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ESFR metallic fuel study: safety approach

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PLOF, i.e., primary pumps shutdown with scram PSBO, i.e., all pumps shutdown with scram Review of the results

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Table of contents

List of tables

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Summary

The report summarizes the update of the safety approach of the ESFR concept with metallic fuel core. The ESFR concept safety approach states the methodology for the design the safety architecture and establishes that the plant design is adequately safe.

Keywords

Safety, Sodium-cooled Fast Reactor, ESFR, DiD, metallic fuel, oxide fuel

Abbreviations and acronyms

Acronym	Description
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
ATWS	Anticipated Transient Without Scram
C	Confinement
CCF	Common Cause Failure
CFV	Low void worth effect core
CPEM	Curie-Point Electro-Magnet
CSD	Control and Shutdown Device
CRDL	Control Rod Drive Line
DBA	Design Basis Accident
DBC	Design Basis Condition
DEC-A / DEC-B	Design Extension Condition (A/B)
DiD	Defence-in-Depth
DHR	Decay Heat Removal
DHRS	Decay Heat Removal System
DSD	Diverse Shutdown Device
EM	Electro-Magnet
FCI	Fuel Coolant Interaction
GIF	Generation IV International Forum
IAEA	International Atomic Energy Agency
I&C	Instrumentation & Control

IHX	Intermediate Heat Exchanger
ISAM	Integrated Safety Assessment Methodology
IVR	In-Vessel Retention
LOD	Line Of Defense
LOF	Loss Of Flow
LOHS	Loss Of Heat Sink
LOP	Line of Protection
LOOP	Loss Of Offsite Power
MLD	Master Logic Diagram
OPT	Objective Provision Tree
PCSS	Passive Core Shutdown System
PES	Practically Eliminated Situation
PIE	Postulated Initiating Event
PSA	Probabilistic Safety Assessment
QSR	Qualitative Safety features Review
RC	Reactivity Control
RCC-MR	Règles de Conception et de Construction des Matériels Mécaniques des îlots nucléaires RNR
R&D	Research and Development
RP	Reactor Pit
RPL	Reactor Pit Liner
RV	Reactor Vessel
SAF	Sub-Assembly Fault
SFC	Single Failure Criterion
SFR	Sodium Fast Reactor
SIRIB	Système d'Inhibition de Remontée Intempestive de Barres », meaning Control Rod Withdrawal Inhibition System.
SG	Steam Generators
TOP	Transient Over-Power
UCS	Upper Core Structure
ULOF	Unprotected Loss Of Flow

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ULOHS	Unprotected Loss Of Heat Sink
ULOOP	Unprotected Loss Of Offsite Power
UT	Unprotected Transients
UTOP	Unprotected Transient Over-Power
WENRA	Western European Nuclear Regulators Association

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1 Introduction

Within ESFR-SIMPLE project WP1, the objective of the Taks-1.2 is to update the safety approach for the ESFR concept developed in the past projects considering a metallic fuel core option. The safety approach describes the methodology for the design of the safety architecture and for the demonstration of the safety provisions.

The safety approach for the ESFR concept relies mainly on the European Safety Framework developed for new LWRs, in particular the European Utility Requirements and the safety approach for EPR, and on the Sodium cooled Fast Reactor operational and licensing feedback (in particular, the outcomes of the European Fast Reactor Project developed within a British, French and German partnership) [1]. Considered are also general safety recommendations established in international standards, in particular those of the International Atomic Energy Agency and the Generation IV International Forum.

The Safety related documents for ESFR concept mainly include:

- CP ESFR SP3.2 D1 Synthesis of safety requirements for ESFR and orientations for suitable R&D Rev.1 (2010-03-25)
- CP ESFR SP3.2 D2 Safety architecture to master the safety functions Rev.0 (2013-07-12)
- CP-ESFR SP3.1 D1 Safety objectives and design principle Rev.0 (2010-07-07)
- ESFR-SMART D1.1.3 Specification of the new system safety measures (Version 11 2018-07-19)
- ESFR-SMART D1.1.1 Definition of safety requirements
- WENRA statement on safety objectives for new nuclear power plants – November 2010.

The present report is based entirely on these documents and uses sections of the documents without further reference.

It has been agreed that the safety provisions implemented for the ESFR concept with the MOX fuel core remain in place. In the view of this premise, the report gives only what is deliberated in the past projects as a concise review including some deliberations regarding the metallic fuel core.

The ESFR safety approach is principally complainant with current European and international safety principles [1]:

- The approach is deterministic and complemented by probabilistic studies.
- The safety provisions are defined and dimensioned with respect to the potential risks, to assure safety objectives and principles based on defence in depth principle application. These safety provisions include provisions used to prevent accidents as well as those used to mitigate accident consequences. The impact of internal and external hazards is also considered.
- The design adequacy with respect to safety objectives and principles is demonstrated in particular by:
  - The analysis of the consequences of "dealt with" events with deterministic analysis, which allows to check the performances of the safety provisions and to design the plant equipment.
  - The “practical elimination” of a limited set of situations, which relies on the implementation of successive diverse and reliable design and operating

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prevention provisions, and allows justifying that their potential severe con sequences are not considered in the design of the plant.

