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Identification and categorisation of safety issues for ESNII reactor concepts. Part I: Common phenomena related to materials

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ABSTRACT

With the aim to develop a joint proposal for a harmonised European methodology for safety assessment of advanced reactors with fast neutron spectrum, SARGEN_IV (Safety Assessment for Reactors of Gen IV) Euratom coordination action project gathered together twenty-two partners' safety experts from twelve EU Member States. The group consisted of eight European Technical Safety Organisations involved in the European Technical Safety Organisation Network (ETSON), European Commission's Joint Research Centre (JRC), system designers, industrial vendors as well as research & development (R&D) organisations.

To support the methodology development, key safety features of four fast neutron spectrum reactor concepts considered in Deployment Strategy of the Sustainable Nuclear Energy Technology Platform (SNETP) were reviewed. In particular, outcomes from running European Sustainable Nuclear Industrial Initiative (ESNII) system projects and related Euratom collaborative projects for Sodium-cooled Fast Reactors, Lead-cooled Fast Reactors, Gas-cooled Fast Reactors, and the lead–bismuth eutectic cooled Fast Spectrum Transmutation Experimental Facility were gathered and critically assessed. To allow a consistent build-up of safety architecture for the ESNII reactor concepts, the safety issues were further categorised to identify common phenomena related to materials. Outcomes of the present work also provided guidance for the identification and prioritisation of further R&D needs respective to the identified safety issues.

1. Introduction

European energy and climate policies are aiming at increasing share of different low-carbon energy technologies to deliver secure, competitive, and sustainable energy supply in the EU (European Strategic Energy Technology Plan (SET-Plan), 2007). The objective is achieving an 80–95% reduction in domestic green-house gas emissions by 2050 compared to 1990 (COM, 2011). Alongside wind, solar, smart electricity grids, bioenergy, and carbon capture and storage, nuclear fission has been identified as one of the six prospective technologies to be deployed towards this end (Sustainable Nuclear Energy Technology Platform (SNETP), 2007).

To demonstrate technical and economic feasibility of a sustainable nuclear fission power with fast neutron spectrum reactors operated in closed U–Pu fuel cycle, the European Sustainable Nuclear Industrial Initiative (ESNII) has been launched in 2009 (European Sustainable Nuclear Industrial Initiative (ESNII), 2015). ESNII brings together stakeholders from EU Member States, including industry, research organisations and academia, with the aim to have these technologies ready for industrial deployment by 2040. The reactor concepts considered within ESNII are:

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Nomenclature

ADS	Accelerator-Driven Systems
AIM1	Austenitic Improved Material 1
ALFRED	Advanced Lead Fast Reactor European Demonstrator
ALLEGRO	European Gas-cooled Fast Reactor Demonstrator
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
BOL	Beginning-of-Life
BOR-60	Fast Experimental Reactor – rated power 60 MW_{th}
BN-600	Fast Neutron Reactor – rated power 600 MW_e
BN-800	Fast Neutron Reactor – rated power 880 MW_e
CDT	Central Design Team for a Fast-spectrum Transmutation Experimental Facility
CEFR	China Experimental Fast Reactor
CP ESFR	Collaborative Project on European Sodium-cooled Fast Reactor
DBC	Design Basis Conditions
DEC	Design Extension Conditions
DHR	decay heat removal
dpa	displacements per atom
EBR-II	Experimental Breeder Reactor-II
EFIT	European Facility for Industrial Transmutation
EIB	European Investment Bank
ELFR	European Lead-cooled Fast Reactor Industrial Plant
ELSY	European Lead-cooled System
ESFRI	European Strategy Forum on Research Infrastructures
ESNII	European Sustainable Nuclear Industrial Initiative
ETSON	European Technical Safety Organisation Network
EU	European Union
EUROTRANS	European Research Programme for the Transmutation of High Level Nuclear Waste in an Accelerator Driven System
FALCON	Fostering ALfred CONstruction Consortium
FASTEF	FAst Spectrum Transmutation Experimental Facility
FBTR	Fast Breeder Test Reactor
FFTF	Fast Flux Test Facility
FP	Framework Programme
FP6	6th Framework Programme
FP7	7th Framework Programme
FWTC	Feed Water Temperature Control
GFR	Gas-cooled Fast Reactor
GIF	Generation IV International Forum
GoFastR	European Gas Cooled Fast Reactor Project
GUINEVERE	Generator of Uninterrupted Intense NEutrons at the lead VEnus REactor
HFR	High Flux Reactor
HLLW	High-level Long-lived Radioactive Waste
HTR	High Temperature Reactor
HX	heat exchanger
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
ISI	in-service inspection
ISI&R	in-service inspection & repair
LBE	Lead-Bismuth Eutectic
LD50	median lethal dose
LEADER	Lead-cooled European Advanced DEmonstration Reactor
LFR	Lead-cooled Fast Reactor
LIPOSO	LIaison POmpe SOmmier (pump-to-core pressurised pipe)
LMAC	Liquid Metal Assisted Creep
LME	Liquid Metal Embrittlement
LMR	Liquid Metal Cooled Fast Reactors
LOCA	Loss-of-Coolant Accident
MA	minor actinide
MAXSIMA	Methodology, Analysis and eXperiments for the "Safety In MYRRHA Assessment"
MEGAPIE	MeGAwatt PIlot Experiment
MOX	Mixed-Oxide Fuel
MYRRHA	Multipurpose hYbrid Research Reactor for High-tech Applications
N/A	not available
ODS	Oxide Dispersion Strengthened
PFBR	Prototype Fast Breeder Reactor
PPP	Public-Private Partnership
PSA	Probabilistic Safety Assessment
PWR	Pressurised Water Reactor
R&D	Research & Development
RVACS	Reactor Vessel Auxiliary Cooling System
SA	sub-assembly
SARGEN_IV	Safety Assessment for Reactors of Gen IV
SFR	Sodium-cooled Fast Reactor
SG	steam generator
SGTR	steam generator tube rupture
SGU	steam generator unit
SNETP	Sustainable Nuclear Energy Technology Platform
SPX1	Superphénix
V4G4	Visegrád-4 for Generation-4 reactors group
XT-ADS	eXperimental facility demonstrating the technical feasibility of Transmutation in an Accelerator-Driven System
β	thermal expansion coefficient
β_{eff}	effective delayed neutron fraction

- *Sodium-cooled Fast Reactor (SFR)*, represented by the 1500 MW_{th} (600 MW_e) Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) and developed by a Consortium led by CEA (France). The ASTRID project has available budget of 650 million euro granted by the French government to complete basic design phase of ASTRID by 2019, when a decision to build is to be taken. The total budget of the ASTRID project, which also includes supporting R&D facilities, is 5 billion euro. ASTRID is proposed be built in France and shall start operating by 2025 (Gauché, 2013).
- *Lead-cooled Fast Reactor (LFR)*, represented by the 300 MW_{th} (125 MW_e) Advanced Lead Fast Reactor European Demonstrator (ALFRED). In cooperation with other European partners, ALFRED is developed by members of FALCON (Fostering ALfred CONstruction) Consortium composed of Italian, Romanian, and Czech organizations (ANSALDO, ENEA, RATEN-ICN, CV Řež). Total project costs are estimated at 1.4 billion euro and envisaged to be covered by public–private partnership funding (PPP), including European Investment Bank (EIB) loan, EU's structural funds, national funding, and partners' in-kind contributions. Romanian Government has expressed its interest to host ALFRED in the Pitesti region, aiming at starting construction at around 2025 (Alemberti et al., 2013a,b).
- *Gas-cooled Fast Reactor (GFR)*, represented by the small-power (tentatively 75 MW_{th}) ALLEGRO GFR demonstrator and developed by members of V4G4 (Visegrád-4 for Generation-4 reactors) group (VÚJE – Slovakia, ÚJV – Czech Republic, MTA EK – Hungary, NCBJ – Poland) in cooperation with CEA. The project financing structure is expected to be based on national governmental funding and contributions from EU's structural

funds, in this way covering a major part of the estimated total project cost of 1.2 billion euro. ALLEGRO is envisaged to be hosted by one of the Central European Visegrád-4 countries and is aiming at starting construction at around 2030 (Horváth and Stainsby, 2012; Horváth, 2015).

In line with available European know-how, operating experience feedback, and technological maturity of different concepts the SFR has been identified in ESNII as the reference technological concept. Thanks to its extended technological base, LFR technology is considered in ESNII as a shorter-term alternative, while GFR technologies are pursued as a longer-term option.

ESNII also promotes development of necessary supporting infrastructures and research facilities, including a new European fast spectrum irradiation facility – **MYRRHA**[1]/**FASTEF**[2] (Abderrahim, 2013). The lead–bismuth eutectic (LBE) cooled MYRRHA shall serve as a testbed for both the LFR as well as Accelerator Driven System (ADS) technologies to, in the latter case, experimentally demonstrate feasibility of transmutation of high-level, long-lived radioactive waste (HLLW) in an ADS. MYRRHA's position as a key infrastructure of pan-European relevance has also been recognised by the European Strategy Forum on Research Infrastructures (ESFRI) (2010). With the support of other European partners, mostly in the frame of Euratom Framework Programme (FP) projects, MYRRHA is being developed under the leadership of SCK·CEN (Belgium), aiming at starting the construction in 2022. A MYRRHA zero-power mock-up facility, GUINEVERE, has already been operating since 2011. The total project costs for MYRRHA are 1.1 billion euro, out of which 60 million has been received as grant from the Belgian Federal Government for the period 2010–2014 to complete the conceptual engineering design of the facility. Many initiatives are being pursued to further establish an International Consortium and funding partnership of MYRRHA beyond 40% of capital costs already pledged by the Belgian Federal Government.

Consistently with Generation-IV goals (GIF Roadmap, 2014), the ESNII concepts aim at improving safety characteristics through maximising inherent safety characteristics as well as the use of passive systems and components. These objectives call for the application of innovative design solutions, options, and materials with limited qualification domain or with safety issues, out of which some are of a different nature than typical for light-water reactors.

