ORIENT- NM

Organisation of the European Research Community on Nuclear Materials

A Coordination and Support Action in Preparation of a Co-Funded European Partnership on Nuclear Materials

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Work Package 2 – Vision Paper and Strategic Research Agenda for an EJP on nuclear materials

Deliverable D2.1: Nuclear materials identity cards

Author(s) name and affiliation	Marjorie Bertolus, CEA	Massimo Angiolini, ENEA
	Marco Cologna, JRC	Pål Efsing, KTH
	Miguel Ferreira, VTT	Benoit Tanguy, CEA
	+ TAG members indicated in individual cards	
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List of abbreviations

AAR	Alcali Activated Materials
ACR	Alkali-Carbonate Reaction
ACS	Above core structure
AFCEN	Association française pour les règles de conception, de construction et de surveillance en exploitation des matériels des chaudières électro-nucléaires
ALD	Atomic Layer Deposition
ASR	Alkali-Silica Reaction
BN	Beloyarsk nuclear plant
BWR	Boiling Water Reactors
CANDU	CANada Deuterium Uranium reactor
CCA	compositionally complex alloys
C/M	Carbon-to-Metal (ratio)
CVD	Chemical Vapour Deposition
DB	Design-basis
DBTT	Ductile Brittle Transition Temperature
DEC	Design Extension Conditions
DEF	Delayed Ettringite Formation
dpa	Displacement per atom
EPR	European Pressurized Reactor
FIMA	Fissions per Initial Metal Atom
Gen II	Second Generation (of nuclear reactors)
Gen III/III+	Third (plus) Generation (of nuclear reactors)
Gen IV	Fourth Generation (of nuclear reactors)
GMAW	gas metal arc
GTAW	gas tungsten arc
HAZ	Heat Affected Zone
HEA	High Entropy Alloy
HIP	Hot Isostatic Pressing
IASCC	Irradiation-Assisted Stress Corrosion Cracking
ID	Identity
JOG	Joint Oxide-Gaine
LEU	Low Enrichment Uranium
LME	Liquid Metal Embrittlement
LOCA	Loss of coolant accident
LTA	Lead Test Assembly
LTO	Long-term operation
LWR	Light Water Reactor
MA	Minor Actinides
ML	Machine learning
MOX	Mixed Oxides (fuel)
MSR	Molten salt reactor
MTR	Materials Testing Reactor
NDT	Non-destructive test(ing)
O/M	Oxygen-to-Metal (ratio)

ODS	Oxide Dispersion Strengthening /Strengthened
PCMI	Pellet-cladding mechanical interaction
PLD	Pulsed Laser Deposition
PGM	Platinum Group Metal
PWR	Pressurized Water Reactors
RIA	Reactivity Insertion Accident
ROG	Rection Oxyde Gaine
SC	Steel-concrete
SFR	Sodium Fast Reactor
SG	Steam Generator
SMR	Small and medium size Modular Reactor
SPS	Spark Plasma Sintering
SS	Stainless steels
TIG	Tungsten Inert Gas
WAR	Weak Acid Resin
WWER	Water-Water Energetic Reactor

Executive Summary

This document contains the nuclear materials identity cards that were prepared by the international technical advisory group (TAG) of the ORIENT-NM project to help the identification and prioritization of materials issues to be investigated in a future partnership on nuclear materials. These ID cards indicate the (potential) use of the various materials, their degradation mechanisms in this use, as well as the data available and the gaps in knowledge.

TAG expert groups were set up for all materials families and produced materials identity cards for the following ones:

- Metallic alloys for structural components
- Fuel cladding materials
- Nuclear fuel materials
- Concrete.

This is a technical annex to the Strategic Research Agenda written in the project.

1. Introduction

One of the main objectives of the ORIENT-NM project is to produce a strategic research agenda exposing the industrial and research needs concerning nuclear materials for all nuclear fission reactor generations and the activities to perform to reach the objectives identified.

In support of the identification and prioritization of materials issues to investigate, the present document contains the nuclear materials identity cards that were prepared by the international technical advisory group (TAG) of the ORIENT-NM project.

TAG expert groups were set up for all materials families and produced materials identity cards for the following ones:

- Metallic alloys for structural components
- Fuel cladding materials
- Nuclear fuel materials
- Concrete.

For the first three categories, each card indicates, for the main materials used or envisaged to be used in nuclear reactors, the (potential) use of the various materials, their degradation mechanisms in this use, as well as the data available, the open issues and the gaps in knowledge. For concrete, the cards indicate for the various degradation mechanisms the consequences of this mechanisms and the R&D efforts made and needed on materials aspects monitoring and structural modelling.

This is a technical annex to the Strategic Research Agenda written in the project.

2. Metallic alloys for structural components

Coordinators: Pål Efsing, KTH, Benoit Tanguy, CEA

2.1 Low-alloy bainitic steels

Compositions of interest	(16MND5, 18MND5, SA533/508, 22NiMoCr37, 20MnMoNi55 A302B, 15Kh2MFA, 15Kh2NMFA) *C(%)<0.2;Mn(%) 1.15-1.55; Mo(%) 0.45-0.57; Ni(%) 0.5-0.8; Si(%) 0.15-0.3, Cu(%)<0.2 [WWER materials Cr (%) 1.2-3.00 ; V (0.10-0.35)]*		
Type of reactor	**Large reactors**	**SMR (if applicable)**	
Light water reactor	X	X	
Heavy water reactor	X	X	
Sodium-cooled fast reactor			
Lead alloy- fast cooled reactor			
Gas-cooled fast reactor			
(Very) High Temperature Reactor			
Water-cooled supercritical reactor	X		
Molten salt reactor			
Other (specify here)			
Fusion			
Component type	Reactor Pressure Vessel, steam generator vessel, secondary and primary piping, valves, pumps		
Method of manufacture	Forging, hot-rolling		
Assembly method	Welding		
Target operating environment	Up to 0.2 dpa for LWR reactor, 270-320°C (up to 390ºC (Magnox))		
Properties that make the material suitable for the target environment			
Fracture toughness against fast neutron irradiation and thermal ageing, mechanical properties at temperature, applicable corrosion properties ease of fabrication, good weldability, availability, cost, low activation			
Open questions to consider			
Creep at high temperature, DBTT (Ni-Mn-Si role), segregation, fabrication of large ingots, additive manufacturing, thermal ageing at higher temperatures and longer duration (thermal embrittlement). Embrittlement for low neutron flux and non-hardening embrittlement. References: Kolluri, M., ten Pierick, P., Bakker, T., Straathof, B. T., Magielsen, A. J., Szaraz, Z., D'agata, E., Ohms, C., Martin, O. (2021). Influence of Ni-Mn contents on the embrittlement of PWR RPV model steels irradiated to high fluences relevant for LTO beyond 60 years. Journal of Nuclear Materials, 553, 153036. https://doi.org/10.1016/j.jnucmat.2021.153036</div>			
Property improvements required or in			
Homogeneity of the ingots Iron chromium alloys with improved long term behaviour			
Level of technological readiness	9	**Quality of use**	100 %
Other observations			

List of identity card authors
Pal Efsing (KTH, Vattenfall), Benoit Tanguy (CEA), Murthy Kolluri (NRG), Marta Serrano (CIEMAT)

2.2 Micro-alloy or non-alloy C-Mn steels (base metals and welds)

Compositions of interest	C-Mn (A48, A52), C-Mn micro-alloyed (P355)		
Type of reactor	**Large reactors**	**SMR (if applicable)**	**Notes**
Light water reactor	X		PWR 900, 1300, N4, EPR
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy- fast cooled reactor			
Gas-cooled fast reactor			
(Very) High Temperature Reactor			
Water-cooled supercritical reactor			
Molten salt reactor			
Other (specify here)			
Fusion			
Component type	Main secondary piping (main steam system, feedwater supply, turbine bypass system)		
Method of manufacture	Forged, stretched		
Assembly method	Manual or orbital TIG welding, coated electrode welding		
Target operating environment	280°C, 80 bar		
Properties that make the material suitable for the target environment			
- Individual Kv > 47 J at -20°C - Average Kv > 60 J at 0°C - Average Kv > 100 J at 20°C - Re mini > 355 MPa References : RCC-M. Design and Construction Rules for Mechanical Components of PWR Nuclear Islands – Edition 2020			
Open questions to consider			
- Normal decrease in resilience observed in Heat Affected Zone (HAZ)" - Optimal chemical composition for base and filler metals for compliance with properties - Ideal welding conditions for compliance with properties References: Jorge and al. Microstructure characterization and its relationship with impact toughness of C–Mn and high strength low alloy steel weld metals – a Review. Journal of Materials Research and Technology 10 (january 2021): 471-501. https://doi.org/10.1016/j.jmrt.2020.12.006. Wang, X.L., Y.R. Nan, Z.J. Xie, Y.T. Tsai, J.R. Yang, C.J. Shang, Influence of welding pass on microstructure and toughness in the reheated zone of multi-pass weld metal of 550 MPa offshore engineering steel, Materials Science and Engineering: A702 (august 2017): 196-205. https://doi.org/10.1016/j.msea.2017.06.081. Song, H. Y., G. M. Evans, S. S. Babu, Effect of microstructural heterogeneities on scatter of toughness in multi-pass weld metal of C–Mn Steels,. Science and Technology of Welding and Joining 19, no 5 (July 2014): 376-84. https://doi.org/10.1179/1362171814Y.0000000194.			
Property improvements required or in progress			
- Compliance with the required tensile and resilience properties - Zero defects in welds - Continuous monitoring of welding parameters References: Koen Faes and Jürgen Feyaerts (Belgian welding institute - IBS), Axel Vlaminck et Eli Reekmans (Oqton Belgium) In live quality control of welding processes – To detect weld quality during or just after welding, Belgian welding institute (2020)			
Level of technological readiness	7-8	**Likeliness of**	100 %

		use	
Other observations			
- International feedback - Other industries feedback (oil and gas)			
List of identity card authors			
E. Molinié (EDF), P. Todeschini (EDF), B. Yriex (EDF), F. Villaret (EDF), J. Odinot (EDF)			