- The use of methods such as Lines Of Defence and Objective Provision Tree, to verify the sufficiency of the safety provisions in regard to the safety objectives and principles.

The Systems, Structures and Components (SSC) are identified and classified with respect to their safety importance. Adequate requirements (e.g., for their design, qualification, manufacturing, operation) are defined in accordance with their safety classification.

The general safety objectives are completed by a set of more practical qualitative and quantitative safety targets to guide the safety design.

The safety objectives for the ESFR concept have been defined based on a systematic investigation of WENRA's publication on Safety Objectives for New Power Reactors.

The safety objectives are achieved by a safety design based on Defense in Depth for each plant state.

The safety approach is developed and implemented in the design at early stage.

2 Safety approach of the ESFR concept

The ESFR concept safety approach states the methodology for the design the safety architecture and establishes that the plant design is adequately safe.

2.1 Safety Objectives

The safety objectives proved the basis for the requirements for minimizing the risks associated with nuclear power plants. The IAEA safety objectives for nuclear installations as outlined in [12] are fully considered in the development of the ESFR safety approach.

These high-level safety objectives are augmented by qualitative and quantitative safety targets, which are used for the safety design of the plant. In particular, safety objectives defined in the European safety framework for new nuclear plants and technical Guidelines considered for the EPR are considered as a basis [1].

2.2 ESFR concept Defence-In-Depth

The framework for ESFR concept safety design to achieve the safety objectives is implemented through the concept of “defence in depth” (DiD). The DiD concept used in the ESFR projects is slightly deferent from that of WENRA/IAEA [6]. A mapping of the WENRA DiD onto the ESFR is given in Table 1.

	Level of DiD	Objective of the level	Essential means	Associated plant condition categories	Radiological consequences	ESFR
Original design of the plant	Level 1	Prevention of abnormal operation and failure	Conservative design and high quality in construction and operation	Normal operation	Regulatory operating limits for discharge	DBC1
	Level 2	Control of abnormal operation and failure	Control, limiting and protection systems and other surveillance features	Anticipated operational occurrences	Regulatory operating limits for discharge	DBC2
	Level 3a	Control of accident to limit radiological releases and prevent escalation to core damage conditions	Safety systems Accident procedures	DiD Level 3.a Postulated single initiating events	No off-site radiological impact or only minor radiological impact (see NS-G-1.2/4.102)	DBC3
	Level 3b	Control of accident to limit radiological releases and prevent escalation to core melt conditions	Engineered safety features (4) Accident procedure	DiD Level 3.b Selected multiples failures events including possible failure or inefficiency of safety systems involved in DiD level 3.a	No off-site radiological impact or only minor radiological impact (see NS-G-1.2/4.102)	DBC4
	Level 4	Practical elimination of situation that could lead to early or large releases of radioactive materials Control of accidents with core melt to limit off-site releases	Engineered safety features to mitigate core melt Management of accidents with core melt (severe accidents)	Postulated core melt accidents (short and long term)	Limited protective measures in area and time	DEC1
Emergency planning	Level 5	Mitigation of radiological consequences of significant releases	Off-site emergency response Intervention levels	-	Off site radiological impact necessitating protective measures	DEC2

Table 1 Mapping of WENRA DiD onto ESFR project

The assertion of the adequacy of the DiD implementation for the ESFR concept is in the realm of the safety assessment for both "Dealt with" events and "Practically eliminated" situations. For both categories, initiating events have been identified by various methods. Within the first category are considered:

- Design Basis Conditions, DBC1 – DBC4 and
- Design Extension Conditions, DEC1-DEC2.

Table 2 including the plant states, events, plant availability and indicative probabilities for the occurrence of events. Table 3 lists the ESFR DiD for the Design Extended Conditions.

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	Plant conditions	Event	Plant availability	Indicative Fr/reactor-year
DBC1	Normal operation condition	Power operation, normal transients, commissioning		
DBC2	Anticipated Operational Occurrences	initiating events might occur several times during the plant life	restart in reasonably short term after fault rectification.	$> 10^{-2}$
DBC3	Design Basis Accident	initiating events are not expected to occur during the plant lifetime (single failures)	restart following plant inspection, rectification and qualification	10^{-2} - 10^{-4}
DBC4	Design Basis Accident	initiating events are not expected to occur during the plant lifetime (Selected multiples failures)	Restart is not required	$< 10^{-4}$

Table 2 ESFR DiD concept DBC1-4

DEC1	Situations without whole core accident	**Complex sequences,** corresponding to sequences, which consider initiating events combined with failures of mitigating systems beyond those considered for DBC analysis (DBC2, DBC3, DBC4 combined with the complete failure of one or several mitigating provisions).
		Limiting events, corresponding to accident conditions, which represent bounding cases of particular fault types for which it is anticipated that they may require evaluation for licensing purposes. They are not defined based on their occurrence frequency, but they are postulated to assess specific risks related to the technology.
DEC2	Situations corresponding to whole core accidents	

Table 3 ESFR DiD for DEC situations

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Detailed rules for deterministic safety analysis, radiological and design criteria are given in [1].