SARGEN_IV coordination action project, co-financed by the Euratom 7th Framework Programme (FP7), gathered together twenty-two partners from twelve Member States consisting of European Technical Safety Organisations (i.e., the eight TSOs involved in the ETSON network), system designers, industrial vendors as well as R&D organisations with the objective to develop a commonly agreed safety assessment methodology for the ESNII concepts (SARGEN_IV, 2011). To this end, the project included several technical Work Packages to identify and categorise critical safety features associated with the four ESNII concepts, to review available safety methodologies and propose harmonised safety assessment practices, to apply the proposed methodologies on a few test cases, and to develop a European roadmap for the fast reactor safety R&D.

A specific task (Task 2.5) was also established in the SARGEN_IV project to summarize and assess the identified safety issues for the four ESNII concepts and further categorise them to several common "families". The objective was to systematise the consideration of safety issues aiming at a consistent build-up of the safety architecture, such that initiating events can be properly identified and adequate safety provisions and mitigation measures developed. In course of assessments, we adopted the categorisation according to common phenomena related to materials, i.e.:

- Fuel.
- Coolant.
- Structure, and
- Absorber.

In Part II of this paper, aspects specific to fast reactors and design solutions envisaged for the ESNII concepts together with a possible impact of safety issues on the fulfilment of Fundamental Safety Functions were also considered.

The task group consisted of eight participants: IRSN (France), ENEA (Italy), ANSALDO (Italy), AREVA (France), Bel-V (Belgium), AMEC (United Kingdom), MTA EK (Hungary), and JRC (European Commission). Representatives of three organizations, CEA (France), PSI (Switzerland), and SCK·CEN (Belgium), acted as reviewers. Each organisation contributed with its expertise, knowledge and experience from the running ESNII system projects to identify and review system-specific safety issues for SFRs (IRSN, AREVA, CEA), LFRs (ENEA, ANSALDO), GFRs (AMEC, MTA EK, PSI), and FASTEF/MYRRHA (Bel-V, SCK·CEN). This information was further complemented from other literature sources, specifically what concerns the operating experience feedback (Cacuci, 2010; Waltar et al., 2012; IAEA, 2007, 2002; Sauvage, 2009; Tuček et al., 2006).

In the Part I of this paper, we first summarise the identified safety issues for the individual ESNII concepts (Section 2), which is then followed by their review and categorisation according to the common phenomena related to materials (Section 3). Section 4 summarises the main conclusions of this work.

2. Identified safety issues for the ESNII concepts

Safety related aspects and issues were identified for the four representative fast neutron spectrum reactor concepts considered in the SNETP/ESNII Deployment Strategy: SFR, LFR, GFR, and FASTEF/MYRRHA, the latter cooled by liquid LBE. During the assessment, outcomes of the following ESNII-related recent or running system projects were in particular considered:

- *SFR:* Euratom FP7 Collaborative Project on the European Sodium Fast Reactor (CP ESFR), system-wise specifically with respect to 3600 MW_{th} oxide fuelled, pool type design configuration (CP ESFR, 2008; Fiorini and Vasile, 2011).[3]
- *LFR:* Euratom 6th Framework Programme (FP6) Specific Targeted Research Project on European Lead-cooled System (ELSY) and FP7 Collaborative Project on Lead-cooled European Advanced DEmonstration Reactor (LEADER), system-wise specifically with respect to the ALFRED LFR demonstrator (LEADER, 2009; Alemberti et al., 2013b).
- *GFR:* Euratom FP7 Collaborative Project on Gas Cooled Fast Reactor (GoFastR), system-wise in particular with respect to the ALLEGRO GFR demonstrator as well as the 2400 MW_{th} GFR2400 pilot plant (GoFastR, 2009; Horváth and Stainsby, 2012).
- *MYRRHA/FASTEF:* Euratom FP7 Collaborative Projects on Central Design Team (CDT) for FASTEF and Methodology, Analysis and eXperiments for the "Safety In MYRRHA Assessment" (MAXSIMA) (CDT, 2008; MAXSIMA, 2012).

[1] Multi-purpose hYbrid Research Reactor for High-tech Applications.
[2] Fast Spectrum Transmutation Experimental Facility.
[3] No detailed considerations were given to design solutions and/or options envisaged for the ESNII SFR prototype, ASTRID. The SARGEN_IV analysis is performed on the basis of the SFR concept described in the CP ESFR Euratom FP7 Collaborative Project.

Table 1
Power and coolant-related characteristics of the considered ESNII concepts compared to PWR. The displayed parameters are: thermal power, coolant core inlet (T_{in}) and outlet (T_{out}) temperatures, handling temperatures (T_{handl}), volume (V) and mass (M) of coolant in the primary circuit. The data for the PWR correspond to French N4 reactor series (1450 MW_e) (Durand-Smet, 1997).

System	Coolant	Power [MW_{th}]	T_{in} [°C]	T_{out} [°C]	T_{handl} [°C]	V [m^3]	M [t]
SFR (ESFR pool)	Na	3600	395	545	180–250	3154	2700
LFR ALFRED	Pb	300	400	480	380	330	3500
GFR ALLEGRO	He	75	260	530[a]	20–250	N/A	N/A
GFR2400 concept		2400	400	780[b]	N/A	~1400	~6
MYRRHA/FASTEF	Pb/Bi 45 at.%/55 at.%	100	270	410	200	418	4320
PWR	H_2O	4250	292	330	10–60	380	≈ 270

[a] For ALLEGRO GFR, the upper plenum temperature is given.
[b] For GFR2400, the upper plenum temperature is given.

Detailed design descriptions of the considered concepts can be found in the aforementioned references and they are therefore not repeated here. For convenience of the reader, power and coolant-related characteristics are given in Table 1, in comparison to representative PWR values.

2.1. Sodium-cooled fast reactor

Sodium is selected as fast reactor coolant due to:

- its superior thermal properties (heat capacity, thermal conductivity);
- its low density allowing a low pumping power;
- its good neutronic characteristics (low moderating power and capture cross-section);
- its low activation;
- its low level of corrosion of metallic structures, when pure;
- the easiness of procurement.

Reasonably sizeable operating experience and feedback, with considerable R&D and with analyses performed in the frame of the licensing process, have also been accumulated on SFRs since 1950s (ca. 400 reactor·years), specifically in France, United Kingdom, Germany, former Soviet Union and Russia, Japan, India, China, and USA. While several reactors have been shut down, such as Experimental Breeder Reactor-II (EBR-II) and Fast Flux Test Facility (FFTF), BOR-60, Fast Breeder Test Reactor (FBTR), and BN-600 are still operating, the latter being in quasi-commercial operation since 1982. The China Experimental Fast Reactor (CEFR) was connected to the electrical grid in July 2011, while BN-800 in Russia achieved criticality in June 2014. One SFR is under construction, namely Indian Prototype Fast Breeder Reactor (PFBR).

On the basis of outcomes of the CP ESFR project and in view of the operation experience gained in particular for Phénix and Superphénix (SPX1), the following safety issues have been identified for SFRs (SFR Safety Features, 2012):

- Sodium exhibits high chemical activity with both air and water. Some aerosols resulting from sodium fire are chemically toxic.
- Sodium reacts with oxide fuel.
- Sodium density (and temperature) reactivity coefficients might be positive in some core regions.[4]
- In case sodium freezes (<98 °C), mechanical stresses on structures might be exerted during melting due to expanding sodium. Sodium solidification might lead to coolant blockages.[4]
- Large quantities of coolant in the main vessel of pool SFRs may lead to complex flow patterns and interactions between the coolant and structures.[4]
- Loss of core geometry (core compaction) might lead to a positive reactivity insertion and power increase.[5]
- Ruptures of steam generator tubes might lead to overheating, over-pressurisation and consequent propagation of shockwaves in the intermediate loop able to damage the intermediate heat exchanger. The sodium-water reaction also produces hydrogen, which has to be managed.
- Sodium is optically opaque.[4]

2.2. Lead-cooled fast reactor

The choice of lead as a coolant is motivated by its high boiling point – 1749 °C – making coolant boiling during accidental conditions very unlikely, high thermal inertia of the primary circuit, good neutronic[6] as well as natural convection characteristics. Additionally, lead (as well as LBE) is characterised by low, not strongly exothermic chemical activity in the contact with air and water (Beznosov et al., 2005) and this provides an opportunity for the elimination of the intermediate circuit, reducing number of components and hence possibly also costs.[7]

Lead is considered as a more attractive coolant option than LBE mainly due to the lower amount of induced polonium activity (by a factor of 10^3–10^4) (Tuček et al., 2006) and due to the limited availability of bismuth resources, which, for economic reasons, excludes extensive use of LBE coolants (Mihara et al., 2003). In contrast, the lower melting point of LBE (125 °C) vs. that of lead (327 °C) allows operating at lower temperatures, reducing material issues and other operational challenges.

There is no operating experience and feedback on LFRs. About 80 reactor·years of experience and feedback have been accumulated during operation of LBE-cooled reactors used for Alfa/Lira-class submarines and land-based facilities in the former Soviet Union (IAEA, 2007). The related feedback as well as experience from licensing of these reactors is, however, not easily available due to the associated defence aspects.

On the basis of analyses performed in the ELSY and LEADER Euratom Framework Programme Projects, specifically with respect to the ALFRED technological demonstrator, the main safety features and issues for LFRs can be summarised as follows (LFR Safety Features, 2012; ALFRED, 2012):

[4] This is a generic characteristic for any liquid metal coolant.

[5] This is a generic characteristic for any fast reactor, among others depending on the core configuration and size.

[6] Lead moderating power is lower than that of sodium and the neutron capture cross-section is also lower (typically about three times lower for lead than for sodium). On the other hand, the energy loss in inelastic scattering is notably larger for lead than for sodium.