2.3 Austenitic stainless steels for Gen II &III applications

Compositions of interest	304, 316, 347 (Nb stabilized), 321 (Ti stabilized)	
Type of reactor	**Large reactors**	**SMR (if applicable)**
Light water reactor	X	X
Heavy water reactor		
Sodium-cooled fast reactor		
Lead alloy- fast cooled reactor		
Gas-cooled fast reactor		
(Very) High Temperature Reactor		
Water-cooled supercritical reactor		
Molten salt reactor		
Other (specify here)		
Fusion		
Component type	Reactor Internals, all primary piping, vessel cladding	
Method of manufacture	Forging/machining/ rolling/welding	
Assembly method	welding	
Target operating environment	Doses up to 120 dpa, deionized water, 350°C (up to 400°C)	

Properties that make the material suitable for the target environment

- Good mechanical properties at low and high temperature including creep and creep/fatigue
- Structural stability during ageing
- Weldability
- Ability to be fabricated
- Excellent corrosion behaviour in water environment (LWRs applications)

From these investigations, database suitable for 40 years lifetime.
Material codification: RCCM, DIN, ASME

Open questions to consider

- Behaviour under corrosive environment (Stress Corrosion Cracking, Irradiated Assisted Stress Corrosion Cracking)
- Swelling at high doses
- Combination of thermal and irradiation for creep
- Phase stability at high doses combined with deformation (martensite transformation)
- Environmental effect on fatigue, thermal fatigue
- Ageing & Plant Life Management
- Life assessments of components operated an load-following mode
- How to best qualify components additive manufacturing;
- Properties for components supplied by additive manufacturing: compared to conventional manufacturing (tensile, fracture toughness, fatigue, creep etc., pristine and service exposed)

References:

E. Lemaire, N. Monteil, N. Jardin, M. Doll, Contribution of Materials Investigations and Operating Experience of Reactor Vessel Internals to PWRs' Safety, Performance and Reliability
O.K. Chopra, A.S. Rao, A review of irradiation effects on LWR core internal materials – Neutron Embrittlement, Journal of Nuclear Materials 412 (2011) 195–208

Property improvements required or in progress

-weld behaviour

- Accurate multiscale modelling for long-term structural performance with respect to degradation and cracking (swelling, IASCC, fatigue) - Active survey for additive manufacturing linked with design opportunities: reducing machining, area with stress concentration, weldments…			
Level of technological readiness	8 (LWRs-more than 60 years life duration)	**Likeliness of use**	100 %
Other observations			
List of identity card authors			
P. Efsing (KTH, Vattenfall), B. Tanguy (CEA), Ł. Kurpaska (NCBJ)			

2.4 Austenitic steels of type 316SS for Gen IV applications

Compositions of interest	316L(N), 316L(N)-IG and 316L 1) X2CrNiMo17-12-2(N) nitrogen controlled (base metals) for main primary circuit components – “316L(N)”, A3.1S, properties Group in RCC-MRx, 316L(N)-IG is the low activation ITER Grade for fusion 2) X2CrNiMo17-12-2 / X2CrNiMo17-12-3 / X2CrNiMo18-14-3 (base metals) for main primary circuit components A3.3S – “316L” Chemical compositions given in RCC-MRx Section III Tome 2M Materials		
Type of reactor	**Large reactors**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor	X		
Lead alloy- fast cooled reactor	X		MYRRHA, Alfred
Gas-cooled fast reactor	(X)		Allegro project
(Very) High Temperature Reactor			
Water-cooled supercritical reactor			
Molten salt reactor			
Other (specify here)	Target of European Spallation Source		
Fusion	X		316L(N)-IG is ITER Grade
Component type	<u>Fission</u>: Vessels/Core Support: irreplaceable Above Core Structure (ACS): possibly replaceable Pipes, Intermediate Heat Exchanger, core internals <u>Fusion</u>: vacuum vessel and other vessels, Test Blanket Modules		
Method of manufacture	Forging/machining/ rolling/welding/laminated Additive Manufacturing (development)		
Assembly method	Welded		
Target operating environment	• Fission = Gen IV: Until 550°C. Sodium & lead environment • Fusion: high neutron energy + corrosive environment. 316L(N)-IG will be used for ITER vacuum vessel but has drawbacks that limit its further use in DEMO and fusion reactors (due to long term activation product and irradiation swelling at high doses)		
Properties that make the material suitable for the target environment			
The 316SS stainless steels have evolved for 316L is a low carbon version and 316L(N) is a further development for the French SFR programme with tighter restrictions. 316L/316L(N) is probably the most used. Specifications of 316L(N) resulted from previous R&D and feedback for SFR to obtain the best compromise between different required properties: • Good mechanical properties at low and high temperature including creep and creep/fatigue; • Structural stability during ageing; • Weldability;			

- Ability to be fabricated;
- Excellent behaviour in sodium (SFR applications)

From these investigations, data base suitable for 40 years lifetime.
Material codification in RCC-MRx (ed. 2018): alloy A3.1S 316L(N) and A3.3S for 316L

References:

- P. Yvon, Structural materials for GENIV Nuclear Reactor, chap 17 (F. Dalle et al.) Conventional austenitic steel as out of core materials for GENIV nuclear reactor, Elsevier 2017

- Kimura and al.: Creep strength and microstructural evolution of type 316L(N) stainless steel, ECCC creep conference, May 2014

- D. Bonne and al.: Codification of 316L(N) in RCC-MR code- Experience and prospective project, PVP2010

- M. Blat and al.: Getting the most from feedback on the past French SFR structural materials for ASTRID components, ICAPP 2015, Paper 15438

Open questions to consider

- Evolution of the data for 60 years life duration: ageing, creep, creep-fatigue, ratcheting, thermal striping.
- Low dose irradiation (<1dpa) for 60 years life duration (with He production from transmutation (ACS));
- Life assessments of components operated an load-following mode
- Behaviour in accidental conditions: higher temperature, different loading
- Weldability: new processes, new filler metal: long term behaviour
- How to best qualify components additive manufacturing;
- Properties for components supplied by additive manufacturing: compared to conventional manufacturing (tensile, fracture toughness, fatigue, creep etc., pristine and service exposed)

LFR specific

- Compatibility with liquid lead and lead-bismuth eutectic for base material and weld (liquid –metal embrittlement, corrosion, erosion

References:

- KF. Nilsson and al.: EERA JPNM Task Force 60 years operational life future reactors: review and roadmap for future activities (2017)

- M. Blat and al.: Important parameters to take into account to get reliable structural materials data for 60 years design duration, ICAPP 2016, paper 16522

- J. Aktaa and al.: Effect of hold time and neutron irradiation on the low cycle fatigue behavior of 316 CL and their consideration in a damage model, Nuclear Engineering and Design 213 (2002) 111-117

Property improvements required or in progress

Gen4/SFR :

- Creep and creep fatigue modelling: robust long-term prediction needs more comprehensive approach. Creep/fatigue: test with more realistic creep dwell to agree with operation plant deformation mechanism.
- Alternative welding processes; to obtain better properties after ageing in weld metal (no filler metal or filler metal without or very ferrite content) but also to obtain low hardness in HAZ and weld area. Final objective is to reduce or suppress weld coefficient
- Low dose irradiation and materials embrittlement: representative irradiation (combined irradiation flux and long-term ageing)
- Assess the effect of lead, lead-bismuth on mechanical properties
- Corrosion and erosion data and mechanisms in contact with liquid lead or lead-bismuth eutectic. .