2.3 Practically eliminated situations

The concept of practical elimination (PE) is implemented mainly for situations with a large or early radiological release. Typically, PE is justified:

- If it is physically impossible for the situation to occur or
- If the situation can be considered with a high level of confidence to be extremely unlikely to arise: The prevention of a particular sequence should be demonstrated primarily by deterministic arguments complemented with probabilistic, where appropriate, taking into account the uncertainties resulting from the limited knowledge about particular physical phenomena.

In addition, robust demonstrations need to be provided relying on the implementation of several successive preventive provisions.

2.4 Design Criteria

For the ESFR concept, only preliminary qualitative criteria have been established. Concerning design criteria associated with loadings to SSCs, few criteria are outlined. Table 4 lists such preliminary qualitative criteria on the fuel and cladding. In DEC2, the damage of the containment structures should not lead to radiological consequences of significant releases.

	Fuel	Fuel pin clad
DBC1	No melting	No open clad failure
DBC2	No melting	No clad failure except due to random effects
DBC3	No melting	No systematic (i.e., large number of) clad failure
DBC4	Any predicted localized “melting” to be shown to be acceptable. Simultaneous and coincident clad failure and fuel melting must be excluded	No systematic clad melting. Any predicted localized clad melting may be acceptable provided that it can be shown that it does not lead to material relocation
DEC1	No whole core accident	
DEC2	No unacceptable damage of containment structures	

Table 4 Preliminary qualitative criteria on the fuel and cladding

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For mechanical design of general SSCs, proposal is made to link RCC-MR criteria level to category of events as show in Table 5.

	Criteria level of Codes and Standards (RCC-MR design rules)			
Event category	Safety classified components	Components requalification, repair or replacement of which are not acceptable	Components the leaktightness of which is required	Active components the functional operability of which is required
DBC1	A	A	A	A
DBC2	A	A	A	A
DBC3	C	A	C	A
DBC4	D	D	C	A
DEC1-2	D	D	C	A

Table 5 to Link RCC-MR criteria level to category of events

Level A: highest level of safety margins against P-type and S-type damages

Level C: lower that Level A safety margins against P-type and buckling damages

Level D: lower that Level C safety margins against P-type and buckling damages

P-type damage: resulting from constant or monotonic loadings

S-type damage: resulting from cyclic loadings

The criteria need to be seen as recommendations. Other criteria could be used when justified.

2.5 Assessment Tools

2.5.1 Qualitative Safety Feature Review

Qualitative Safety features Review (QSR) has been performed within the ESFR-SMART project [2]. The objective of the QSR is to provide the designer with a check list summarizing the good practices and recommendations which can be useful to verify that the design details are coherent with the recommendations which are available from different sources, and applicable to the future nuclear systems. The check list is compatible with the DiD structure consistent with the international and European safety framework. However, a dedicated list of ESFR-SMART was not available. Instead a generic rather detail SFR list was used.

A dedicated list has not been elaborated in the framework of ESFR-SMART. An available existing list established for SFR has been assessed. This list needs to be validated by SFR design and safety experts and cannot be used as a reference.

The main outcome of the assessment are [2]:

- Recommendations related to the overall safety architecture of the plant are included in the level 1 of defence-in-depth, which should be only related to the prevention of abnormal operation and failures. So, the organization in defence-in-depth levels is not sufficient and not adequate for such overall recommendations. However, these general recommendations as well as recommendations about the normal operation could be relevant to judge and compare innovative design options, but results of analyses (thermo-hydraulic, thermo-mechanical…) are needed (end of ESFR-SMART or next R&D projects).
- Recommendations related to levels 2 to 3b are in general duplicated for all initiating events, the list of which is not available at the design stage of ESFR-SMART. The recommendations are also duplicated in all defence-in-depth levels.
- A list of top-level guidelines adapted to the design stage of ESFR-SMART has been established. The objective is to provide a framework useful to assess the relevance of the innovative design features proposed.

In relation to metallic fuel, the top-level guidelines remain the same and therefore the outcome of the QSR study remain valid.

2.5.2 Objective Provision Tree method

Within the ESFR-SMART project, the Objective Provision Tree (OPT) method has been used to assess the provisions implemented to ensure the fundamental safety functions for each level of DiD [2]. These provisions are grouped in Lines Of Protection (LOP) which comprise:

- the main equipment ensuring the prevention or limitation of the consequences of accidents. They are forming the so-called safety architecture of the design.
- the safety features in support to ensure the robustness of main equipment, that is of adequate performances and reliability, in particular once previous levels have failed.