[7] In the ALFRED demonstrator, steam generators (SGs) are planned to be located directly in the primary reactor vessel. In this context, water interactions with heavy liquid metal coolant, in case of steam generator tube rupture, need to be carefully considered and its potentially important safety-related consequences analysed.

- Molten lead is corrosive as well as might erode structural materials.[8]
- Lead vapours are chemically toxic.
- Lead has high freezing point (327 °C) with a potential for coolant solidification. Mechanical stresses might be exerted on structures during melting due to expanding lead. Lead solidification might lead to coolant blockages.
- Accumulation of corrosion products in the coolant might lead to coolant blockages, in particular in fuel sub-assemblies.
- Large specific weight of lead and its quantities in the primary pool might, in case of external excitations, challenge structural integrity or functionality of systems or components.
- Large quantities of coolant in the main vessel of pool LFRs may lead to complex flow patterns and interactions between the coolant and structures.
- Loss of core geometry (core compaction) might lead to a positive reactivity insertion and power increase.
- Ruptures of steam generator tubes might lead to over-pressurisation of the primary side, sloshing and steam/water entrainment resulting in a positive reactivity insertion.
- Lead density reactivity coefficient might be positive in some core regions.
- Lead is optically opaque.

2.3. Gas-cooled Fast Reactor

The use of helium as primary coolant means there are no heat transfer limits associated with coolant phase change, unlike for liquid metal cooled reactors. Helium has high specific heat, is chemically inert and has also favourable neutronic characteristics, specifically negligible neutron absorption and hence neutron-induced coolant activity.

Similarly to LFR, an important issue in the design, safety analysis and licensing process is the lack of GFR operational experience and feedback for an industrial concept.

Use of helium coolant introduces issues concerning core coolability, behaviour in accidental conditions, including margins to fuel damage, and material performance.

The main safety issues for the GFR technology and specifically in relation to ALLEGRO GFR demonstrator, as identified on the basis of outcomes of the GoFastR Euratom FP7 Project, can be summarised as follows (GFR Safety Features, 2012; Safety Approach within European Consortia, 2012):

- GFR primary circuits have low thermal inertia.
- Small safety margins exist to clad melting[9] for the ALLEGRO first core (MOX fuel with stainless steel cladding). For the ALLEGRO "refractory core"/ceramic clad fuel and the GFR2400 core, margins appear to be reasonable.
- Control of helium leakages (primary circuit leak tightness) during normal operation can be difficult.
- Loss of primary pressure (which is typically at ~7 MPa)/loss of coolant accident might challenge system integrity and impair (decay) heat removal function.
- Loss of core geometry (core compaction) might lead to a positive reactivity insertion and power increase.
- High pressure issues have to be assessed, e.g., helium leaking into the containment would not condense at normal temperatures, leaving the containment pressure relatively high.
- Water or steam ingress might lead to a positive reactivity insertion.[10]
- Material performance problems might arise due to the lack of oxygen.
- In case of core meltdown, thermal and radiation effects might have a significant impact on behaviour of surrounding structures in some configurations.

For the ALLEGRO demonstrator, additional safety issues due to the small core have been identified:

- The control and shutdown sub-assemblies have a relatively high reactivity worth (>1$), so if the control rod is ejected by the high primary pressure due to a malfunction, it can initiate a severe accident;
- Losing the control and shutdown mechanism, i.e., inability to shutdown reactor automatically or on-demand, can trigger unfavourable events.

2.4. Fast spectrum irradiation facility

Lead–bismuth eutectic is chosen as a coolant for MYRRHA/FASTEF as it provides a lower melting point (125 °C) compared to lead (327 °C), allowing for lower operating temperatures (270–410 °C) than for LFRs (400–480 °C for ALFRED) limiting thus material corrosion issues due to heavy liquid metal coolant and other operational challenges. As already discussed above, similarly to lead, LBE features a low chemical activity with water and air, and these interactions do not lead to fires.

Main safety issues for the MYRRHA/FASTEF irradiation facility have been identified both for its critical and sub-critical operating modes. Many commonalities exist with safety topics identified already for LFRs while some of the issues are relaxed due to the aforementioned characteristics of LBE or smaller size of MYRRHA/FASTEF. The relevant safety issues for MYRRHA/FASTEF can be summarised as follows (Fast Spectrum Irradiation Facility – Safety Features, 2012):

- Molten LBE is corrosive as well as might erode structural materials.[11]
- Lead (as well as bismuth[12]) vapours are chemically toxic.
- Accumulation of corrosion products in coolant or coolant solidification might lead to coolant blockages.
- In case LBE freezes (<125 °C), mechanical stresses on structures might be exerted both during solidification and melting.
- Sizeable quantities of radiotoxic ^{210}Po and spallation products (for an ADS) are produced and must be adequately confined.
- Large specific weight of LBE and its quantities in the primary pool might, in the case of external excitations, challenge the structural integrity or functionality of systems and components.
- Large quantities of coolant in the main vessel may as well lead to complex flow patterns and interactions between the coolant and structures.
- Loss of core geometry (core compaction) might lead to a positive reactivity insertion and power increase.[13]
- Ruptures of heat exchanger/steam generator tubes might lead to over-pressurisation of the primary side, sloshing and steam/water entrainment resulting in a positive reactivity insertion.

[8] This, apart from the impact on the structural integrity of components wetted by liquid lead, might also involve reactivity insertion effects due to material removal at elevated temperatures during accident conditions.

[9] More generally, issues due to high operating temperatures need to be adequately addressed in normal and accidental conditions. Further detailed studies are required to confirm acceptable clad temperature limits.

[10] On the other hand, the coolant reactivity void effect is low (typically < 1 β_{eff}).

[11] This, apart from the impact on the structural integrity of components wetted by liquid LBE, might also involve reactivity insertion effects due to material removal at elevated temperatures during accident conditions.

[12] Toxicity of bismuth is considered to be less of an issue than that of lead (NEA, 2007).

[13] In the ADS operating mode, this issue could be present to a lesser extent.

- Rupture of the accelerator beam window might lead to a containment bypass.
- LBE is optically opaque.

3. Categorisation of material-related safety issues

To systematise the consideration of safety issues summarised in Section 2 for the individual ESNII concepts, this Section further discusses the issues and categorises them according to the identified common phenomena with respect to materials – fuel, coolant, structure, and absorber. The objectives are to further facilitate:

- a coherent treatment of initiating events and associated accident sequences;
- a consistent identification of measures and provisions to be implemented to accomplish the fundamental safety functions, and;
- an identification of technical options for prevention, control, and mitigation of possible consequences of their impairment.

3.1. Fuel

Mixed oxide (MOX) fuels (U, Pu)O_2 are being considered as reference start-up core fuels for all ESNII systems (i.e., SFR ASTRID, LFR ALFRED, GFR ALLEGRO, and MYRRHA/FASTEF). The reason is the large available operating experience feedback, qualification domain of these fuels in the fast reactor spectra, and experience with fabrication, reprocessing, and handling of material and waste. The oxide fuels feature a high thermal stability[14] as well as chemical stability, but the thermal conductivity of oxides is significantly lower[15] than that of carbide, nitride or metallic fuels, resulting in comparatively high temperature gradients in the fuel pellet.

One of the objectives of the future ESNII systems is an improved nuclear waste management, including the incineration of minor actinides (MAs). Specifically, in France, this aim is stipulated by the 2006 law on the sustainable management of radioactive materials and waste (Law no. 2006-739, 2006).

For the purpose of MA burning, two ways are currently considered for ESNII prototypes/demonstrators:

- Homogeneous strategy when a relatively limited proportion of MAs (a few %) is dispersed in the core fuel; and
- Heterogeneous strategy when a sizeable fraction of MAs (>10%) is dispersed in blanket or target sub-assemblies (SAs). These contain MAs stabilised with fertile (^{238}U) or inert matrix materials, either ceramic or metallic (such as ^{92}Mo).

The consequences of these MA burning strategies for safety are:

- For the homogeneous strategy: change of reactivity coefficients in the comparison to a reference core configuration without MAs in the fresh fuel. Particularly, the absolute value of the Doppler effect becomes smaller, the coolant void worth increases, and the effective delayed neutron fraction decreases, which pose concerns with respect to the reactivity control for the management of accidental conditions.
- For the heterogeneous strategy: important increase of both the initial heat of the fabricated SAs with MAs as well as the decay heat after the irradiation compared to the SAs with no initial MA content, having an impact on handling. In this case, the impact on safety parameters (and in particular reactivity coefficients) needs to be carefully evaluated as well.
- Additionally, the chemical reactivity between coolant and MA-containing fuels has to be assessed and adequately addressed (cf. also below).

In the long-term perspective and in the frame of a double-strata fuel cycle strategy, burner reactors specifically dedicated to burning of HLLW are also being considered with the main goal to reduce the amount of HLLW from LWRs. Among burner reactors, the consideration of the above safety issues led to a concept of inherently sub-critical ADS, such as is the case of MYRRHA/FASTEF, where both reactivity effects and delayed neutron fractions have more limited impact on system's transient characteristics. As such, ADSs are studied to confirm whether they would allow the loading of a high HLLW content per unit.

Other aspect to consider is the compatibility of the MOX fuel with the coolant. In case of SFRs, sodium interacts with MOX fuel and two situations have to be studied:

- When the fuel is **solid**, the contact between fuel and sodium in case of a clad rupture may lead to voluminous reaction products able to increase the rupture size, resulting in more fuel damage and possibly fuel dispersion in sodium. This reaction is enhanced for temperatures above 500 °C, but becomes very slow and clad rupture does not propagate in case of decreased temperatures (about 400 °C). The reaction is stopped when the reactor is shut down. Consequently, the clad rupture has to be detected and the failed sub-assembly has to be removed out of the core;
- When the fuel is **molten** (in the case of core meltdown), a thermodynamic interaction might lead to an enhanced vaporisation of sodium (as well as the structure material to a lesser extent) leading to pressure increases and mechanical loads on the primary circuit, possibly challenging its integrity (especially of the vessel and roof). Additionally, pressure driven fuel motion and consequent recriticalities need to be considered.