General needs:

- Accurate multiscale modelling for long-term structural performance with respect to

degradation and cracking • Active survey for additive manufacturing linked with design opportunities: reducing machining, area with stress concentration, weldments, … • Life assessment of welded joints			
Level of technological readiness	8 (SFR – 40 years life duration)	**Likeliness of use**	100 %
Other observations			
List of identity card authors			
M. Blat-Yrieix (EDF R&D), C. Petesch (CEA), M. Serrano (CIEMAT), L. Kurpaska (NCBJ), K.-F. Nilsson (JRC)			

2.5 Martensitic stainless steels

Compositions of interest	X6 CrNiCu 17-04, X6 CrNiMo 16.04, X5 CrNiCuMo 15-6 and X12 Cr 13		
Type of reactor	**Large reactors**	**SMR (if applicable)**	**Notes**
Light water reactor	X		PWR 900, 1300, N4, EPR
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy- fast cooled reactor			
Gas-cooled fast reactor			
(Very) High Temperature Reactor			
Water-cooled supercritical reactor			
Molten salt reactor			
Other (specify here)			
Fusion			
Component type	valve stems, bolts and nuts / steam generator tie rods		
Method of manufacture	Stretched		
Assembly method	N/A		
Target operating environment	325°C, 155 bar		
Properties that make the material suitable for the target environment			
Stress corrosion cracking, fatigue, wear, thermal embrittlement resistance References : B. Yrieix, M. Guttmann, Aging between 300 and 450 °C of wrought martensitic 13-17 % Cr stainless steels, Materials Science and Technology, February 1993			
Open questions to consider			
- Chemical composition optimization to minimize thermal embrittlement - Minimum temperature for thermal embrittlement according to the chemical composition - Up to date modelling to simulate mechanical properties References: J.M. Boursier, D. Buisine, M. Fronteau, Y. Meyzaud, D. Michel, Y. Rouillon, B. Yrieix, in-service ageing of martensitic stainless steels, minutes of the Fontevraud 4 international seminar organized by SFEN (1998) E. Molinié, R. Tampigny, F. Foct, P. Dignocourt, In service thermal ageing of martensitic stainless steels, minutes of the Fontevraud 7 international seminar organized by SFEN (2010)			
Property improvements required or in progress			
- To avoid replacement during lifetime (60 or 80 years) - steelmakers works during last 25 years References: Synthesis of AFCEN study, SG02.097, 2002, Substitution materials for X6 CrNiCu 17-04 steel for main primary and main secondary systems valve stems of PWR power plants			
Level of technological readiness	7-8	**Likeliness of use**	100 %
Other observations			
- International feedback - Other industries feedback (oil and gas)			
List of identity card authors			
E. Molinié (EDF), B. Yrieix (EDF)			

2.6 Cast austenitic stainless steels

Compositions of interest	Ni-Cr-Mo		
Type of reactor	**Large reactors**	**SMR (if applicable)**	**Notes**
Light water reactor	X	X	
Heavy water reactor			
Sodium-cooled fast reactor	X		
Lead alloy- fast cooled reactor			
Gas-cooled fast reactor			
(Very) High Temperature Reactor			
Water-cooled supercritical reactor			
Molten salt reactor			
Other (specify here)			
Fusion			
Component type	Valve and pump casing, elbows, fittings. Internals core components		
Method of manufacture	Casting		
Assembly method	Welding (narrow gap)		
Target operating environment	Full T and P for primary circuit + irradiation (less than 1 dpa)		
Properties that make the material suitable for the target environment			
Equivalent chromium (Cr%+Si%+Mo%) < 23% 10 < ferrite (%) < 25 Low content in inclusions, without cast defects Mechanical properties (Charpy impact energy and J-R curves)/ ease of manufacturing including welding / good resistance to corrosion			
Open questions to consider			
Thermal ageing (spinodal decomposition) and impact on mechanical properties Combined effect of thermal & irradiation ageing References: [1] O.K. Chopra, A.S. Rao, Methodology for estimating thermal and neutron embrittlement of cast austenitic stainless steels during service in light water reactors, J. Pressure Vessel Techno. 138 (2016) https://doi.org/10.1115/1.4031909 , 1-24 [2] Y.Miura, T. Sawake, K.Betsuyaku, T. Arai, Thermal ageing behavior of grade CF3M cast austenitic stainless steels, in : proccedings of ASME 2017 Pressure and Vessel Piping Conference, 2017, p. 1A, https://doi.org/10.1115/PVP2017-65959 [3] S. Saillet, P. Le Delliou, Prediction of J-R curves and thermoelectric power evolution of cast austenitic stainless steels after very long-term aging (200,00 h) at temperatures below 350°C, Journal of Nuclear Materials, 540 (2020), https://dio.org10.1016/j.jnucmat.2020.152328			
Property improvements required or in progress			
Thermal regeneration: thermal ageing kinetic after regeneration – impact on mechanical properties Thermal ageing for operating temperature > 400°C (SFR reactor)			
Level of technological readiness	8	**Likelyness of use**	100%
Other observations			
International feedback on thermal regeneration and thermal ageing for operating temperature > 400°C (Energy, Oil and Gas)			
List of identity card authors			
Pal Efsing (KTH; Vattenfall), B. Tanguy (CEA), E. Molinié (EDF), M. Blat (EDF).			

2.7 Nickel-based alloys

Compositions of interest	Inconel 617, 625, 690 718, Hastelloy C276 Alloy 800, Incolloy 800 H, Nimonic Alloy 90, Alloy PE16 Alloy 230, Haynes 230, UNS N06230, Alloy 617 Incolloy 800: Limiting Chemical Composition, (% by Weight) Ni=30.0-35.0; Cr=19.0-23.0; Fe=39.5min.; C=0.10max.; Mn=1.50max.; S=0.015max.; Si=1.0max.; Cu=0.75max.; Al=0.15-0.60; Ti=0.15-0.60 Alloy 230: Limiting Chemical Composition, (% by Weight) Ni=balance; Cr=20.0-24.0; Co=5.0 max.; Fe=3.0 max.; Mo=1.0-3.0; W=13.0-15.0; C=0.05-0.15.; Mn=0.30-1.00; S=0.015 max.; Si=0.25-0.75; Al=0.20-0.50; P=0.030 max.; La=0.005-0.050; B=0.015 max		
Type of reactor	**Large reactors**	**SMR (if applicable)**	**Notes**
Light water reactor	X	X	
Heavy water reactor	X		SG tubing
Sodium-cooled fast reactor	X		
Lead alloy- fast cooled reactor			
Gas-cooled fast reactor			
(Very) High Temperature Reactor	X		
Water-cooled supercritical reactor	X		
Molten salt reactor	X		
Other (specify here)			
Fusion	X		
Component type	Steam generator turbine and tubing, reactor pressure vessel head/penetration, piping safe ends, dissimilar metal welds, fasteners and brackets, fuel alignment pins, springs in fuel elements. Intermediate heat exchangers Resistance-heated alloy 230: superheater tubes (used to produce about 885°C high-pressure steam)		
Method of manufacture	Forging, rolling, cold-drawing, welding, Extrusion for tubes, hot-working (alloy 230)		
Assembly method	Welding		
Target operating environment	Similar than stainless steels Fission reactor, CANDU, heavy water - water cooled: 265-320°C@10.5 MPa		
Properties that make the material suitable for the target environment			
High-temperature strength, ductility, toughness, corrosion resistance (including stress corrosion cracking), heat resistance, irradiation creep High corrosion resistance at high temperature Resistance to oxidation, carburization, and sulfidation and stability for service up to 1500°F (816°C) References: Pearl, W.L., Brush, E.G., Gaul, G.G. and Leistikow, S., 1967. General Corrosion of Inconel Alloy 625® in Simulated Superheat Reactor Environment. *Nuclear Applications*, *3*(7), pp.418-432. K. Siva Rama Krishna Rao, K.Praveena, Manufacturing of Incoloy- 800 Tubes Nuclear Steam Generator Tubes, International Journal of Science Engineering and Advance Technology, IJSEAT, Vol 2, Issue 9, p.426 – 431. Structural Materials for Innovative Nuclear Systems (SMINS-3), Workshop Proceedings Idaho National			

Laboratory, 7-10 October 2013, NEA/NSC/WPFC/DOC(2015)9
Open questions to consider
- Alloy Composition Search in weld materials (?) - Swelling at high doses - Gamma bis precipitates formation under irradiation and temperature - The elevated temperatures irradiations cause a loss in strength as compared with the low temperature irradiations at equivalent fast fluences. <u>General needs</u>: - Accurate multiscale modelling for long-term structural performance with respect to degradation and cracking - Active survey for additive manufacturing linked with design opportunities: reducing machining, area with stress concentration, weldments - Increase melting temperature of Inconel-like alloys through composition searches. -creep fatigue degradation (alloy 617) at high temperature for VHTR applications -effect of impure environment on crack growth for alloy 617. References: Angeliu, T., Ward, J., Witter, J., Assessing the Effects of Radiation Damage on Ni-base Alloys for the Prometheus Space Reactor System, LM-06K033, 2006.
Property improvements required or in progress
Weldability for modern alloys Multicomponent 2-phase alloys and thermodynamics for increasing melting temperature using ab-initio methods. High temperature strength References: K. Siva Rama Krishna Rao, K. Praveena, Manufacturing of Incoloy- 800 Tubes Nuclear Steam Generator Tubes, International Journal of Science Engineering and Advance Technology, IJSEAT, Vol 2, Issue 9, ISSN 2321-6905, p.426 – 431.