The OPT method for reactivity control and core heat removal functions is in consistency with WENRA approach. Safety mitigating provisions used for core melting prevention are organized in OPT levels 1, 2, level 3a and level 3b. Specific OPT method related to confinement function has been developed.

The OPT developed for ESFR-SMART consists of [2]:

- **Level of defence**
- **Objective**: objectives consistent with WENRA objectives are proposed in the available examples. However, for ESFR-SMART, OPT is not developed in regard to initiating faults and the categories are not yet defined. So, performances of levels 2, 3a and 3b LOP cannot be compared to category criteria. For these levels, prevention of core melting accident is proposed as a general objective.
- **Challenges**: in general, this item corresponds to a qualitative degradation of the implementation of the safety function. It is proposed to indicate directly the sub-functions if any.
- **Mechanisms**: it is proposed to indicate the phenomena likely to degrade the main equipment ensuring the safety function and in particular those which can be a common cause failure with the previous levels.
- **Lines Of Protection**: it is proposed to indicate the <u>main equipment</u> (and phenomena) performing the safety function and the related <u>safety features</u> to prevent/overcome the mechanisms likely to degrade main equipment performances.

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The outcome of the OPT method application provided safety requirements for each fundamental safety function including top level guidelines, provisions for core melting and practically eliminated situation. Furthermore, the safety requirements of the implementation of the safety architecture are provided. The outcome of the OPT application proved the basis for the following discussion of the safety design approach.

3 Safety design

Based on the outcome of the OPT applications, guidelines for the implementation of the main safety functions are discussed.

3.1 Reactivity control

3.1.1 System layout

The ESFR-SMART concept implements following provisions for reactivity control [5]:

- two automatic shutdown systems (24 Control and Shutdown Devices & 12 Diverse Shutdown Devices) with a high level of diversification. Two separate and diverse I&C systems are also foreseen, each one triggering the protection systems of one control rod group (each group gathers 12 CSD & 6 DSD). These active control rods are responding to usual safety criteria, especially for diversification and single failure criteria, with two different types of control rods. The diversity should also include the monitored parameters,
- The provision of passive negative-reactivity insertion devices (12 DSD equipped with Curie Points Electro-Magnets (CPEM), and if necessary some hydraulic rods) in addition to the automatic shutdown systems. The number of hydraulic rods should be deduced from accidental transient simulations. Each of these passive shutdown devices should stop the reactor if a given threshold on a physical parameter is reached (temperature and/or flowrate) in case of an unprotected transient (reactor automatic shutdown failure), they initiate the reactor shutdown without any I&C order or electrical supply,
- The prevention of the core meltdown by valuing the core favourable natural behaviour (negative feed-back effects, low void effect core design, e.g. CFV-like core) in case of the reactor automatic actuation failure up to passive shutdown implementation if needed.

Further safety provisions against core damage accidents are:

- Low void worth core
- Corium transfer tube and
- In-vessel core catcher

These severe accident related measures may need to be further investigated in relation to the metallic fuel core. In particular, the sodium plenum efficiency should be re-evaluated as well as efficiency of the corium transfer tubes. The necessity of the In-vessel core catcher should be assessed based on severe accident analysis.

In addition, provisions for core geometry maintenance (upper pads) and surveillance (ultrasonic devices) are considered.

3.1.2 Safety objective

The reactivity control for the ESFR concept is provided by the control rod systems, which need to assure [5]:

- Sufficient sub-criticality margin for shutdown states, in any situation (fuel handling, maintenance), in any circumstances (fuel handling failure, earthquake…),
- Control the reactivity during the plant start-up, the divergence steps and when criticality is reached,
- Control the power evolutions related to normal reactivity variations during normal operation, and ensure the reactor stability through favorable inherent safety characteristics (e.g. a negative power and temperature coefficients),
- Make possible the reactor shutdown (within an allowable time) in case of normal, incidental or accidental situation,
- Guarantee the return to a coolable sub-critical state in the long term for the core-degraded situations.

More detailed overview of the safety objectives are given in [2]. **The safety objectives remain the same concerning metallic core.**

3.1.3 Safety requirements

The OPT application resulted in detailed safety requirements for reactivity control covering [2]:

- automatic shutdown systems;
- inherent core behavior;
- complementary passive shutdown systems and
- mitigation of core damage.

The requirements are intended to be completed at further design steps, especially regarding normal operation requirements, control rods worth requirements, fuel handling requirements and core melting accident mitigation requirements [2].

For the metallic core design, some of the quantitative requirements may need to be reinvestigated, in particular, practical elimination of the core meltdown.

3.1.4 Implementation and safety design options

The outcome of the OPT application is synthesized in Table 6, showing the main equipment used at each level of DiD. Some of the safety measures are used at multiple DiD level, which undermines the independence of the LOP for each level. In order to assure the independence of the LOPs some recommendations are provided in [2].