The above mentioned interaction with the solid fuel is typical for MOX fuel and sodium and will not occur in case of carbide, nitride or metallic fuels and other coolants considered for ESNII systems: Pb, LBE, and He. However, the thermodynamic interaction with the liquid fuel has to be addressed for any fuel type (oxide, nitride, carbide, or metallic).

Nitride, carbide, and metallic fuel have also specific safety issues, both during reactor operation (e.g., fuel-cladding interaction & increased potential for a fuel–coolant interaction for high thermal conductivity fuels) and during fuel cycle process (e.g., risk of fire for carbide fuel).

Concerning the fuel for the demonstration core of the ALLEGRO demonstrator for which a silicon carbide fibre-reinforced silicon carbide (SiC/SiC_f)[16] cladding is considered, a necessity exists to avoid pellet-clad interactions. This together with high clad operating temperatures (~1000 °C) imposes operational and safety challenges on the fuel performance (both in nominal and transient situations) and such fuel designs require further qualification steps.[17]

3.2. Coolant

The requirement of a fast neutron spectrum for the efficient fertile fuel utilisation (breeding) and actinide waste burning implies the use of coolants with suitable neutronic characteristics (low

[14] High melting temperature: about 2700 °C for a MOX fuel with 20% of Pu.
[15] About 3 W/m/K at 1000 °C for a MOX with 20% of Pu.
[16] Lined by a refractory metal.
[17] The qualification steps are: (1) Out-of-pile testing; (2) In-pile testing in the existing reactors, e.g., in the HFR in Petten or in BOR-60; (3) Final qualification in the starting core of ALLEGRO.

Table 2
Characteristics of reactor coolants used in the ESNII concepts (SFR, LFR, GFR, MYRRHA/FASTEF) compared to water (PWR). The displayed parameters are: reference/typical temperature (T_{ref}), reference/typical pressure (p_{ref}), mass density (ρ), freezing point at atmospheric pressure (T_{freez}), boiling point at atmospheric pressure (T_{boil}), specific heat capacity (c_p), thermal conductivity (k), and saturation pressure (p_{sat}) at T_{ref} (Review of international cooperation and between different liquid metal systems, 2011).

System	Coolant	T_{ref} [°C]	p_{ref} [MPa]	ρ [g/cm^3]	T_{freez} at p_{atm} [°C]	T_{boil} at p_{atm} [°C]	c_p [kJ/kg/K]	k [W/m/K]	p_{sat} [Pa]
SFR	Na	400	0.1	0.856	98	882	1.28	72	52
LFR	Pb	400	0.1	10.51	327	1749	0.147	17	$2.9 \cdot 10^{-5}$
GFR	He	400	7	$5.0 \cdot 10^{-3}$	−272	−269	5.2	0.25	N/A
MYRRHA/FASTEF	Pb/Bi 45 at.% /55 at.%	300	0.1	10.33	125	1670	0.146	13	$3.1 \cdot 10^{-5}$
PWR	H_2O	300	15.5	0.727	0	100	4.18	0.6	$8.6 \cdot 10^{6}$

moderating power, long neutron diffusion length, and low neutron absorption): Na, Pb, LBE, or He. Liquid metals due to their low partial pressure (correlated to available margin to boiling) allow operating the system close to atmospheric pressure, while gas cooled systems require high pressures in order to achieve an adequate heat transfer capability.

Basic properties of these coolants are summarised in Table 2. More detailed comparison of coolant properties is detailed in Ref. (IAEA, 2002).

The considered or typical coolant operational ranges are 395–545 °C for European Sodium-cooled Fast Reactor concepts (CP ESFR), 400–480 °C for LFR ALFRED, 260–530 °C for GFR ALLEGRO (however much higher for the 2400 MW$_{th}$ GFR concept: 400–780 °C), and 270–410 °C for MYRRHA/FASTEF.

3.2.1. Thermal and thermal hydraulic properties

3.2.1.1. Freezing/boiling point. As indicated in Table 2, the margin to **coolant boiling** is much greater for lead and lead–bismuth cooled systems than for SFRs, which makes coolant boiling very unlikely since system structures would disintegrate and/or melt well before the onset of boiling. On the other hand, helium occurs in single, gaseous phase and the appropriate heat transfer is achieved by pressurising, typically to about 7 MPa.

The **freezing temperature** of sodium is 98 °C, while 125 °C for LBE and 327 °C for Pb. In all cases, the coolant solidification must be prevented, especially in shutdown, decay heat removal situations as well as during handling operations in order to prevent the degradation or the loss of heat transfer capability (via forced or natural convection) and to ensure the integrity of fuel elements, core structures and other components. The latter requirement is also relevant from the point of view of the investment protection. Critical points are contacts with cold surfaces, where the circulation is not sufficient, and the solidification might occur due to a lack of or because of an insufficient heat source. The excessive cooling might occur in heat exchangers during the nominal operation, during operational transients or decay heat removal.

During nominal and transient conditions, the coolant temperature must be continuously monitored and secondary/tertiary feed-water conditions (either through feed-water inlet temperature or feed-water mass flow rate) well controlled. The actuation logic and performance control of the DHR systems need to be well established as well. Additionally, during shutdown states (and in particular during handling operations), in case decay heat is not sufficient, coolant must be externally heated.

For **SFRs,** depending on flow conditions and power of pumps, the frictional heat delivered by pumps is usually enough to avoid freezing (i.e., to keep sodium temperature in the range of 150 °C). However, a particular attention has to be paid to draining pipes in the secondary circuits to ensure that they are not blocked by frozen coolant, when the secondary circuits have to be drained. Up to now, the pipe heating has been accomplished by electrical heaters.

The risk of sodium freezing has also to be considered if sodium is used for removing the decay heat. In this case, sodium freezing in the coldest areas of the circuits leads to failure (partial or complete) of DHR function.

Sodium expands as it melts (sodium density is 0.968 g/cm^3 at room temperature, while 0.927 g/cm^3 at melting point), see also Fig. 1, and might therefore create mechanical loadings on systems, structures and components, possibly impairing their functionality or challenging their integrity.

For **LFRs**, the risk of freezing of the coolant is specifically relevant due to the high melting point (327 °C) of lead. Consequently, all components in the primary circuit shall be designed to assure a sufficient circulation of liquid lead in all parts of the primary circuit. Necessary features for a heat input should be foreseen to keep lead at the required temperature at least in all circumstances when the liquid state is required, including the planned shutdown and during emergency conditions. As a consequence, the control of secondary feed water conditions is of a particular importance.

Also, consequences of any possible depressurisation of secondary circuits (at nominal and accidental conditions) need to be evaluated with respect to the risk for the coolant solidification.

In **ALFRED LFR**, due considerations need to be given to the aforementioned aspects in order to assure that lead remains in the liquid state when required and that the coolant circulates sufficiently in all parts of the primary circuit.

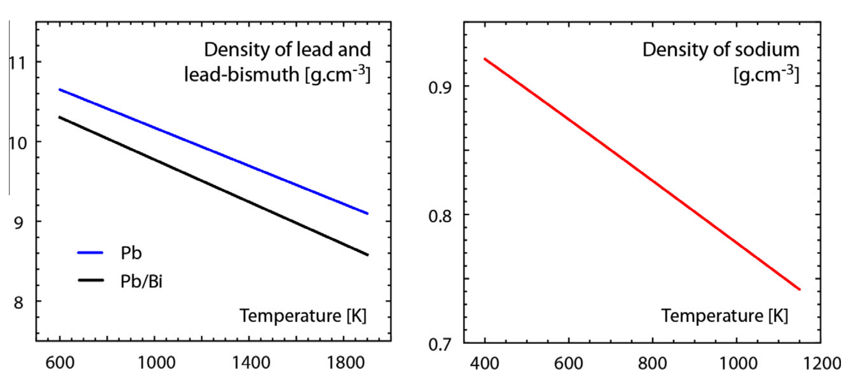

Fig. 1. Coolant density as a function of temperature for liquid metal coolants – Pb, LBE and Na (Tuček et al., 2006).

For this purpose, ALFRED is equipped with a Feed Water Temperature Control (FWTC) at the steam generator inlet to ensure that feed water temperature is not lower than 335 °C. The complete depressurisation of the secondary side (e.g., for outages) is performed such that the feed water pumps are stopped, emptying the SGs (still at high pressure), which are then depressurised. The small amount of steam shall not lead to the solidification of the lead coolant.

During shutdown states and maintenance outages decay heat given off by radioactive isotopes of the core could be enough to keep lead above its melting point. At Beginning of Life conditions (BOL) or during long outage periods, when decay heat is not enough to keep lead molten, a non-safety grade auxiliary heating system is included in the ALFRED design in order to ensure the minimum temperature of lead by transmitting heat from the secondary system. Due to the design with very limited heat losses, the investigation whether pumps rotation would be enough to keep lead molten is also ongoing for ALFRED LFR.

Similarly to sodium, lead expands when melting and the integrity of primary circuit structures and components shall not be challenged by the solidification/re-melting, taking into account volume changes when solidifying/melting the coolant and the presence of impurities leading to oxides and corrosion products, which cannot be re-molten. In ALFRED, all components inside the pool shall be designed to withstand with margins loads exerted by expanding lead. The structural analysis shall confirm the fulfilment of this design criterion.

Despite of the lower melting point for LBE vs. lead, the risk for coolant freezing/solidification needs to be considered also for **FASTEF/MYRRHA** and adequate measures implemented thereafter.

As for ALFRED, the secondary feed water conditions (either feed water HX inlet temperature or feed water mass flow rate) under nominal and transient conditions must be again carefully monitored to assure that secondary system temperatures remain above the melting point of LBE (125 °C). Additionally, a spurious activation of the secondary circulation pumps or spurious cooling via Reactor Vessel Auxiliary Cooling System (RVACS) must also be prevented. For FASTEF, it is expected that keeping the primary pumps in operation can help to avoid a too fast freezing of LBE.