Level of technological readiness	8	**Likeness of use**	80%

Other observations
List of identity card authors
Stefanos Papanikolaou (NCBJ), Pal Efsing (KTH, Vattenfall), Benoit Tanguy (CEA), D. Lucan (RATEN ICN), Łukasz Kurpaska (NCBJ)

Compositions of interest	Alloy 230, Haynes 230, UNS N06230 Limiting Chemical Composition, (% by Weight) Ni=balance; Cr=20.0-24.0; Co=5.0 max.; Fe=3.0 max.; Mo=1.0-3.0; W=13.0-15.0; C=0.05-0.15.; Mn=0.30-1.00; S=0.015 max.; Si=0.25-0.75; Al=0.20-0.50; P=0.030 max.; La=0.005-0.050; B=0.015 max.

Type of reactor	**Large reactors**	**SMR (if applicable)**
Light water reactor		
Heavy water reactor		
Sodium-cooled fast reactor		
Lead alloy- fast cooled reactor		
Gas-cooled fast reactor		
(Very) High Temperature Reactor	X	
Water-cooled supercritical reactor		
Molten salt reactor		
Other		

Fusion		
Component type	Intermediary Heat Exchangers, Resistance-heated alloy 230 superheater tubes. Used to produce about 1625°F (885°C) high-pressure steam.	
Method of manufacture	Excellent forming and welding characteristics. It may be forged or otherwise hot-worked, providing it is held at 2150°F (1177°C) for a time sufficient to bring the entire piece to temperature. As a consequence of its good ductility, alloy 230 is also readily formed by cold-working. All hot- or cold-worked parts should be annealed and rapidly cooled in order to restore the best balance of properties.	
Assembly method	The alloy can be welded by a variety of techniques, including gas tungsten arc welding (GTAW), gas metal arc welding (GMAW), and resistance welding.	
Target operating environment	Alloy 230 exhibits excellent retained ductility after long-term thermal exposure at intermediate temperatures. It does not exhibit sigma phase, mu phase, or other deleterious phase formation even after 16,000 hours of exposure at temperatures from 1200 to 1600°F (649 to 871°C). Principal phases precipitated from solid solution are all carbides.	
Properties that make the material suitable for the target environment		
Best balance of strength, thermal stability, oxidation resistance, thermal cycling resistance and fabricability of any major high-temperature alloy. Excellent high-temperature strength, outstanding resistance to oxidizing environments up to 2100°F (1149°C) for prolonged exposures, premier resistance to nitriding environments, and excellent long-term thermal stability. It is readily fabricated and formed, and is castable. Other attractive features include lower thermal expansion characteristics than most high-temperature alloys, and a pronounced resistance to grain coarsening with prolonged exposure to high temperatures. References: Physical and Thermal Properties – Inconel Alloy N06230 – www.specialmetals.com. TOTEMEIER, T., C., REN, W., Procurement and Initial Characterization of Alloy 230 and CMS Alloy 617, INL/EXT-06-11290, 2006. JIAN, L., JIAN, P., BING, H., XIE, G., Oxidation Kinetics of Haynes 230 Alloy in Air at Temperatures, between 650 and 850°C, Journal of Power Sources 159, p. 641–645, 2006. DONGMEI, L., RUI, H., JINSHAN, L., YI, L., HONGCHAO, K., HENGZHI, F., Isothermal Oxidation Behavior of Haynes 230 Alloy in Air at 1100 °C, Rare Metal Materials and Engineering, 37(9), p. 1545-1548, 2008.		
Open questions to consider		
The effects of strain at higher temperatures results in almost continuous precipitation along the grain boundaries with coarse particles being also randomly present within the grains. The effects of creep strain on the microstructure were most pronounced at the highest test temperature where the grain boundary precipitation was continuous with no evidence of discrete particles being present. References: G. Marchant, G. McColvin, A. Strang, Microstructural Degradation of Haynes 230 Combustor Hardware, Proceedings of the 8th Liège Conference Materials for Advanced Power Engineering 2006. WRIGHT, R., SIMPSON, J., WERTSCHING, A., SWANK, W. D., High Temperature Behavior of Candidate VHTR Heat Exchanger Alloys, Proceedings of the 4-th International Topical Meeting on High Temperature Reactor Technology HTR, September 28-Octomber 1, 2008.		
Property improvements required or in progress		
- High temperature microstructure		

References: G. Marchant, G. McColvin, A. Strang, Microstructural Degradation of Haynes 230 Combustor Hardware, Proceedings of the 8th Liège Conference Materials for Advanced Power Engineering 2006.			
Level of technological readiness	6-7	**Likeliness of use**	
Other observations			
-			
List of identity card authors			
Dumitra Lucan (RATEN ICN)			

Compositions of interest	Alloy 617 Limiting Chemical Composition, (% by Weight) Ni=44.5 min.; Cr=20.0-24.0; Co=10.0-15.0; Fe=.3.0 max.; Mo=.8.0-10.0; C=0.05-0.15.; Mn=1.0 max.; S=0.015 max; Ti=.0.6 max.; Si=1.0 max.; Al=0.8-1.5; Cu=0.5 max.; B=0.006 max.	
Type of reactor	**Large reactors**	**SMR (if applicable)**
Light water reactor		
Heavy water reactor		
Sodium-cooled fast reactor		
Lead alloy- fast cooled reactor		
Gas-cooled fast reactor		
(Very) High Temperature Reactor	X	
Water-cooled supercritical reactor	X	
Molten salt reactor		
Other		
Fusion		
Component type	Intermediary Heat Exchangers	
Method of manufacture	Extrusion Ref: H. Jiang, J. X. Dong & M. C. Zhang, Manufacture process design of Inconel 617B superheater tubes for ultra-supercritical power plants, Materials Research Innovations, 18:sup4, (2014), S4-369-S4-374	
Assembly method	Alloy 617 has a good fabricability. Forming, machining, and welding are carried out by standard procedures for nickel alloys. Techniques and equipment for some operations may be influenced by the alloy’s strength and work-hardening rate	
Target operating environment	482-700°C@35.5 MPa	
Properties that make the material suitable for the target environment		
- a high degree of resistance to oxidation and carburization at high temperatures. Alloying elements, along with the molybdenum content, also enable the alloy to withstand many wet corrosive environments. References: Physical Constants and Thermal Properties – Inconel Alloy 617 – www.specialmetals.com. https://www.energy.gov/ne/articles/new-alloy-material-approved-use-high-temperature-nuclear-plants, 24.05.2021. TOTEMEIER, T., C., REN, W., Procurement and Initial Characterization of Alloy 230 and CMS Alloy 617, INL/EXT-06-11290, 2006.		
Open questions to consider		
- low cycle fatigue, creep crack growth, biaxial fatigue References:		

Ren, W., Swindeman, R. W., Assessment of Existing Alloy 617 Data for GEN IV Materials Handbook, ORNL/TM-2005/510, GenIV Nuclear Energy Systems, 2005. Wright, R., Simpson, J., Wertsching, A., Swank, W. D., High Temperature Behavior of Candidate VHTR Heat Exchanger Alloys, Proceedings of the 4-th International Topical Meeting on High Temperature Reactor Technology HTR, September 28-Octomber 1, 2008.
Property improvements required or in progress
- manufacture homogeneity References : H. Jiang, J. X. Dong & M. C. Zhang, Manufacture process design of Inconel 617B superheater tubes for ultra-supercritical power plants, Materials Research Innovations, 18:sup4, (2014), S4-369-S4-374.

Level of technological readiness	8	**Likeliness of use**	**80-90%**

Other observations
-
List of identity card authors
Dumitra Lucan (RATEN ICN)

Compositions of interest	Alloy 800, Incoloy 800 (UNS N08800, W. Nr. 1.4876) Nickel-iron-chromium alloy Limiting Chemical Composition, (% by Weight) Ni=30.0-35.0; Cr=19.0-23.0; Fe=39.5min.; C=0.10max.; Mn=1.50max.; S=0.015max.; Si=1.0max.; Cu=0.75max.; Al=0.15-0.60; Ti=0.15-0.60		
Type of reactor	**Large reactors**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor	X		SG tubing
Sodium-cooled fast reactor			
Lead alloy- fast cooled reactor			
Gas-cooled fast reactor			
(Very) High Temperature Reactor			
Water-cooled supercritical reactor			
Molten salt reactor			
Other			
Fusion			
Component type	Structural application in steam generators' tubing;		
Method of manufacture	Extrusion for tubes. Products: pipe, tube, fitting, flange, sheet, strip, plate, round bar, flat bar, fastener, forging stock, hexagon and wire		
Assembly method	TIG		
Target operating environment	Fission reactor, CANDU, heavy water - water cooled: 265-320°C@10.5 MPa;		