Further, the OPT approach recommends to dedicate safety equipment for each DiD level and initiator family, as illustrated in Table 7; This approach allows to identify gaps in the implemented LOPs and to propose measures for improvements. For example fast TOP cannot be adequately mitigated at all DiD level within the current safety architecture and thus has to be practically eliminated. In addition, for SAF in DiD level 2 and 3, dedicated core instrumentation could prove LOP.

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DiD Level	1	2	3a	3b	4
Components used during normal operation					
CSD	Dedicated I&C				
DSD	Dedicated I&C				
CPEM					
Hydraulic rods					
Core natural behavior					
Corium discharge tubes					
Core Catcher					

Table 6 Equipment associated to the DiD levels in ESFR-SMART

DiD Level	2	3a	3b	4
LOHS	CSD and DSD	CSD or DSD	Core natural behavior and/or **CPEM**	Mitigation devices
LOF	CSD and DSD	CSD or DSD	Core natural behavior and/or **CPEM** or **hydraulic rods**	Mitigation devices
Slow TOP	SIRIB[1], CSD and DSD	CSD or DSD		
Fast TOP				
SAF[2]			CSD or DSD	Mitigation devices

Table 7 Equipment dedicated to each initiator family for each DiD levels

[1] French acronym for « Système d'Inhibition de Remontée Intempestive de Barres », meaning Control Rod Withdrawal Inhibition System.

[2] SubAssembly Fault

3.2 Decay Heat Removal

3.2.1 System layout

The ESFR concepts aims to practically eliminate the prolonged loss of decay heat function. In order to achieve this objective redundant safety systems, diversified by design, operation mode (active and passive), heat transfer modes and the heat sinks (air and water), and physically separated are considered. Three decay heat removal systems are implemented [5]:

- The **DHRS-1,** consisting of cooling circuits by sodium/air heat exchangers connected to each of the six IHXs secondary sodium. The DHRS-1 loop operates in parallel to the secondary loop using the hot secondary sodium extracted from the IHX as the working medium. The heat is rejected to the environment using sodium/air heat exchanger located at the bottom of the air stack the largest part of which is located outside of the reactor building. The cold secondary sodium comes back to the IHX cold sodium entry. Such a scheme promotes cooling of the primary sodium in the IHX and therefore enhances the primary sodium natural convection through the core and IHX.
- The **DHRS-2,** making use of the steam generator modules to promote the cooling of SG casings by convection of atmospheric air. Even in case of loss of feed water in the steam generators and loss of electricity supply for the secondary pumps, the measures taken on the secondary loops should be able to assure an efficient decay heat removal by active or passive ways.
- The **DHRS-3** or **DHRS-Pit**, consisting of two independent active cooling systems:
    - a first circuit with oil, installed in the gap between the insulation and the reactor vessel or inside the insulation (to be decided based on the thermal calculations). The oil system will be in direct view of the main vessel. With the suppression of the security vessel, this circuit should be very efficient during normal operation of the plant. The oil under forced convection can remove the heat transferred by radiation from the reactor vessel at high temperature.
    - a second circuit with water inside the mixed concrete/metallic structure is added to be sure to maintain this structure under 70°C, even in mitigation case with a loss of the oil circuit.

In addition to these main DHR systems, further improvements are considered to enhance the decay heat removal function [5]:

- a primary pool and secondary loop design enabling an easy establishment of natural convection will be adopted;
- the use of passive electromagnetic pumps (thermal pumps) for the secondary sodium loop and the DHRS-1 using permanent magnets and thermoelectricity provided by the difference in temperatures. They need no external electricity supply and provide a flow rate also in nominal conditions able to passively assure some flow rate or increased possibilities of operation in natural convection regime.

In normal operation, decay heat removal is ensured by means of primary pumps, secondary sodium loops and a tertiary system.

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3.2.2 Safety objectives

The complete failure of decay heat removal function has to be practically eliminated through adequate safety design. The decay heat removal function is maintained mainly by cooling of the core and internal structures by primary sodium. Therefore, it key to maintain a sufficient sodium level in the primary circuit to achieve decay heat removal by forced convection or by natural circulation.

The challenges related to the achievement of the decay heat removal function, considered at the current stage of studies, are:

- Decrease of primary sodium level;
- Degradation of primary sodium circulation;
- Degradation of implementation of decay heat removal systems (either used during normal operation or used in accident conditions), with consideration of the cold shutdown state, the reactor start-up and the power operation states.

The ESFR concept implements three independent Lines Of Defence for practical elimination demonstration. In accordance with SFR licensing feedback, two strong Lines Of Defence and one medium Line Of Defence are requested for the prevention of this particular situation for ESFR concept design studies.

The safety objectives remain the same concerning metallic core.

