to PARCS and FAST for neutronics and fuel performance modelling, respectively.	
Publication	TRACE V5.0	
Developer/maintained organisation	Oak Ridge National Laboratory	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☒ Accidents	☐ Severe accidents
Range of applicability	May need modification to implement thermal-physical properties of materials applicable to advanced reactors.	
Key phenomena	Two-phase flow.	
Usage by	Regulators, TSOs	

Table B.3.25: **WIMS**

Name:	**WIMS**	
Brief description of code/method	WIMS is a general purpose code for deterministic reactor physics and neutronics analysis. Utilises nuclear data in a 172 group scheme group. Various international data libraries, JEFF, ENDF, CENDL, etc. Also has fine group libraries containing 1968 groups for use with the ECCO cell code for fast spectrum systems. Modelling capabilities range from a simple pin cell calculation of reactivity to whole core estimates of power and flux distributions. Extensively used for SFR applications. Flexible lattice (Cartesian and hexagonal) New feature – applicable for extension to SMR variants	
Publication	www.answerssoftwareservice.com	
Developer/maintained organisation	Wood	
Type (select all applicable boxes)	☒ Neutronics, type:	
	☐ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☒ Transient (e.g. AOO)
	☐ Accidents	☐ Severe accidents
Range of applicability	Experiments, benchmarks, plant performance and safety assessments Example applications include: • UOX, MOX, metal and special fuel assemblies • Pin, pellet, plate and pebble geometry options • Lattice cell calculations with pins on a square or triangular pitch and for cluster geometry with pins on concentric rings • Whole core calculations for water and gas-cooled systems • Fuel depletion analysis using both deterministic and Monte Carlo methods • Design and analysis studies for novel and Advanced Generation IV reactor systems • Optimisation of fuel loading and reloading • Temperature co-efficient calculations • Poison worth calculations • Gamma heating studies • Fine structure effects due to local absorbers and end-gaps • Generation of neutronics data for use in PANTHER whole-core analysis for steady state and transient conditions • Experimental comparisons • Internal benchmarking by comparing deterministic and Monte Carlo options • Criticality searches and surveys, for example to optimise fuel pool storage • Analyses of PWR UOX and MOX core fuel loadings	
Key phenomena	Neutron transport. 2D lattice calculations, 3D neutron transport using 3D CACTUS method of characteristics including parallelisation	
Usage by	Industry, TSOs, universities, regulators	
Knowledge gaps	Thermal expansion, key in fast reactors	
Ongoing research and validation activities	Continuing extensive V&V against experimental test programmes, international benchmarks, plant data	

Table B.3.26: **WIMSTER**

Name:	**WIMSTER**	
Brief description of code/method	A custom CFD model for the analysis of pebble bed reactor core thermal hydrualics. Designed to provide accurate temperature feedback to neutronic analysis. Incorporates multiscale (whole-core, single pebble and TRISO-particle scale) temperature models and a multi-batch fuel temperature modelling (i.e. calculates heat transfer between old and fresh fuel pebbles).	
Publication	Described in IAEA CRP-5 reports	
Developer/maintained organisation	Wood	
Type (*select all applicable boxes*)	☐ Neutronics, type:	
	☒ Thermal hydraulics	
	☐ Mechanical codes and structure analysis	☐ Fuel performance
	☐ Fission product behaviour and transport	☐ Severe accident phenomena
	☐ System code	☐ CFD/High fidelity
	☐ Other specific types	Type:
Technical area	☒ Steady-state	☐ Transient (e.g. AOO)
	☐ Accidents	☐ Severe accidents
Range of applicability	Whole-core thermal modelling of pebble bed reactors. Multiscale (TRISO particle, pebble, whole core)	
Key phenomena	Single-phase flow in porous bed, detailed heat transfer	
Usage by	TSO, regulators	

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Regulatory Perspectives on Analytical Codes and Methods for Advanced Reactors

Analytical codes and methods play a critical role in the design and safety analysis of nuclear reactors, helping to evaluate complex systems against potential severe events. This NEA report addresses the regulatory requirements for these tools and their application in reactor design and safety analysis.

Informed by a comprehensive survey of experts drawn from the NEA Working Group on the Safety of Advanced Reactors (WGSAR), the report highlights key areas such as verification, validation, and uncertainty estimation. It discusses differences in regulatory approaches, particularly in code certification and user qualifications, and the necessity for simulation tools to address Generation IV reactor-specific phenomena.

For more detailed insights, the report covers common positions on confirmatory analysis, providing guidance on code requirements for Generation IV reactors. It also outlines safety-relevant phenomena across various reactor concepts and emphasises the importance of simulation codes in future reactor development projects.

Nuclear Energy Agency (NEA)
46, quai Alphonse Le Gallo
92100 Boulogne-Billancourt, France
nea@oecd-nea.org **www.oecd-nea.org**

NEA No. 7617