Properties that make the material suitable for the target environment
Corrosion resistance (including stress corrosion cracking), heat resistance, rupture and creep strength, resistance to oxidation, carburization, and sulfidation and stability for service up to 1500°F (816°C). References : K. Siva Rama Krishna Rao, K.Praveena, Manufacturing of Incoloy- 800 Tubes Nuclear Steam Generator Tubes, International Journal of Science Engineering and Advance Technology, IJSEAT, Vol 2, Issue 9, ISSN 2321-6905, p.426 – 431.
Open questions to consider
- The elevated temperatures irradiations cause a loss in strength as compared with the low

temperature irradiations at equivalent fast fluence References: Angeliu, T., Ward, J., Witter, J., Assessing the Effects of Radiation Damage on Ni-base Alloys for the Prometheus Space Reactor System, LM-06K033, 2006.			
Property improvements required or in progress			
- High temperature strength References : K. Siva Rama Krishna Rao, K. Praveena, Manufacturing of Incoloy- 800 Tubes Nuclear Steam Generator Tubes, International Journal of Science Engineering and Advance Technology, IJSEAT, Vol 2, Issue 9, ISSN 2321-6905, p.426 – 431.			
Level of technological readiness	8-9	**Likeliness of use**	90%
Other observations			
-			
List of identity card authors			
D. Lucan (RATEN ICN)			

2.8 Aluminium alloys

Compositions of interest	2XXX, 5XXX, 6XXX and 8xxx series		
Type of reactor	**Large reactors**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy- fast cooled reactor			
Gas-cooled fast reactor			
(Very) High Temperature Reactor			
Water-cooled supercritical reactor			
Molten salt reactor			
Other (specify here)	X		Research reactors
Fusion			
Component type	Pressure vessels, low pressure tanks, connecting pipes, internal core structures, beam-port nozzles, core support structures and poolside irradiation facilities fuel cladding		
Method of manufacture	forging, rolling		
Assembly method	Welding electron beam, MIG, Friction Stir welding		
Target operating environment	-Up to a conventional thermal fluence of 10^{23}n/cm^2 (thermal neutrons) at temperature < 100°C for the core structure -Up to a conventional thermal fluence of 10^{22}n/cm^2 at temperature ranging from 100°C to 200°C for the core structure		

Properties that make the material suitable for the target environment

- Very high thermal neutron transparency (close to Zirconium)
- water corrosion resistance
- mechanical properties (<75-100°C)
- formability and relatively easy welding
- high tolerance to radiation effects when irradiated at ambient temperatures (<75-100°C)
- irradiation swelling resistance

References:

K. Farrell, "Performance of Aluminum in Research Reactors", comprehensive nuclear materials, Elsevier, chapter 5.07, pp 143-177, 2012

D. J. Alexander, "The Effects of Irradiation on the Mechanical Properties of 606 1 -T651 Aluminium Base Metal and Weldments", pp. 1027- 1044 in Effects of Radiation on Materials: 18th International Symposium, ASTMSTP 1325, Amer. Sot. for Testing and , Mater., Amer. Sot. for Testing and Mater., 1999

Open questions to consider

Limited availability of data on mechanical properties exposed to high neutron fluence (fracture toughness) and swelling. Most studies have been done with a thermal/fast flux ratio close to 2 (HFIR) and 20 (few data from HFBR). No data are available with a harder spectrum (swelling, corrosion, mechanical properties).

Only few data on mechanical properties of welds after irradiation are available.

To improve fatigue and fretting wear resistance, surface treatments as anodizing could be done. Today, no coating has been qualified under irradiation.

No data have been published on irradiation creep behaviour.

Irradiation of model alloys (with simplified composition and other tempering T4, T6, T7 for

heat treatable series) and microstructural characterisations (TEM, TAP…) would improve the understanding of physical mechanisms (stability of the pre-existing beta'' and nature of the new phase induced by the transmutation). References: M. Kolluri, Neutron Irradiation Effects in 5xxx and 6xxx Series Aluminum Alloys: A Literature Review, Radiation Effects in Materials, Waldemar A. Monteiro, IntechOpen, DOI: 10.5772/63294. Available from: https://www.intechopen.com/chapters/50632			
Property improvements required or in progress			
Validation of fracture mechanics testing of sub-size CT specimens for surveillance programs			
Level of technological readiness	9	**Likeliness of use**	90%
Other observations			
Corrosion of Aluminium alloys			
List of identity card authors			
N.V.V.R.M. Kolluri (NRG), J. Garnier (CEA)			

2.9 High entropy alloys

Compositions of interest	Fe, Ni, Cr, Mn + Al, Nb, Ti, Mo, Cu		
Type of reactor	**Large reactors**	**SMR (if applicable)**	**Notes**
Light water reactor			High entropy or compositionally complex alloys (HEA, CCA) can be tailored to be used in most reactor concepts.
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy- fast cooled reactor	X	X	
Gas-cooled fast reactor			
(Very) High Temperature Reactor			
Water-cooled supercritical reactor			
Molten salt reactor	X	X	
Other (specify here)			
Fusion	X	X	
Component type	Structural components working at high-temperatures, Wear-resistant materials, Diffusion barriers, Reactor core components		
Method of manufacture	Vacuum Arc Melting, Casting, Coating, 3D printing		
Assembly method			
Target operating environment	Contact with molten salts, or (depending on composition) molten Pb		
Properties that make the material suitable for the target environment			
The CrFeMnNi famiIty with additions of other elements (Al, Nb, Ti, Mo, Cu) have shown a combination of good properties: - High microstrutural stability - Irradiation resistance (also resistance to phase transformation and element separation) comparable or superior to austenitic stainless steels - Very low volume swelling. - High hardness and yield strength, - Strong resistance to high-temperature softening, - Creep strength, - Excellent oxidation resistance. References: S. Shen, et al. Journal of Nuclear Materials 540 (2020) 152380, W. Mutfag, et al. Vacuum 188 (2021) 110181 N.A.P Kiran Kumar, et al. Acta Materialia 113 (2016) 230-244, M. A. Tunes, et al. Thin Solid Films 649 (2018) 115–120			
Open questions to consider			
- Understand radiation damage resistance at high temperature - Volume swelling has been seen dependent on composition. The effect of Mn on magnetic properties seems to play an important role and further investigations are required. - Microstructural stability at long exposure times has not been properly addressed yet. Heat treatments at operational temperatures for 5000 -10000 h are required. - As the current TRL is low, processability, weldability and other aspects related to production and scalability can be still optimized. References: Pickering, E.; Carruther, A.; Barron, P.; Middleburgh, S.; Armstrong, D.; Gandy, A. High-Entropy Alloys for Advanced Nuclear Applications. Entropy 2021, 23, 98. https://doi.org/10.3390/e23010098			
Property improvements required or in progress			

- Irradiation embrittlement occurs in CrFeMnNi, an important decay on ductility has been observed.C. Li *et al*., Journal of Nuclear Materials 527 (2019) 151838 - Corrosion is poorly explored in this system. While it is expected that Cr offers some protection, it may be insufficient and alumina scale protection needs to be explored. CrFeMnNiAl system, with other additional elements has been initially explored with promising results H. Shi, Corrosion Science 170 (2020), 108654 - The development of Complex Concentrated Alloys (CCAs) with secondary phase reinforcement and/or martensitic HEAs (both with alumina layer) are among the most promising solutions for nuclear materials			
Level of technological readiness	2-3	**Likeliness of use**	100%
Other observations			
List of identity card authors			
Alfons Weisenburger (KIT), I.Toda-Caraballo (CENIM-CSIC), Kurpaska Lukasz (NCBJ), Stefanos Papanikolaou (NCBJ)			

2.10 Ferritic / martensitic steels (base metals and welds)

Compositions of interest	T91 (X10CrMoVNb9-1), EUROFER (Mo, Nb, and Ni is replaced by W, V, Mn, Ta)		
Type of reactor	**Large reactors**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor	X	x	
Lead alloy- fast cooled reactor	X	x	Cannot be used in LBE due to LME in the target operations temperature. Potentially applicable to pure Pb after verification LME is not an issue.
Gas-cooled fast reactor			
(Very) High Temperature Reactor			
Water-cooled supercritical reactor			
Molten salt reactor			
Other			
Fusion	X		Blanket and Diverter cassette structures
Component type	Heat exchanger, steam generator, Fusion (DEMO) : in-vessel components		
Method of manufacture	Forged, rolling Vacuum arc or induction melting, vacuum arc re-melting, Products: plates, bars, tubes		
Assembly method	Fusion welding (TIG, EB, Laser). HIP (DEMO). Additive manufacturing		
Target operating environment	350°C to 550°C Fusion reactor, helium coolant: 300-550°C@8 MPa; Fusion reactor, water coolant: 285-328°C@15.5 MPa;		

Properties that make the material suitable for the target environment

- High resistance to swelling
- Good heat conductivity, low thermal expansion coefficient, good thermal fatigue resistance
- Low induced radioactivity for RAFM variants

References:

[1] C. Cabet, F. Dalle, E. Gaganidze, J. Henry, H. Tanigawa, Ferritic-martensitic steels for fission and fusion applications, J. Nucl. Mater. 523 (2019) 510-537
[2] A.-A.F. Tavassoli and al., Current status and recent research achievements in ferritic/martensitic steels, J. Nucl. Mater. 455 (2014) 269-276