As sodium and lead, LBE expands as it melts, but contrary to Na or Pb solid LBE also recrystallizes and increases in volume, i.e., expands, with further decreasing temperature below that corresponding to the freezing point of LBE (NEA, 2007; Stankus et al., 2008). Issues concerning the functionality and structural integrity of systems, structures, and components need to be again considered.

For **GFRs**, there is no issue associated with coolant freezing/solidification since the coolant is in single phase at all times.

During the commissioning of the liquid metal cooled reactor (LMR) concepts, when the primary circuit (without fuel subassemblies) is filled with liquid metal, all surfaces having contact with the liquid metal must be preheated. Appropriate commissioning procedures therefore need to be developed.

For LMRs, possible failures due to vapour deposits on structural wall surfaces (e.g., at annular penetrations in the reactor roof of SFRs) should be properly addressed. This aspect is considered to be less of an issue for lead and LBE cooled reactors due to the lower partial vapour pressure, but an adequate attention needs to be paid to impurities and dust formation specifically for LBE-cooled systems (cf. Section 3.2.2).

3.2.1.2. Thermal inertia. Volumetric heat capacity (ρc_p) of liquid metals is high, roughly 1.10 J/cm^3/K for sodium and even about 40% higher for heavy liquid metals (1.51 J/cm^3/K for LBE and 1.54 J/cm^3/K for lead). In this perspective, the volumetric heat capacity of helium coolant is negligible (0.026 J/cm^3/K).

For liquid metal cooled systems, the high volumetric heat capacity combined with the inventory of the coolant (Table 2) available in the primary circuit provides a high thermal inertia, which contributes to a slow down of any transient related to a loss of forced coolant mass flow or loss of heat sink. Pool design configurations with higher available coolant inventories present, in this respect, advantages over loop-type configurations.

On the other hand, the inherent thermal inertia in GFRs is very low as it is provided only by the core and materials of supporting structures. In the case of the depressurisation of the primary circuit, it is essential that either a forced circulation is maintained within the primary circuit or that a minimum pressure adequate for a sufficient natural circulation is kept.[18]

3.2.1.3. Heat transfer/removal. **Liquid metals** have high thermal conductivities (in the range of 72 W/m/K for sodium and 13–17 W/m/K for LBE and Pb). This enhances heat transfer from the fuel cladding to the coolant allowing limiting temperature gradients between the cladding and bulk coolant. Heat flux through cladding is higher for an SFR than for an LFR, but as heavy liquid metals have more than five times lower conductivity compared to Na, the maximum temperature difference between the outer cladding and bulk coolant is usually similar in both systems and typically in the range of 10–20 °C at the clad upper part.

For **helium**, the corresponding temperature difference is 100–150 °C at typical nominal operating conditions of 7 MPa. Due to the high coolant outlet temperatures, the difference between cladding melting and peak operational temperature (i.e., the corresponding safety margin) is therefore small. In this case, the effect of the high temperatures and primary flow direction needs to be also considered, in particular what concerns the operation and reliability of safety-related systems and components for achieving the adequate reliability over the plant lifetime and for the spectrum of considered accident scenarios (including the possible coolant heat-up accidents and resulting deterioration of the material performance). Specific examples include the design and provision of shutdown systems and particularly the location of the absorber systems.

Additionally, the risk of damaging structures if they are heated-up by hot helium has to be addressed. This could occur for example in case of a loss of heat sink, which would lead to temperature increase at the core inlet.

3.2.1.4. Natural convection capability. The liquid metal coolants have:

- a large volumetric expansion coefficient (variation of the density as a function of the temperature, see Fig. 1) with a value for sodium ($\beta_{Na} = 2.4 \cdot 10^{-4}$ 1/K) about twice as high as for lead and LBE ($\beta_{Pb} = 1.2 \cdot 10^{-4}$ 1/K and $\beta_{LBE} = 1.1 \cdot 10^{-4}$ 1/K, respectively);
- the possibility to operate in a large range of temperatures, typically a few hundred degrees, without excessive material corrosion/erosion in short-term (typically a few days), until:
  - coolant boiling (for sodium), and/or
  - onset of fast creep for structural materials (all ESNII concepts).[19]

The large volumetric expansion coefficient is also provided by gaseous coolants ($\beta_{He} = 1/T \approx 1.0–1.3 \cdot 10^{-3}$ 1/K at nominal, isobaric ideal gas conditions).

[18] By provision of a guard vessel, for example.

[19] An example value for T91 at End-Of-Life conditions (burn-up of 100 GWd/t and pin gas pressure of 5 MPa) is 847 °C at which clad failure is to be expected in ~30 min (Schikorr et al., 2012).

These characteristics enhance the possibility of core cooling by the natural convection (i.e., without any active system such as a pump/blower), for which however pressure losses in the primary circuit need to be adequately compensated by buoyancy forces. Considering that the driving buoyancy force $\propto \beta\rho(T_{outlet} - T_{inlet})$, where T_{outlet} and T_{inlet} are core outlet and inlet coolant temperatures, respectively, the potential for the development of a good natural convection flow exists in particular for LMRs. On the other hand, He density is low ($5.0 \cdot 10^{-3}$ g/cm^3 at 400 °C and 7 MPa), which limits the strength of the buoyancy force in GFRs.

As employed DHR systems for all ESNII concepts at least partly rely on the natural circulation, the stability and degree of the established natural circulation in all envisaged operating modes must be such as to first ensure that cladding temperatures can be kept within acceptable limits.

In this respect, particular attention shall be paid to **interaction between fluid and structures**, which in view of large quantities of coolant inside the main vessel of the liquid metal cooled pool type reactors, might possibly lead to complex flow patterns and structure behaviour. This might result in an occurrence and evolution of asymmetric flow patterns with possible instabilities (e.g., free level oscillations), thermal stratification, thermal stresses, and fatigue. These issues need to be evaluated comprehensively.

Tests performed in particular on Phénix, Superphénix, and EBR-II **SFRs** have shown that such natural convection is possible (Sackett, 1997; IAEA, 2013). The natural circulation is predicted to be also well established in **LFR** and **FASTEF** primary systems. This is due to the simple flow path design and due to neutronic characteristics of lead and LBE that allow larger pin pitches and lower coolant velocities (NB. the latter shall also be kept below 2 m/s to control corrosion/erosion of wetted structures, cf. Section 3.3.1), together resulting in low pressure drops (cf. also Section 3.2.3). Safety analyses performed for ELFR and ALFRED (Bubelis et al., 2013a,b) indicate that the DHR system (i.e., the Isolation Condenser which is a passive system based on an evaporation and condensation, already adopted in other nuclear plants), is able to remove decay heat and the natural circulation of lead coolant can be maintained.[20]

For **GFRs**, as discussed above, the degree of the natural convection at nominal pressure is already significantly lower than that for liquid metal cooled systems and it will be further deteriorated when pressure decreases, e.g., during loss-of-coolant accident (LOCA). There is hence the need for the implementation of a "close" (guard) containment and potentially also for the use of a heavy gas injection system to improve natural circulation characteristics.

3.2.2. Sensitivity to and control of impurities

The structure of impurities and their amount in the coolant depend on the type of structural materials, operating mode, plant design and its purpose. During the plant operation, impurities can be formed due to the diffusion of structural material components through protective oxide films, corrosion and erosion processes caused by the coolant interaction with structural materials and formation of new elements in the irradiated coolant. Additionally, air may enter into the circuit from cover gas in case of the depressurisation of the cover gas system during reactor refuelling, repair operations, and heat exchanger/steam generator leaks.

The formed solid particles might be carried by the coolant flow, they might relocate and deposit inside the sub-assemblies with a possibility to subsequently create coolant blockages. Additionally, in presence of impurities the coolant viscosity would increase with an increased pressure drop as a consequence.

In sodium, impurities typically lead to the formation of sodium oxide or sodium hydride due to oxygen or hydrogen, respectively, while for lead and LBE, there is the potential for the formation of an excessive amount of lead oxides.

Based on the experimental experience gained at ENEA facilities (CHEOPE III, LECOR, and CIRCE) in Brasimone (Italy), the dust formation might be a particular issue to control for the LBE-cooled systems. This dust and other impurities might possibly cause pipe occlusions, loops' malfunctions or blockages of gas piping. On the other hand, lead appears to be less sensitive to this issue and no solid impurities have been observed (nor any operational issues identified) in flowing Pb even after 10,000 h of operation.

Consequently, the main steps to be adopted for LMRs in order to ensure the reliable plant operation (with respect to the coolant and impurities) are:

- control of coolant parameters and quality;
- control of concentration of dissolved oxygen in the coolant;
- removal of solid oxides and other impurities from coolant (e.g., using a cold trap); and
- purification and control of the cover gas.

The reader is referred to Section 3.3.1 for a more detailed discussion on corrosion and erosion issues as well as those related to the oxygen control.

For **GFRs**, contrary to liquid metal cooled reactors, the attention needs to be paid to consequences of the lack of oxygen. A possible absence of protective oxide films on structural materials might consequently lead to an increased material degradation due to gaseous impurities potentially present in the gas (such as CO, CO_2, CH_4, H_2, H_2O), and to the subsequent failure or loss of the functionality of some systems or components (Corwin et al., 2005). Findings from High Temperature Reactor (HTR) projects could provide a useful input as these issues have already been identified and addressed there.

3.2.3. Pressure

For pool-type liquid metal cooled reactors, due to physical characteristics of the coolants, the pressure in the cover gas can be maintained close to the atmospheric pressure, while in the pool itself the pressure depends on the hydrostatic level. As discussed above, GFRs, due to thermal characteristics of the gas coolant, have to operate at high primary pressures (e.g., 7 MPa).