3.2.3 Safety requirements

The application of OPT and LOD methods resulted in the following requirements [2]:

– With regard to one or several secondary loops spurious draining likely to degrade DHRS efficiency:
    - High prevention of several secondary loops draining (adequate procedures)
    - Design of DHRS-1 to maintain its performances in case of secondary loops draining
    - In case of spurious draining of one secondary sodium loop (considered as a DBC2), DHRS-1 and DHRS-2 should meet DBC4 limits without one secondary loop and with consideration of one single failure.

– With regard to primary pumps failure likely to degrade DHRS efficiency:
    - High prevention of primary pumps failure
    - In case of loss primary forced circulation (due to the primary pumps failure or Station Black-Out, both considered as a DBC4), DHRS-1 and DHRS-2 should prevent core melting accident with natural convection in the cooling loops and without the consideration of one single failure.
    - In case of a loss of offsite power (considered as a DBC2), DHRS-1 and DHRS-2 should meet DBC4 limits with natural convection in the cooling loops and with consideration of one single failure (if primary pumps are emergency supplied, primary forced convection is considered).

– With regard an IHX tube failure likely to degrade DHRS efficiency (medium term draining of the affected secondary sodium loop):
    - Prevention of IHX tube failure (DBC2 or DBC3)

- DHRS-1 and DHRS-2 should be designed to meet DBC4 limits without the degraded loop and with consideration of a single failure for both of them (case of DBC2 IHX leak)
- DHRS-1 or DHRS-2 could be designed to meet DBC4 limits without the degraded loop and without a single failure (case of DBC3 IHX leak)

– In general internal and external hazards (e.g. fire, earthquakes, flooding) need to be considered with particular attention due to the potential of common cause failure conditions, e.g.:
  - Risk of damage by aircraft crash of both DHRS1 and DHRS2 chimneys has to be addressed.
  - Very high prevention of hazards in the above roof area, likely to damage equipment used for DHRS, DHRS2 and DHRS-3 (e.g. I&C), has to be provided.

– Reactor vessel leakage likely to degrade DHRS efficiency:
  - Prevention of the reactor vessel leakage (adequate quality, surveillance of the reactor vessel)
  - Implementation of the reactor pit liner and the reactor pit with the capability to provide a sufficient primary sodium level and to withstand primary sodium loadings

– The transfer of decay heat from the core to the DHRS is performed by primary sodium natural circulation when forced circulation is no more ensured. Particular attention has to be paid to the transition between the forced circulation and the natural circulation. The hydraulic path shall always remain available

The safety requirements remain the same concerning metallic core.

3.2.4 Implementation and safety design options

The list of main equipment ensuring the safety function identified for each level of Defence-in-Depth is indicated the following Table 8.

DID Level	Sodium level maintained by	Sodium circulation maintained by	DHR function
DID Level 1	Reactor Vessel	Forced Circulation	1 secondary loop + 6 SGs
DID Level 2	Reactor Vessel	Forced Circulation	DHRS-1//DHRS-2//DHRS-3
DID Level 3a	Reactor Pit Liner	Natural Circulation	DHRS-1//DHRS-2; DHRS-2// DHRS-3; DHRS-3//DHRS-1;
DID Level 3b	Reactor Pit	N/A	DHRS-1; DHRS-2; DHRS-3
DID Level 4[3]	Reactor Pit Liner	Natural Circulation	DHRS-1; DHRS-2; DHRS-3

Table 8 Equipment ensuring the safety function identified for each level of Defence-in-Depth

[3] DHR main equipment to be tested to supply its function despite the mechanism occuring under severe accident conditions.

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3.3 Confinement function

3.3.1 System description

In the ESFR concept, the confinement function is achieved by means of [5]:

- the first confinement barrier (i.e. the fuel clad): during normal operation and in case of frequent events,
- the second confinement barrier (reactor roof, reactor main vessel, cover gas system, primary sodium purification system, heat exchangers tubes) for events with degradation of the first confinement barrier: in case of rarer events,
- the third confinement barrier (e.g. reactor containment structure and ventilation isolation devices) for events with degradation of the first and second confinement barriers: in case of very rare events and a core melting accident.

With these provisions, the ESFR concept implements adequate confinement function to prevent release of radioactive nuclides to the environment in all plant states.

3.3.2 Safety objectives

Main safety objectives for confinement function include [2]:

- Prevention of abnormal operation and failures leading to the degradation of the first confinement barrier – fuel clad,
- Ensuring confinement of radioactive material, once C-LOP1 safety features failure has occurred,
- Ensuring confinement of radioactive material, once C-LOP1 and C-LOP2 safety features failures have occurred. At this level of defence-in-depth, the first confinement barrier (fuel clad) is postulated to be failed,
- Ensuring confinement of radioactive material, once C-LOP1, C-LOP2 and C-LOP3 safety features failures have occurred. At this level of defence-in-depth, the first (fuel clad) and second (primary cooling system) confinement barriers are postulated to be failed and the bounding case is the core melting accident.

The safety objectives remain the same concerning metallic core.