Open questions to consider

- The influence of transmutation formed helium and hydrogen on mechanical (evolution of DBTT)

and thermo-physical properties - Loss of uniform elongation after low temperature neutron irradiation - The use of SSTT for data generation in the irradiated state-Creep-fatigue interactions - Liquid metal corrosion and liquid metal embrittlement (steam oxidation is also a key issue), coating - Large welds, welding and heat treatment are critical - Fracture toughness at high temperature - Codification in RCC-MRx References: E. Gaganidze, J. Aktaa, Assessment of neutron irradiation effects on RAFM steels, Fus. Eng. Des. 88 (2013) 118- 128, http://dx.doi.org/10.1016/j.fusengdes.2012.11.020 H. Tanigawa, E. Gaganidze, T. Hirose, M. Ando, S.J. Zinkle, R. Lindau and E. Diegele, Development of benchmark reduced activation ferritic/martensitic steels for fusion energy applications, Nucl. Fusion 57 (2017) 092004 (13pp), https://doi.org/10.1088/1741-4326/57/9/092004 C. Cabet, F. Dalle, E. Gaganidze, J. Henry, H. Tanigawa, Ferritic-martensitic steels for fission and fusion applications, J. Nucl. Mater. 523 (2019) 510-537			
Property improvements required or in progress			
- Creep resistance can be improved by thermo-mechanical properties and/or oxide dispersion. - HIP and other non-fusion joining techniques are under study - Chemical composition optimization can improve low temperature embrittlement (DBTT) and high temperature strength References: [1] G. Pintsuk, E. Diegele, S. L. Dudarev, M. Gorley, J. Henry, J. Reiser, M. Rieth, European materials development: Results and perspective, Fus. Eng. Des. 146 (2019) 1300-1307, https://doi.org/10.1016/j.fusengdes.2019.02.063			
Level of technological readiness	T91 (9), EUROFER97 (7) For Fusion : 5	**Likeliness of use**	
Other observations			
The use of F/M steel is not considered in Liquid lead bismuth due to liquid metal embrittlement issues. LME in molten lead needs assessment. In Fusion reactor, EUROFER97 is on the qualification path, LME by PbLi needs assessment.			
List of identity card authors			
M. Serrano (CIEMAT), E. Gaganidze (KIT), J. Aktaa (KIT), M. Angiolini (ENEA)			

2.11 Alumina Forming Austenitic Steels

Compositions of interest	Ni AFA steels, AFA Superalloys, Mn AFA steels Fe-(12-35) Ni - (12-20) Cr - (2-6) Al - (0.1-3) Nb +Alloying (Mo,Si,C).		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor	x	x	
Sodium-cooled fast reactor			
Lead alloy-cooled fast reactor	x	x	
Gas-cooled fast reactor			
(Very) high temperature reactor			
Supercritical water-cooled reactor	x	x	
Molten salt reactor	X	X	
Other (specify here)			
Fusion			Al activation
Type of component	Structures subjected to HLM/water/Supercritical water corrosion, Heat exchanger tubes, core components Fuel Cladding		
Manufacturing method	Standard cast, weld overlay, coating		
Joining method	Welding (under development)		
Target operational environment	High temperature (550°C and higher), long term contact with fluid corrosive médium (Contact with heavy liquid metal up to 650°C)		
Properties that make the material suitable for the target environment			
- The alumina forming austenitic steels are a new class of high-temperature alloys that promise both superior corrosion and creep resistance while keeping low cost, formability, and the weldability of conventional high-temperature austenitic steels. - AFA steels demonstrated good corrosion resistance in contact with heavy liquid metals due to in-situ of thin protective alumina scales References: Muralidharan, Govindarajan & Yamamoto, Y. & Brady, M.P. & Pint, Bruce & Voke, D. & Pankiw, " Development of cast alumina-forming Austenitic stainless steel alloys for use in high temperature process environments" NACE - International Corrosion Conference Series. 2015; M. P. Brady doi.org/10.1007/s11837-008-0083-2; P. Szakálos doi.org/10.1016/j.jnucmat.2015.03.011			
Open issues to be investigated			
- Stability of the austenite phase - Alloy additions that improve the ability to form a protective scale reducing the Al content - Relationship between composition design and precipitation behavior of different phases. -Development of AFA alloys with no or very low content of carbon. - Role of some phases (e.g. Laves) on creep strength. - Stability of austenite in liquid metal. - Increase of the internal oxidation transition temperature. - Radiation resistance. References: Yamamoto et al. https://doi.org/10.1007/s11661-010-0295-2, H.Shi et al. Corrosion Science (2021), 10915 P. Dömstedt et al. J. Nucl. Mat. 531 (2020) 152022			
Property improvements required or in progress			
- Alloy design to improve the stability of the austenite phase - Improve the ability to form a protective scale			

- Long term phase stability - Decrease the costs decreasing the Ni content, alternative austenite stabilizers References: "The Development of Alumina Forming Austenitic Alloy for Core Application in Advanced Nuclear Reactors", Z. Zhou et al., Materials Science Forum, ISSN: 1662-9752, Vol. 999, pp 72-80, 2020.			
Technology readiness level	TRL3- 4	**Likeliness of use**	
Other observations			
List of authors of the ID card			
Massimo Angiolini (ENEA), Alfons Weisenburger (KIT)			

2.12 FeCrAl alloys

Compositions of interest	Fe, Cr, Al, + Y, Hf, Zr,Nb,Ti		
Type of reactor	**Large reactors**	**SMR (if applicable)**	**Notes**
Light water reactor	X	X	
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy- fast cooled reactor	X	X	Fe10Cr4Al
Gas-cooled fast reactor			
(Very) High Temperature Reactor			
Water-cooled supercritical reactor			
Molten salt reactor			
Other			
Fusion	X		Al activation
Component type	Plates (e.g. grid assemblies), bars (e.g. supportive rods), heat exchanger, core components Structural components for advanced reactor designs, First wall and blanket structures for fusion reactors.		
Method of manufacture	Vacuum melt process + Hot-work or cold-work combined with re-heating process. Casting, coating, weld overlay. Powder metallurgy to avoid macroscopic segregations.		
Assembly method	Welding		
Target operating environment	Contact with heavy liquid metal up to 650°C (up to 850°C in Lead alloy cooled reactor for some FeCrAl alloys (Fe10Cr4Al))		
Properties that make the material suitable for the target environment			
Excellent oxidation and corrosion resistance in steam and air environment (In-situ formation of thin protective alumina scales), Ductile at all temperatures. I			
Open questions to consider			
High temperature mechanical properties Creep and fatigue properties as a function of composition. Fracture toughness of neutron irradiated FeCrAl. Environmental effects during or after irradiation of FeCrAl (Corrosion performance in HLM need extensive testing and assessment). Irradiation creep data at high doses Change of deformation mode from low dose to moderate/high dose regime Fretting and wear characteristics of lower Cr (<20 wt.% Cr) FeCrAl alloys. Role of Al content on the dislocation loop nature and density. Role of minor alloying elements on the dislocation loop formation and growth. Effect of C on the embrittlement. Precipitation of α’ at high doses. Weldability as a function of Cr and Al content. Weldable, but reduction in ductility due to grain growth might be a problem. Laser welding with rapid cooling gives ductile welds (Development of a welding process to overlay FeCrAl alloy on a thin wall austenitic stainless steel tube References: Yang, Sheng et al. , Nuclear materials and energy, 2021-06, Vol.27, p.100958)			

Handbook on the material properties of FeCrAl alloys for nuclear power production applications (FY18 version: Revision 1.1) ORNL/SPR-2018/905 Rev.1 Mechanical properties of neutron-irradiated model and commercial FeCrAl alloys Field, K and al., Journal of nuclear materials, 2017-06, Vol.489 , p.118-128.			
Property improvements required or in progress			
• Fine adjustment of the chemical composition and thermoplastic treatments to eliminate the formation of transient alumina at temperatures <600°C. • New welding technology to avoid Al depletion in the welding area. • Produce martensitic or ferritic-martensitic microstructures with appropriate compositional changes and thermal or thermo-mechanical treatments.			
Level of technological readiness	TRL4-5	**Likeliness of use**	
Other observations			
For Fe10Cr4Al alloys: Excellent oxidation properties up to 800°C: Dömstedt, Peter; Lundberg, Mats; Szakálos, Peter Journal of nuclear materials, 2020-04-01, Vol.531, p.152022. Structure stability and good ductility at all temperatures. Liquid metal embrittlement resistant in pure Pb but not in PbBi. Irradiation stability up to at least 14 dpa.			
List of identity card authors			
M. Angiolini (ENEA), A. Weisenburger (KIT), P. Szakalos (KTH)			