In past designs of SFRs, the separation of cold and hot zone was accomplished by a dedicated wall (e.g., so called Redan for SPX1), but for recent SFR designs, a concept without a wall separating the cold and hot sodium is also under investigations. The separation of cold and hot zones by a dedicated wall is currently not envisaged for present LFR designs (ALFRED, ELFR), utilising the high density characteristics of lead coolant, for which systems can be designed such that the difference between hot and cold coolant levels is small (e.g., 1.5 m corresponding to a hydrostatic pressure difference of ~0.15 MPa).

In past **SFR** designs, due to higher coolant velocities (5–6 m/s) and lower pin pitches, pressure drops were in the range of 0.3 to 0.6 MPa. This necessitated the implementation of a pressurised pipe connecting the pump and core supporting structure/diagrid (so called LIPOSO for Phénix and Superphénix). In recent designs, shorter and robust connection of the pump to the integrated diagrid-strongback core support structure is provided (Fiorini and Vasile, 2011).

Due to favourable neutronic characteristics of **lead** and **LBE** that allow larger pin pitches and hence lower coolant velocities (1–2 m/s)[21], pressure drops are in the range of 0.1–0.2 MPa and a

[20] As an example, the natural convection core flow stabilises at about 27–28% of the nominal core flow as a consequence of the Unprotected Loss-of-Flow in the 1500 MW$_{th}$ ELFR core.

[21] Consistently with the requirement to control the material erosion, cf. Section 3.3.1.

dedicated pressurised piping connection of the pump to core supporting structure is not necessary (LEADER, 2009; Lemekhov et al., 2013).

For **GFRs**, in view of the high design pressure and requirement to maintain the leak tightness of the primary circuit (i.e., retain He), monitoring, identification of the location, as well as the control of helium leakages are of the particular importance. These issues and characteristics of helium (which does not condense) must also be considered in the containment design process.

3.2.4. Induced radioactivity, coolant activity

The irradiation can in some materials lead to the formation of radio-nuclides that should be confined or their production limited from a radioprotection point of view. These nuclides could complicate inspection and maintenance of the reactor, and its future decommissioning.

For SFRs, the only stable isotope of **sodium** is ^{23}Na, which due to neutron-induced reactions leads to the production of:

- ^{24}Na (β^- emitter), which quickly decays to stable ^{24}Mg (with a half-life of 15 h);
- ^{22}Na (β^+ emitter), which decays to ^{22}Ne with a half-life of 2.6 years, but with a reaction rate much lower than that leading to the formation of ^{24}Na.

These two radioisotopes do not pose particular difficulties, when the reactor is in operation, because the coolant remains in the main vessel. Several days of shutdown provide sufficient time to lower ^{24}Na contents adequately for the maintenance and repair to be carried out on the primary circuit. This aspect is not of a particular concern for the sub-assembly handling, which is performed remotely.

A possible drawback interconnected with **LBE** is the accumulated radioactivity in LBE, mainly due to the formed polonium, which could pose difficulties during fuel reloading or maintenance on the primary circuit and the spallation target. Po isotopes, which are dominantly α-emitters, are formed mainly as a product of (n,γ) and, in case of ADS, (p,xn) reactions according to the following processes:

$$^{209}\mathrm{Bi}(n,\gamma) \rightarrow {}^{210}\mathrm{Bi}(\beta^-) \rightarrow {}^{210}\mathrm{Po}\ (\mathrm{T}_{1/2} = 138\ \text{days}), \quad \text{and}$$

$$^{209}\mathrm{Bi}(p,xn) \rightarrow {}^{208}\mathrm{Po}\ (\mathrm{T}_{1/2} = 2.9\ \mathrm{y}) \quad \text{and} \quad {}^{209}\mathrm{Po}\ (\mathrm{T}_{1/2} = 102\ \mathrm{y}).$$

Because of the high radiotoxicity of polonium (the median lethal dose LD50 $\approx$ 1 μg) its behaviour and confinement is of utmost importance with respect to the safe operation and post-irradiation handling of systems, structures, components, and materials which are in the vessel (e.g., during in-service inspections and repairs, ISI&R) as well as in case of accidents. The first estimations of the coolant activation of the FASTEF facility led to a ^{210}Po inventory of a few kilograms. The important aspect is also the decay heat due to ^{210}Po, which several hours after the shutdown of the FASTEF is still at the level of decay heat due to fission.[22] Due to Po, the activity of the cover gas of LBE-cooled reactor is also considerably higher than that of SFR, which has an impact on reactor design.

When FASTEF operates in the sub-critical mode, products of spallation reactions and accompanying nuclear reactions are isotopes of practically all chemical elements, ranging from hydrogen up to polonium.[23] Apart from polonium, the attention needs to be paid to noble and volatile gases at the operation conditions of the target. With respect to the radiotoxicity, the latter concerns specifically mercury (especially ^{194}Hg), which has also low retention in LBE (cf. Section 3.2.5) and needs to be also monitored and retained from cover gas (e.g., by using filters). Additionally, even after almost complete decay of ^{210}Po, a certain amount of radiotoxicity will still exist in the coolant over very long period of time as a result of the decay of the metastable ^{210m}Bi, an α-emitter with a half-life of $3 \cdot 10^6$ years (IAEA, 2007). There are also concerns regarding extracted components that can be contaminated with solidified radiotoxic nuclides.

Pure **lead** is not exempt from the polonium formation. This is due to bismuth impurities present in lead as well as due to additional bismuth generation after an extended operation time of the reactor by the (n,γ) reaction in ^{208}Pb, followed by further activation of Bi. However, the rate of Po production is several orders of magnitude lower than in case of LBE, and consequently its decay heat is negligible in comparison to that of the fuel.

The production of ^{210}Po in lead has been evaluated during the ELSY project for a commercially available lead (C00 grade, that is constituted by Pb at 99.9985%): the total mass of Po produced after 40 years of the irradiation is about 0.9 g. Recently, an analogous evaluation for ALFRED (assuming a Pb purity of 99.985%, corresponding to the lead grade C1 in accordance with Russian standard GOST 3778-98 (Lead Specifications, 2001)) has found a lower value (about 0.4 g, at equilibrium).

The volatility of polonium appears to be lowered, typically by several orders of magnitude, through strong chemical interaction with the lead coolant (e.g., via the formation of lead-polonide) and only a very small fraction, depending on lead temperature is vaporised into the Cover Gas System (see also Section 3.2.5). This quantity has been evaluated during the EUROTRANS project: for EFIT (European Facility for Industrial Transmutation) lead cooled reactor, the estimated maximum quantity of ^{210}Po is 1.2 g (assuming a Pb purity of 99.985%, corresponding to lead grade C1), while the fraction of Po volatised into the cover gas at 700 °C (maximum lead temperature evaluated during safety analyses) is $2.4 \cdot 10^{-8}$ (EFIT, 2009).

For comparison, the MEGAPIE experiment[24] has also indicated that only a very small fraction ($6.64 \cdot 10^{-15}$) of the total Po in the spallation target escapes to the cover gas above the target (Neuhausen, 2005).

Helium does not get notably activated (threshold for $(n,2n)$ reactions is at 26 MeV).

3.2.5. Retention of volatile fission and activation products

A number of studies were performed in the 60s and 70s on the interaction between **sodium** and fission products in case of cladding failures or fuel melting. Their conclusions can be synthesised in a few below mentioned points according to the different nature of important radionuclides:

- Noble gases, particularly krypton, have negligible solubility in liquid metals.
- Iodine remains in the form of NaI under anticipated conditions such as in the presence of caesium and/or during sodium fire. NaI is stable – it does not release any free iodine even in oxidising conditions.
- If the excess of sodium is available, studies have shown that the most of iodine will be retained in sodium, with a large fraction concentrated to the gas–liquid interface.

[22] However, this might also have an advantage in preventing and/or at least delaying freezing of LBE.

[23] Based on the results of neutronic calculations using the FLUKA and ORIHET3 computer codes, the most important volatile elements which would be expected for a 200 days irradiation of the MEGAPIE target irradiated at PSI with 1.4 mA of 575 MeV protons are: Po, Bi, Pb, Tl, Hg, Xe, I, Cs, Cd, Kr, Rb and Br.

[24] The MEGAPIE target contained 920 kg of liquid lead–bismuth eutectic.

- Caesium should be expected to appear in the elemental state rather than combined with other fission products and, although more homogeneously distributed than iodine, its concentration is somewhat higher at the gas–liquid interface.
- Alkaline earth metals as strontium and barium seem to interact with the dissolved hydrogen to form relatively no-volatile species.

Less is known about the retention of fission and activation products in **Pb** and **LBE** and R&D studies are currently ongoing to assess the corresponding retention capabilities in order to evaluate related occupational hazards and possible accidental source terms. Nevertheless, a large body of literature on the chemical and thermo-physical properties of lead and its compounds with caesium, iodine as well as polonium is available (cf. References in EFIT (2009)) and give indications of relatively good retention properties of these nuclides in lead. Among the fission products, halogen containing species, for example iodine and caesium compounds, can be again important since they may be volatile. In MYRRHA/FASTEF, considerations need to be given to activation and fission products formed by high energy protons in the spallation target alongside those produced from neutron fissions in the surrounding fuel elements.

Analyses performed in the frame of the EUROTRANS FP6 project (EFIT, 2009) have determined the upper theoretical limits for the release of polonium, caesium, strontium, and iodine in case of a postulated core disruptive accident in the LBE-cooled XT-ADS design and the lead cooled EFIT design as function of the coolant temperature.[25] It was evaluated that only a small fraction (depending on the coolant temperature) is expected to be vaporised into the cover gas system. Moreover, these results indicated that the accompanying source term and the related doses inside the containment should not pose particular concerns with respect to the design of the containment.

The definition of the quantity of fission and activation products that can reach the cover gas after a core melt is of the major importance for SFRs and LFRs/MYRRHA and needs further R&D effort, for example taking into account the kinetics of fission products from fuel to coolant and then from the coolant to the cover gas.