3.3.3 Safety requirements

The confinement of radioactive materials shall be provided in all situations, both normal and accidental, including considered situations of general core meltdown, and for all plant states.

A containment system shall be provided to ensure or contribute to the achievement of the following safety functions:

- confinement of radioactive substances in operational states and in accident conditions,
- protection of the reactor against external natural and human induced events,
- radiation shielding in operational states and in accident conditions,
- control of radiological releases.

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A particular attention should be paid to control the risk of a containment bypass and to the performance of containment in severe accident conditions, including the resistance of the primary circuit to the mechanical energy release, the behaviour of the reactor building and the effectiveness of auxiliary systems.

The safety requirements remain the same concerning metallic core.

3.3.4 Implementation and safety design options

The confinement function is achieved by means of [2]:

– the first confinement barrier (i.e. the fuel clad): effective during normal operation and in case of anticipated frequently occurring events,
– the second confinement barrier (reactor roof, reactor main vessel, cover gas system, primary sodium purification system, heat exchangers tubes): for events with degradation of the first confinement barrier, i.e. in case of rarer events,
– the third confinement barrier (e.g. reactor containment): for events with degradation of the first and second confinement barriers, i.e. in case of very rare events, leading to core melting accident.

This approach addresses primarily the confinement of radioactive substances contained in the core. The confinement of radioactive substances contained in the primary circuit during normal operation or those that may be released during fuel handling or relocation from/to the core is not addressed here.

This specific approach is based on the implementation of confinement barriers. The following levels are defined:

– Level 1: prevention of the degradation of the first confinement barrier in normal operation,
– Level 2: maintenance of the first confinement barrier by means of implementation of reactor shutdown, decay heat removal systems, dedicated systems (e.g. corrosion risk),
– Level 3: implementation of the second confinement barrier,
– Level 4: implementation of the third confinement barrier, in particular in case of core melting accident.

The OPT application provides further details for each DiD level for the confinement function [2].

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4 Further considerations

4.1 Safety assessment

Recommendations for safety assessments are provided in the [2] covering:

- Assessment of transient from forced to natural convection and assessment of primary pumps in case of PLOOP, ULOF, PSBO;
- Additional analyses on the behavior of hydraulic diodes;
- Assessment of decay heat removal systems;
- Assessment of passive core shutdown system considering unprotected transients;
- Assessments of ULOF, ULOOP, UTOP and ULOHS.

Preliminary acceptance criteria on the fuel, cladding, reactor vessel and reactor are given Table 9.

Category	Fuel limits	Fuel pin clad limit	Vessel	Reactor pit
Normal operating conditions	No melting	No open clad failure	Below 400°C	Below 80°C
DBC2	No melting	No clad failure (Tclad < 700°C) except due to random effects.	Below 450°C	Below 80°C
DBC3	No melting	No systematic (*i.e.* large number of) clad failure (Tclad > 700°C)	Below 550°C	Below 80°C
DBC4 and complex sequences	Any predicted localized “melting” (Tfuel > 2700°C) to be shown to be acceptable. Simultaneous and coincident clad failure and fuel melting must be excluded.	No systematic clad melting (Tclad > 1320°C). Any predicted localized clad melting may be acceptable provided that it can be shown that it does not lead to material relocation. Decoupling criterion: no clad dry- out, then no sodium boiling.	Below 650°C	Below ~100°C
Very rare complex sequences and limiting events	No severe core degradation (*e.g.* no criticality risk, decay heat removal capability maintained)		Below 650°C	Below ~100°C
Severe accidents	Coolability of the damaged core within the primary system enclosure (*e.g.* sub-criticality in the long term, decay heat removal capability)		Below 650°C	Below ~100°C

Table 9 Preliminary criteria on the fuel, cladding, reactor vessel and reactor pit [2]

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The appropriateness of these preliminary acceptance criteria need to be reviewed considering the metallic fuel core. In fact, while its high thermal conductivity (compared to oxide fuel) represents generally a big advantage, the very low melting temperature provides limited margins already in operational plant states (normal operation and anticipated operational occurrences) in particular for reactivity transients. On the other side, metal fuel seems more benign during DEC by inherently mitigating the risk of core disruptive accident transients. In such transients, the high metal fuel's thermal conductivity pushes the peak temperature towards the top of the fuel, where it melts first. The associated fuel relocation, happening before clad failure due to the low melting point, occurs in lower reactivity regions of the core mitigating potential reactivity excursions. It has also been observed how liquid metal fuel forms alloys when in contact with clad material that have a melting temperature that is lower of the sodium boiling temperature, further contributing to the dispersion of fuel limiting the risk of recriticality.