2.13 Hard facing materials

Compositions of interest	Cobalt based alloys (stellite), Cr-based, Ni-based		
Type of reactor	**Large reactors**	**SMR (if applicable)**	
Light water reactor	X	X	
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy- fast cooled reactor			
Gas-cooled fast reactor			
(Very) High Temperature Reactor			
Water-cooled supercritical reactor			
Molten salt reactor			
Other			
Fusion			
Component type	Alignment pins, valve seats, radial key, support structures		
Method of manufacture	In Work Shop: Welding (Manual/Mechanized), Powder Metallurgy PM-HIP & AM(HIP)		
Assembly method	In Situ requirements: Welding (Manual/Mechanized)		
Target operating environment	Up to 320°C (BWR or PWR environment)		
Properties that make the material suitable for the target environment			
Low friction, high surface hardness, corrosion resistance, galling/adhesive wear/fretting			
Open questions to consider			
Activity build up with Co-and Ni-based alloy (radioactive isotopes generating gamma radiation decomposition)			
Property improvements required or in progress			
Decrease the use of Co- and Ni- based alloy, decrease the sticking. Weldability, Solid solution of C and N for Martensite formation & PRE for corrosion resistance			
Level of technological readiness	8	**Likeliness of use**	80%
Other observations			
- High compressive strength, adequate toughness/ductility, machinability, grindability polishability…			
List of identity card authors			
Pal Efsing (KTH, Vattenfall), Björn Forssgren (Vattenfall), Benoit Tanguy (CEA)			

2.14 Duplex austenitic / ferritic steels

Compositions of interest	Cr-Ni-Mo	
Type of reactor	**Large reactors**	**SMR (if applicable)**
Light water reactor	X	X
Heavy water reactor		
Sodium-cooled fast reactor		
Lead alloy- fast cooled reactor		
Gas-cooled fast reactor		
(Very) High Temperature Reactor		
Water-cooled supercritical reactor		
Molten salt reactor		
Other		
Fusion		
Component type	Secondary components	
Method of manufacture	Casting / additive manufacturing	
Assembly method	Welding	
Target operating environment	Lower than 250°C (specifically in oxidizing conditions (PWR or BWR)	

Properties that make the material suitable for the target environment			
Better pitting resistance and mechanical properties than 304/316.			
Open questions to consider			
Spinodal decomposition at high temperature Upper temperature for use: discrepancy between codes (from 250 to 315°C)			
Property improvements required or in progress			
Decrease the spinodal decomposition by allowing to increase the temperature for use in the primary circuit			
Level of technological readiness	8	**Likeliness of use**	100%
Other observations			
-			
List of identity card authors			
P. Efsing (KTH, Vattenfall), B. Tanguy (CEA)			

3. Fuel cladding materials

Coordinator: Massimo Angiolini, ENEA

3.1 Chromium coated Zirconium alloys

Compositions of interest	Cr based coatings on Zirconium alloy	
Type of reactor	**Large reactor**	**SMR (if applicable)**
Light water reactor	X	X
Heavy water reactor		
Sodium-cooled fast reactor		
Lead alloy-cooled fast reactor		
Gas-cooled fast reactor		
(Very) high temperature reactor		
Supercritical water-cooled reactor		
Molten salt reactor		
Other (specify here)		
Fusion		
Type of component	Fuel cladding	
Manufacturing method	CVD, PVD	
Joining method	N.A.	
Target operational environment	LWR water, high temperature steam	

Properties that make the material suitable for the target environment

- Deposition of protective coatings on Zircaloy cladding tubes, which improve corrosion resistance and decrease hydrogen uptake, is considered as a near-term solution of enhanced ATF cladding
- Delay strong oxidation of the Zircaloy after hydrogen release during the design basis accidents (DBA) and beyond design basis accidents (BDBA)

References: Jean-Christophe Brachet et al. https://doi.org/10.1016/j.jnucmat.2019.02.018

Open issues to be investigated¡Error! Marcador no definido.

- Behaviour under neutron irradiation
- Estimation of coating survival time
- Formation of bubbles / blisters / voids between the Cr coating and the Zr substrate
- Ballooning
- Impact of eutectics (1559°C for Cr-rich compositions and 1316°C Zr-rich) on the maximum temperature attainable under irradiation
- Impact of the differences in the thermomechanical properties
- Residual stresses at the interfaces in the given geometry

References:

Jianqiao Yang et al., https://doi.org/10.1016/j.jallcom.2021.162450

ORNL/SPR-2021/6 "Development of Standardized Property Requirements, Measurement Methods, and Reporting Guidance for Coatings 01/15/2021

Property improvements required or in progress

- Fretting performance and wear resistance
- Coating of the end plug joints with cladding tubes

Technology readiness level	8-9	**Likeliness of use**	100%

Other observations

Tests in reactor by industrials in progress

List of authors of the ID card

Mirco Grosse, KIT, Jean Christophe Brachet, CEA, Massimo Angiolini, ENEA

3.2 ODS steels

Compositions of interest	9-18% Cr, FeCrAl, Austenitic ODS		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor	X	X	9-18%Cr
Lead alloy-cooled fast reactor	X	X	Only if coated to protect against LME
Gas-cooled fast reactor			
(Very) high temperature reactor			
Supercritical water-cooled reactor		X	310L ODS
Molten salt reactor			
Other (specify here)			
Fusion	X		Low activation version
Type of component	Fuel cladding, internals, specific parts		
Manufacturing method	Mechanical alloying followed by different sintering process (HIP, SPS,..), Cold Spray, internal oxidation		
Joining method			
Target operational environment	Sodium cooled fast reactors		
Properties that make the material suitable for the target environment			
- Resistance to neutron irradiation at dose higher than 150 dpa - Good corrosion resistance in sodium environment Reference: Yann de Carlan *et al.*, J. Nucl. Mater. 428, 6 (2012) https://doi.org/10.1016/j.jnucmat.2011.10.037			
Open issues to be investigated			
- Reproducibility, quality assurance, anisotropy (tubes) - Welding routes			
Property improvements required or in progress			
- Optimize the fabrication route - Non-fusion weld techniques (e.g. Stir friction)			
Technology readiness level	4-5	**Likeliness of use**	20%
Other observations			
List of authors of the ID card			
M. Angiolini, M. Serrano			

3.3 Coated AIM1 steels

Compositions of interest	15-15 Ti (AIM-1) + Alumina coating		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor	X	X	
Lead alloy-cooled fast reactor	X	X	
Gas-cooled fast reactor			
(Very) high temperature reactor			
Supercritical water-cooled reactor			
Molten salt reactor			
Other (specify here)			
Fusion			
Type of component	Fuel Cladding		
Manufacturing method	Seamless Tubing Process + deposition process for ceramic coatings		
Joining method	TIG welding, electron-beam welding, PLD, ALD, Detonation Gun, CVD (Pack Cementation)		
Target operational environment	Pure lead up to 650°C		

Properties that make the material suitable for the target environment

- Large resistance to creep, irradiation swelling, irradiation creep of 15-15 Ti
- Coating compatibility with lead-coolant environment up to 650°C in a wide range of oxygen concentration
- Good resistance to ion irradiation of Ceramic coatings.

References

- P. Lorusso, S. Bassini, A. Del Nevo, I. Di Piazza, F. Giannetti, M. Tarantino, M. Utili, "Gen-IV LFR Development; Status & Perspectives" Progress in Nuclear Energy, 105 (2018), https://doi.org/10.1016/j.pnucene.2018.02.005
- Tarantino, M.; Angiolini, M.; Bassini, S.; Cataldo, S.; Ciantelli, C.; Cristalli, C.; Del Nevo, A.; Di Piazza, I.; Diamanti, D.; Eboli, M.; Fiore, A.; Grasso, G.; Lodi, F.; Lorusso, P.; Marinari, R.; Martelli, D.; Papa, F.; Sartorio, C.; Utili, M.; Venturini, A.; "Overview on Lead-Cooled Fast Reactors Design and Related Technologies Development in ENEA" Energies 2021, 14, 5157, https://doi.org/10.3390/en14165157
- A. Alemberti, M. Caramello, M. Frignani, G. Grasso, F. Merli, G. Morresi, M. Tarantino, "ALFRED reactor coolant system design" Nuclear Engineering and Design 370 (2020), https://doi.org/10.1016/j.nucengdes.2020.110884
- F. García Ferré, A. Mairov, D. Iadicicco, M. Vanazzi, S. Bassini, M. Utili, M. Tarantino, M. Bragaglia, F.R. Lamastra, F. Nanni, L. Ceseracciu, Y. Serruys, P. Trocellier, L. Beck, K. Sridharan, M.G. Beghi, F. Di Fonzo, "Corrosion and radiation resistant nanoceramic coatings for lead fast reactors" Corrosion Science 124 (2017), https://doi.org/10.1016/j.corsci.2017.05.011
- S. Bassini, S. Cataldo, C. Cristalli, A. Fiore, C. Sartorio, M. Tarantino, M. Utili, P. Ferroni, M. Ickes, A. Alemberti, M. Frignani, "Material Performance in Lead and Lead-Bismuth Alloy", in: Konings, Rudy JM and Stoller Roger E (eds.). Comprehensive Nuclear Materials 2nd edition, vol. 4, pp. 218-241, 2020, http://dx.doi.org/10.1016/B978-0-12-803581-8.00749-9