Recently, also, an order of magnitude greater Po releases have been measured at lower temperatures below 400 °C compared to previous measurements (Neuhausen, 2014). The issue is speculated to be interconnected with the formation of two different Po oxides on the surface of LBE and requires further investigations to clarify, including its potential safety implications and requirements for operation and handling procedures for LBE-cooled reactors.

The retention capability of **helium** is non-existent.

3.2.6. Interaction with oxygen and water

The **effect of water, steam or air ingress** into the coolant needs to be considered for all ESNII systems. In case of **sodium**, important is the high chemical reactivity with water and air (oxygen).

Reaction of sodium with **water,** which is strongly exothermic,[26] might occur in the Steam Generator Units (SGU) in case of a tube rupture. The initial leak may propagate to neighbouring tubes due to wastage and overheating phenomena, which could thus lead to the over-pressurisation and propagation of shockwaves in the intermediate loop, challenging the mechanical integrity of the intermediate heat exchanger (forming a part of the second confinement barrier) and consequently also the reactor core in case of a rupture of the intermediate heat exchanger. In case of an additional rupture of the SGU shell, hydrogen produced by the sodium-water reaction will be released in the SGU building with a risk for burning or explosions.

The occurrence of sodium-water interactions needs to be monitored (usually via the measurement of hydrogen concentrations), adequately prevented, which can be done already at a design stage, e.g., by selecting the modular SG design, and mitigated, e.g., by the depressurisation of the water side and/or sodium draining.

The implementation of a power conversion using gas (e.g., nitrogen (Fiorini and Vasile, 2011)) instead of water-steam is also considered to prevent the occurrence of such a reaction in the steam generator and also its potential degradation to a more severe situation bringing into play water, sodium and air (e.g., involving consequently the risk for hydrogen explosion).

Similarly, the exothermic reaction may also occur when sodium is in the contact with **concrete**. In this case water may be released from the concrete and may react with sodium with a hydrogen production as a result. Also some constituent compounds of concrete (SiO_2 in particular) may react quickly with sodium in an exothermic way.

The chemical reactivity of sodium with **air** could lead to Na fires[27] and the production of Na aerosols as reaction products, which are chemically toxic to humans,[28] lead to the loss of visibility, and may impair the functionality or damage certain equipment (e.g., cause blockages of an equipment such as pumps, plug filters, or damage instrumentation).[29] The resulting thermal and mechanical (pressure) loads could endanger the functionality and/or structure integrity, e.g., the containment tightness. Na fires can be prevented by the inertisation and consequences mitigated by an early detection, appropriate confinement, and extinguishing (by powders).

For certain SFR components the design can anticipate and use the chemical reactivity (based on a controlled reaction between sodium and water) for the implementation of processes for the component washing, thus allowing for the inspection, maintenance and repair. In case of cleaning of spent SAs from sodium by water, the risk for hydrogen explosion should be prevented/mitigated, e.g., by the inertisation and by continuous monitoring of H_2 quantities during the cleaning process, by use of recombiners (Latgé, 2014) and by appropriate design (compartmentalisation) of buildings.

However, interventions and/or the implementation of mitigating measures on sodium circuits and components in a post-accidental situation remain difficult.

In case of **lead** and **LBE**, there are no strong exothermic reactions of these coolants in contact with water or air (Beznosov et al., 2005), which provides conditions for the possible elimination of the intermediate circuit. However, in case of steam generator tube rupture (SGTR), water interaction with lead or LBE needs to be considered and adequately prevented and/or mitigated, specifically in view of the potential for over-pressurisation of the primary circuit, sloshing and steam/water entrainment, which might result in a positive reactivity insertion.

Even though **helium** is chemically inert in contact with water or air, the effect of a large water or steam ingress needs to be considered also for GFRs. Apart from water chemistry effects on structures and fuel, a water ingress into the primary circuit of the ALLEGRO **GFR** demonstrator, for example from the ruptured heat exchanger tube(s), has to be adequately prevented and mitigated since it could lead to an insertion of a large positive reactivity.

[25] As an example for the EFIT reactor, the volatilised fractions at 700 °C for Cs, Sr, and I are $1.1 \cdot 10^{-6}$, $5.1 \cdot 10^{-14}$ and $3.7 \cdot 10^{-6}$, respectively. The fission product inventory for seven fuel sub-assemblies is 2366 g of Cs, 150 g of Sr, and 205 g of I.

[26] $Na + H_2O \rightarrow NaOH + ½H_2$ (Q = 7 MJ per kg of Na at 500 °C).

[27] $2Na + O_2 \rightarrow Na_2O_2$.

[28] Cf. Section 3.2.7.

[29] For example, impending rotation of the rotating plug (based on the operational experience of BN-350 and BN-600) or causing difficulties in the insertion of control rods (KNK-II) (IAEA, 2007).

3.2.7. Toxicity

For **SFRs**, chemical risks and, in particular with respect to the design of the containment the ones related to reaction products of sodium fire, need to be taken into account. The particular difficulty lies in the knowledge of the composition of these products[30] ($NaOH$, $NaHCO_3$, sodium oxides) when released into the environment and in the definition of the allowable concentration limits.

The latter topic seems to be an important point for future SFRs. For Superphénix safety assessments, a limiting value of 250 mg/m^3 of sodium hydroxide concentration in the environment was used, based on American studies. Currently, substantially downward-revised limiting values are however being considered: 5–10 mg/m^3 for $NaOH$ and 60 mg/m^3 for $NaHCO_3$. A clear specification of allowable limits for accidental releases of sodium aerosols to the environment seems not to be available. In any case, the final product resulting from sodium releases into the environment is $NaHCO_3$ which is not toxic for humans and environment (NB. sodium bicarbonate, $NaHCO_3$, is extensively used in cooking).

For **LFRs** and **MYRRHA/FASTEF**, possible releases of chemically toxic lead and its aerosols (lead oxide) need to be considered and properly managed as well. Even stricter general annual limit is set for the concentration of lead in ambient air by Council Directive 1999/30/EC – 0.5 μg/m^3 (Council Directive, 1999). However, due to the low vapour pressure of Pb and LBE coolants the lead concentration inside the containment during refuelling or in-service inspection operation (with vessel open) appears to be reasonably low and the containment mixing itself is expected to reduce this value to an acceptable limit in the external environment. A very rough and conservative evaluation for lead at 400 °C (without considering containment characteristics and assuming a constant atmospheric pressure and temperature for air) estimates the corresponding concentration to about 2 μg/m^3. A detailed and more realistic evaluation has to be further performed for ALFRED.

3.2.8. Opacity

Since liquid metals are opaque, the lack of visual inspection possibilities makes fuel handling, in-service inspections and repairs of internal components difficult. In LMRs, in-service inspections (ISI) are typically carried out by ultrasonic devices, for which the technology has been developed in the context of sodium-cooled fast reactor programmes. Inside the containment, ISI activities are performed during outage periods.

In **ALFRED**, each component inside the reactor vessel is designed to be removable for an in-service inspection and maintenance. As such, all ISI activities (e.g., visual observation, surface examination, volumetric examination with X-ray or ultrasonic devices) are expected to be performed out of lead and thus under the full visibility. At the same time, the upper ends of sub-assemblies are placed above the free level of lead in the cover gas. As such, refuelling operations can be done without a need for in-vessel machines.

For **GFRs**, as helium is transparent, handling operations and in-service inspections in the reactor vessel are performed easier than in liquid metal cooled reactors. However, related risks of anoxia need to be evaluated comprehensively.[31] Also, systems required during ISI and those that will be unavailable during ISI must be considered.[32] In this regard, a benefit could be taken, when possible, from a relevant experience gained during the operation of thermal neutron spectrum gas-cooled reactors, HTRs.

3.3. Structural materials

Due to exposure to service conditions, including normal as well as transient situations, material properties gradually degrade over long periods of time. These ageing mechanisms might, as a consequence, lead to a reduction of the performance or to a loss of the designed function of reactor systems, structures, and components. The examples of these processes are material corrosion/erosion (including cracking assisted by corrosion), embrittlement (including liquid metal induced embrittlement and irradiation embrittlement), creep (including liquid metal assisted creep), fatigue, and wear (fretting and cracking assisted by wear). Representative examples of these processes relevant for ESNII systems are given below.

3.3.1. Structural corrosion and erosion

The cladding materials for **SFRs** are generally austenitic steels, such as AISI 316L or 15/15 Ti stabilised steels, used for Phénix & Superphénix and which are intended to be used in the ASTRID first cores. These steels have a good thermal mechanical behaviour and a low swelling rate in the relation to the dpa rate. They are not particularly sensitive to the corrosion by sodium, even when sodium contains some (small amount of) impurities. This kind of steels can reach dpa rates in the range of 120–150 dpa.

Flowing heavy liquid metals (Pb, LBE) are however **corrosive**[33] and can induce or accelerate a material failure under a static or a time-dependent loading. **LFRs** are consequently designed to operate at a low temperature range (400–480 °C for ALFRED and 270–410 °C for MYRRHA/FASTEF) and maintain a controlled concentration of dissolved oxygen in the coolant, which has to be high enough to support the formation of a protective oxide layer (e.g., of magnetite, Fe_3O_4) on surfaces of structures and, at the same time, low enough to prevent the formation of large amounts of PbO precipitation, which might lead to the fouling and slagging of the primary system and subsequently coolant blockages, in particular in fuel sub-assemblies. For this reason, the concentration of the dissolved oxygen in the coolant has to be constantly monitored and controlled (see also Section 3.2.2). ALFRED components are designed to compensate with the sizing of heat transfer surfaces for the low thermal conductivity of the, over the lifetime, increasing oxide layers.[34]

At temperatures above 500 °C the corrosion protection through the oxide barrier seems to fail and the application of surface coatings is considered.[35] Upper core regions and heat exchanger primary coolant inlet regions are particularly sensitive, because temperatures are highest.

[30] In contact with air, sodium oxides will be transformed into sodium hydroxide ($NaOH$) and sodium bicarbonate ($NaHCO_3$).