4.2 Practically eliminated situations

For the ESFR concept, following situations are considered for practical elimination [2]:

- situations likely to lead to a core meltdown accident with uncontrollable mechanical energy release:
  - Significant flow of gas through the core
  - Significant core compaction
  - Collapse of the core support structures
- Situations likely to lead to the containment degradation and to fuel-sub-assembly accident with a huge radiological release:
  - Massive water ingress into the primary circuit
  - Generalized hydrogen deflagration in the radiological containment
  - Loss of the decay heat removal function
- Fuel-sub-assembly significant deterioration situations when the confinement measures may be not sufficient:
  - Core loading errors leading to fuel melting
  - Fuel-sub-assembly meltdown in spent fuel storage pool

While there are currently no additional situations to be considered in connection with the metallic fuel core, their safety demonstration should be tailored taking into account the fuel's thermo-physical characteristics.

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5 Rationale for metallic fuel

5.1 Fuel property

The main theoretical advantages of metal fuel are its higher density and better thermal conductivity compared to oxide fuel (s. Table 10) [21]. However, the increase of the theoretical density should be balanced by a significant decrease in the smeared density for metal fuel to accommodate its swelling under irradiation, which is much higher than for the oxide fuel [21].

Physical properties	Oxide PuO_2-UO_2	Metal Pu-U-Zr10
Theoretical Density [g/cm^3]	11.0	15.8
Melting point ^oK	2740	1160
Thermal conductivity W/m^{o}K	2.0	22.0
Swelling	low	high
Chemical characteristics	Poor compatibility with Sodium	Eutectic fuel-steel at 725°C

Table 10 Physical properties of MOX and metal fuel

5.2 Comparison of performance

Pervious R&D studies have shown that metal fuel provides certain advantages compared to oxide fuel [22]. The main characteristics supporting these statement including the physical justification are summarized in Table 11 [22].

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	Characteristics	Physical basis
Fuel performance	• High burnup potential; demonstrated in EBR-II; • excellent transient capabilities; • benign run beyond cladding breach (RBCB) performance; • higher breeding Potential;	• Higher heavy metal loading • High thermal conductivity • Improved neutron economy • harder spectrum • low reactivity swing
safety	• lower operating temperatures (typically 350°C inlet and 510°C outlet); • melting temperature is lower than that of oxide fuel, yet difficult to raise the fuel temperature; • better inherent safety characteristics under ATWS events; ULOF, ULOUHS & UTOP • core design with minimum burnup reactivity swing; reducing the UTOP initiator	• eutectic formation (ca 720^{0}C) • High thermal conductivity (20 W/m-K for metal compared to 2 W/m-K for oxide). • large margin to coolant boiling temperature, • large thermal inertia of the pool configuration • reactor system, and the metallic fuel properties all combine to provide a unique inherent passive safety potential • lower stored Doppler reactivity • low reactivity swing
reliability	• access for maintenance easier; • radiation exposures to plant personnel are expected to be lower; • no exposures are expected during maintenance and inspection of major components.	• radioactive corrosion products are not formed in any significant amounts.

Table 11 Metal fuel characteristics with physical justification

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5.3 Metallic core design

In concrete core design comparative studies, following findings were noted [23]:

- With metallic fuel core, higher conversion ratios with related reduced reactivity swing;
- The higher heavy metal fraction in metallic fuel allows lower fissile enrichment and better internal conversion than oxide fuel;
- Thus, the metal fuel core can satisfy nuclear goals with fewer fuel assemblies and a more compact core;
- Metal fuel heavy metal packing fraction improves: internal conversion, cycle burnup swing, fissile Pu requirement, U requirement, core diameter and height, spatial power peaking
- For same peak linear power, lower spatial peaking, in metal allows higher average linear power (less total pin);
- Achievable burnup for both metal and oxide fuels is limited by cladding dose, void swelling and embrittlement issues;
- Metal core pressure drop increases with fewer assemblies and more pins per assembly;
- Lower metal core peaking factor reduces peak temperatures and cladding damage;
- Metal core has no eutectic melting for all design basis events (up to scram). However, beyond design basis events may be limited by cladding wastage limit of 10% or may require development of barrier cladding;
- Metal core higher-pressure drop increases duct radial growth beyond conservative limit, towards empirical (experimental) limit.

6 Conclusion

Within ESFR-SIMPLE project WP1, the objective of the Taks-1.2 is to update the safety approach for the ESFR concept developed in the past projects considering a metallic fuel core option. The safety approach describes the methodology for the design of the safety architecture and for the demonstration of the safety provisions.

The safety approach for the ESFR concept relies mainly on the European Safety Framework developed for new LWRs, in particular the European Utility Requirements and the safety approach for EPR, and on the Sodium cooled Fast Reactor operational and licensing feedback. Considered are also general safety recommendations established in international standards, in particular those of the International Atomic Energy Agency and the Generation IV International Forum

No specific safety concerns have been identified in relation to metallic fuel. Some aspect may need further investigations including:

- acceptance criteria for the safety assessment;
- severe accident related investigations;
- safety provisions for DEC situations;

The outcome of the safety assessment may provide further considerations for enhancement of the current safety approach of the ESFR concept.

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