[31] Risk of anoxia is also relevant for the other reactor concepts, which require large amount of gas (e.g., cover gas).

[32] For example, in-service inspection will be performed during shutdown, during which the refuelling system will be unavailable as the same access path is needed. These operations therefore need to be planned sequentially. After a long term shutdown the in-service inspection could be performed in air.

[33] Structural materials exposed to heavy liquid metals can corrode by a direct dissolution of steel constitutive elements (mostly nickel) in the heavy liquid metal or impurities present in the liquid metal. This process is additionally accompanied by the oxidation of steel constitutive elements. Apart from the corrosion, the potentially damaging effects of heavy liquid metals appear to originate from two other distinct processes: the Liquid Metal Embrittlement (LME) via the reduction of the ductility and fracture toughness and Liquid Metal Assisted Creep (LMAC) involving fatigue and creep.

[34] The degradation of the heat transfer needs to be carefully considered due to the low thermal conductivity of the oxide layer (of about 1–2 W/m/K). Based on experimental results with uncoated T91 at low lead flow velocities, the maximum oxide layer of about 50–60 μm was observed, which would be further reduced due to the erosion at places with higher coolant flow velocities (e.g., 1.5–2 m/s for cladding). The estimated thickness of the oxide layer on coated materials is low, at the level of a few μm for T91 and even lower for 15/15 Ti stabilised steel.

[35] For example, by aluminisation of surfaces (with Fe–Cr–Al–Y) and surface treatment by electron beam (cf. GESA technology, which has been developed at KIT). Moreover, other coating processes, widely used for conventional plants, are also under the investigation.

At any rate, the integrity of the protective layer needs to be ensured during all plant operating conditions, including long-term transients, in order to ensure the integrity of a fuel pin. These techniques are already applied in conventional plants, and an experimental program is foreseen to validate their feasibility and reliability also in the nuclear field.

To limit the **erosion** of structural materials as well as protective oxide films, velocity of lead and LBE needs to be limited to a value resulting in negligible erosion (typically 2–3 m/s). When it is not possible, such as at the tip of a pump impeller, where velocities of the order of 10 m/s need to be expected, specific materials (e.g., MAX-phase Maxthal ceramics, Ti_3SiC_2, or SiC–SiC ceramics) or dedicated coatings (e.g., with tantalum) are currently undergoing evaluations.

For the application to LFRs and LBE-cooled reactors, ferritic/martensitic steels (such as T91) and advanced ODS (Oxide Dispersion Strengthened) steels are studied, but neither these nor the coated steels are currently qualified for nuclear applications.

Therefore, for the short-term deployment, modified austenitic steel, specifically 15/15 Ti stabilised steel, has been chosen as pin clad for the first core of FASTEF/MYRRHA taking the advantage of low LBE operating temperatures, which are beneficial for corrosion control. 316L and T91 have been selected for other FASTEF/MYRRHA primary system components.

The 15/15 Ti stabilised steels are also considered as a component material for the pin cladding and grid spacers of the ALFRED LFR demonstrator, but due to the higher operating temperatures (maximum 550 °C), the protection of steel surfaces (of clad & grid spacers) by coatings is envisaged. For the ALFRED LFR, austenitic low-carbon steels (AISI 316LN) have been selected for components at relatively low temperatures and low neutron flux (e.g., reactor vessel) while T91 is a reference material for components operating at relatively high temperatures and at high neutron flux, such as the sub-assembly wrapper.

Studies have also shown that T91 has an excellent swelling resistance (1% swelling reported in HT-9 after irradiation at 420 °C for 200 dpa) compared to austenitic steels, but their creep resistance decreases significantly above 500 °C. ODS steels seem allow dpa rates higher than 120–150 dpa, but further experimental evidence needs to be gathered to validate these as well as ferritic/martensitic steels and coated materials for the use in nuclear applications, specifically their behaviour under the influence of liquid metal and their mechanical properties thereafter. As apart from tensile and fatigue properties of T91 and 316L for MYRRHA, data to address heavy liquid metal degradation effects on candidate structural materials in design rules and standards for ALFRED and MYRRHA are inadequate and/or not available, 316L(N), T91 and 15/15 Ti behaviour need to be better understood and corresponding data base completed as a matter of priority (Gorse et al., 2011; Design Rules for Heavy Liquid Metal Cooled Fast Reactors, 2014).

In **GFRs**, material performance related problems might arise due to the lack of oxygen, as discussed in Section 3.2.2. The use of existing structural materials is envisaged for the ALLEGRO start-up core, which employs Austenitic Improved Material 1 (AIM1, a variation of 15/15 Ti stabilised steel) for the cladding, wire-spacer, and SA wrapper. Use of refractory metal lined SiC/SiC_f clad pin, which is an evolutionary design concept envisaged for GFRs, is planned to be tested in ALLEGRO in a few experimental assemblies to qualify material and sub-assembly performance (temperatures, burn-up). In this context, it is to be noted that the use of the refractory metal liner (a strong neutron absorber) introduces a reactivity penalty.

3.3.2. Irradiation behaviour

As have already been pointed out above, the irradiation can promote the material embrittlement or corrosion attack. Material properties database and the effect of the irradiation on structural materials used for the construction of safety relevant systems, components and structures in SFRs are well established.

On the other hand, R&D is still ongoing in order to define and/or refine the knowledge base, in normal and in accidental conditions, for materials under irradiation conditions typical to lead and LBE-cooled reactors. For FASTEF/MYRRHA, irradiation effects especially for components located near the core and in the vicinity of the spallation source need to be considered. For GFRs, fast neutron scattering on helium could lead to a penetration of helium into the cladding surface, which might lead to the loss of its mechanical properties. Both for LFRs and GFRs, the level of understanding of synergy between coolant and irradiation degradation effects is low. To address this requires adequate experimental programmes.

3.4. Absorber material

Up to now, boron carbide (B_4C), with boron possibly enriched in ^{10}B, was intensely used as an absorber material in SFRs.

On the basis of the past experience feedback obtained during the operation of Superphénix, following risks have been identified:

- To limit B_4C temperature, the gap between B_4C pellets and the clad may be filled with sodium (this is possible because sodium and B_4C are compatible). This design solution increases the risk for a clad failure by the carburation effect (i.e., the cladding material embrittlement by the diffusion of carbon atoms into the clad through the sodium gap).
- Due temperature gradients inside the B_4C pellets, B_4C fragments appear.

In case of a clad failure, the release of B_4C fragments inside the primary coolant is possible. Since B_4C is a very hard material, it may consequently damage rotating parts of primary pumps. The same issue needs to be considered for LFR ALFRED, GFR ALLEGRO and MYRRHA/FASTEF, which also employ B_4C as an absorber material.

Irradiation of B_4C generates ^3H. This has to be considered as well.

4. Conclusions

A dedicated task was formed within the SARGEN_IV Euratom FP7 coordination action project to review critical safety issues for the four proposed ESNII reactor concepts (SFR, LFR, GFR, and MYRRHA/FASTEF). The identified safety issues were also categorised according to common phenomena related to materials (fuel, coolant, structure and absorber) with the objective to systematize their treatment, find commonalities, similarities, and differences, in turn assisting in the development of harmonised European safety assessment methodologies for advanced fast neutron spectrum reactor systems. Within the SARGEN_IV project, the present work also provided a useful guidance for the identification and prioritisation of R&D needs respective to the identified safety issues (Identification of open issues relevant for research in the safety field and of needed R&D, 2013).

On the basis of the performed assessments and categorisations, following conclusions can be drawn:

- Some safety issues are generally common to all ESNII concepts, as those related to the choice of fuel, while others are strongly system-specific, as those related to choice of coolant and structural materials;
- As regards the choice of coolant:

  - saturation temperatures of considered coolants for ESNII concepts are diametrically different. As a consequence, safety issues associated with coolant boiling are of no concern for GFRs, while they can be considered to be of less importance for heavy liquid metal cooled systems than for SFRs;
  - liquid metal cooled systems generally feature high thermal inertia and natural coolant convection capability, with a potential to provide adequate grace times to take corrective actions in case of transients involving the loss of forced coolant mass flow or loss of heat sink. On the other hand, thermal inertia of GFRs is low since it is provided only by core and structures. Moreover, natural convection of the gas coolant can be achieved only in a pressurised state;
  - coolant reactivity with air and water is of no safety concern for GFRs (NB. with respect to water, however, neutronic reactivity effects still need to be considered). On the other hand, chemically strongly exothermic reaction of sodium with air and water warrants implementation of dedicated design provisions for SFRs. There is no strong exothermic reaction of heavy liquid metals with air and water, but further R&D is needed to understand consequences of thermodynamic steam/heavy liquid metal interaction in case of steam generator/heat exchanger tube ruptures for LFRs and MYRRHA/FASTEF;
- Coolant compatibility with structures and the associated environmental effects are of particular safety relevance for heavy liquid metal cooled reactors, for which dedicated corrosion and erosion prevention measures need to be implemented (through material protective coatings and/or control of the concentration of dissolved oxygen in heavy liquid metal). Improved understanding of underlying phenomena (especially Liquid Metal Embrittlement as well as synergetic effects) and enlarged material property database are necessary for the inclusion of heavy liquid metal environmental effects into Codes and Standards for design of mechanical components of LFRs. For SFRs, austenitic steels are not particularly sensitive to the corrosion or other coolant-induced degradation effects, provided that sodium chemistry is controlled. For all concepts, the coolant chemistry control needs to be established in the entire primary circuit, and well controlled specifically in systems, structures, and components important to safety.

Since designs of ESNII concepts are still evolving and consistently with objectives of the SARGEN_IV project, our analyses did not involve a review of the appropriateness of specific design solutions and adequacy of the safety demonstration for the individual ESNII systems. Neither was the goal of the present work to propose safety options or provisions to be implemented.

The conclusions of this paper might need revisiting in tact with narrowing down of the choice of materials for the ESNII concepts, development of design solutions, and R&D results becoming available respective to the discussed safety issues.

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