Open issues to be investigated

- Neutron compatibility of ceramic coatings up to 100 dpa
- Resistance to lead corrosion up to 800°C

References

- P. Lorusso, S. Bassini, A. Del Nevo, I. Di Piazza, F. Giannetti, M. Tarantino, M. Utili, "Gen-IV LFR Development; Status & Perspectives" Progress in Nuclear Energy, 105 (2018), https://doi.org/10.1016/j.pnucene.2018.02.005
- Tarantino, M.; Angiolini, M.; Bassini, S.; Cataldo, S.; Ciantelli, C.; Cristalli, C.; Del Nevo, A.; Di Piazza, I.; Diamanti, D.; Eboli, M.; Fiore, A.; Grasso, G.; Lodi, F.; Lorusso, P.; Marinari, R.; Martelli, D.; Papa, F.; Sartorio, C.; Utili, M.;

Venturini, A.; "Overview on Lead-Cooled Fast Reactors Design and Related Technologies Development in ENEA" Energies 2021, 14, 5157, https://doi.org/10.3390/en14165157			
Property improvements required or in progress			
- Lead compatibility up to 650°C with mechanical loads (pressurized tubes) - Neutron compatibility up to 5-10 dpa			
Technology readiness level	6	**Likeliness of use**	85%
Other observations			
- Reference solution for ALFRED DEMO-LFR - Under investigation for MYRRHA ADS - Under investigation for WEC-LFR (UK)			
List of authors of the ID card			
Mariano Tarantino, Massimo Angiolini, Fabio Di Fonzo			

3.4 Swelling resistant austenitic steels beyond AIM1

Compositions of interest	FeCrNi stabilized and Si, P optimized alloys		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor	X	X	
Lead alloy-cooled fast reactor	X	X	Coated version
Gas-cooled fast reactor	X	X	1st core @ low temperature
(Very) high temperature reactor			
Supercritical water-cooled reactor			
Molten salt reactor			
Other (specify here)			
Fusion			
Type of component	Fuel Cladding		
Manufacturing method			
Joining method			
Target operational environment	Neutron irradiation at dose higher than 100 dpa		
Properties that make the material suitable for the target environment			
- All the desirable properties of austenitic alloys, weldability, mechanical properties vs. temperature, behaviour under irradiation - Medium term option burn-up improvement - Target voids swelling below 3% above 110 dpa irradiation dose References: J.L. Seran, M. Le Flem (Ed.) Structural Materials for Generation IV Nuclear Reactors, https://www.doi.org/10.1016/B978-0-08-100906-2.00008-2			
Open issues to be investigated			
- - Gain better understanding on the mechanisms leading to loss of ductility after n irradiation - - Cr, Ni, P, Si, carbide former optimization - - stabilize the dislocation microstructure under irradiation - - Increase in sinks amount and efficiency References: M. Le Flem *et al.,* https://www.doi.org/10.1016/j.nucengdes.2015.09.015 Y. de Carlan *et al*. “Development of advanced austenitic steels study of the influence of titanium and niobium on swelling by ion irradiation” https://hal.archives-ouvertes.fr/hal-02418147			
Property improvements required or in progress			
Ductility after neutron irradiation			
Technology readiness level		**Likeliness of use**	
Other observations			
List of authors of the ID card			
M. Angiolini (ENEA), L. Pilloni (ENEA)			

3.5 SiC_f-SiC composites

Compositions of interest	Nuclear grade SiC: highly pure and stoichiometric Si/C		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor	X	X	Accident tolerant candidate
Heavy water reactor			
Sodium-cooled fast reactor	X		Channel box
Lead alloy-cooled fast reactor			
Gas-cooled fast reactor	X	X	
(Very) high temperature reactor	X	X	
Supercritical water-cooled reactor			
Molten salt reactor	X	X	
Other (specify here)			
Fusion	X		
Type of component	Fuel Cladding & core components		
Manufacturing method	Chemical Vapour infiltration, NITE		
Joining method	Brazing & assembly		
Target operational environment	LWR, high temperature steam, High temperature He, Na or molten salt environment		

Properties that make the material suitable for the target environment

- High-temperature mechanical properties, excellent irradiation resistance, inherent low activation and other superior physical/chemical properties
- For LWR, excellent passive safety features both in design basis and beyond design basis severe accidents due to an excellent high-temperature strength and an outstanding oxidation resistance to a high-temperature steam
- Reduced neutron absorption cross-section enabling a smaller uranium enrichment and that could also result in increasing the fuel cycle duration
- Exceptional inherent radiation resistance and a lack of progressive irradiation growth.
- Higher critical heat flux than for conventional Zr-based alloys,
- High stiffness and competitive fatigue behaviour,
- Mechanical properties almost time independent, UO_2 compatibility
- For design basis accident and beyond, outstanding oxidation resistance to high-temperature steam
- Significantly reduced hydrogen generation and improved structural integrity under severe accident conditions.
- Maintain a coolable geometry after quenching

References:

State of the Art Report on Light Water Reactor Accident-Tolerant Fuels, OECD, Nuclear Science 2018

K. Terrani, Accident tolerant fuel cladding development: Promise, status, and challenges, JNM 501 (2018) 13-30.

Open issues to be investigated[¡Error! Marcador no definido.]

- Structural design concept of SiC/SiC composite claddings needs to be further optimized
- Technology for end-plug joining with gas tightness and adequate strength should be developed because the SiC ceramics cannot be welded,
- New devoted industrial network needed for the production of full-length tubes
- Pre-normative research to provide the scientific bases (experimental & modelling) to establish new design rules and reliable mechanical test methods for SiC/SiC composites

for standardization - In-pile testing			
Property improvements required or in progress			
For LWR - Chemical compatibility of SiC with water coolant at about 300°C that leads to a recession - Hydrothermal corrosion of irradiated SiC and effect of radiolysis and mitigation solutions: coating, specific surface treatment or modification of water chemistry - Assessment of low pseudo-ductility (potential Pellet-cladding Interaction issue), establishment of statistical failure properties of SiC/SiC composite claddings, defining design allowable stresses under a probabilistic design approach. - Relatively poor thermal conductivity under neutron irradiation in the LWRs normal operating range which could potentially cause significant mechanical stresses leading to early multi-cracking. - High temperature side of SiC/SiC fuel cladding is subjected to high tensile stress due to differential swelling of irradiated SiC. Irradiation-induced thermal conductivity decrease in SiC could worsen the situation - Irradiation-induced swelling and low thermal conductivity of SiC composite claddings under irradiation would increase the pellet-cladding gap and cause a large temperature drop across the cladding and an increase in fuel temperature. Mitigation solutions: alternative fuels with higher thermal conductivity and modified fuel geometry such as annular pellets - Leak-tightness properties, which depend on the range of acceptable deformation and on the cladding design. A fully-ceramic SiC cladding design cannot prevent the multi-cracking of the matrix beyond the elastic limit to the composite (i.e. at low loading). Mitigation strategy: metal/ceramic clad concept, which would withstand any strain imposed by the deformation of the composite thanks to fair ductility of the metal so that the leak-tightness is guaranteed until the ultimate failure of the composite occurs References L. Snead, Y. Katoh, T. Nozawa, Radiation effects in SiC and SiC/SiC, revised in 2019, https://doi.org/10.1016/B978-0-08-056033-5.00093-8 Y. Katoh, Ceramic matrix composites in fission and fusion energy applications, in advances in Ceramic Matrix composites. https://doi.org/10.1016/B978-0-08-102166-8.00024-4			
Technology readiness level	Between 4 and 5 (LWR)	**Likeliness of use**	Breakthrough solution -> long time to deployment
Other observations			
-			
List of authors of the ID card			
Christophe Lorrette, CEA			

4. Nuclear fuel materials

Coordinators: Marjorie Bertolus, CEA, Marco Cologna, JRC

4.1 UO_2 fuel for light water reactors

Compositions of interest	^{235}U content between ~2 and 5%, O/M ratio between ~1.98 and 2.01		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor	X		PWR, BWR, WWER …
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy-cooled fast reactor			
Gas-cooled fast reactor			
(Very) high temperature reactor			
Supercritical water-cooled reactor			
Molten salt reactor			
Other (specify here)			
Fusion[¡Error! Marcador no definido.]			
Type of component	Fuel		
Manufacturing method	Ceramic powder technology		
Joining method	N/A		
Target operational environment	Cladding temperature (outer surface) ~260-330°C – Typical operating central temperature inside the fuel ~1000°C (temperature at 30-40 % of the melting point) – Average fuel rod Burn-up targeted ~60-75 GWd/t – Middle (example BWR: around ~70-80 bar) or high (example PWR: ~123-160 bar) pressure on outer clad and middle pressure in fuel rods (initial pressure is between ~20 and 35 bar for more recent PWR fuel fabrications and ~2–8 bar for BWR fuels – this pressure increases with the irradiation level) – Flux of neutrons with high energy (around 1.10^{14} $n/cm^{-2}/s$ in the core centre)		
Properties that make the material suitable for the target environment			
- High melting point providing large operating temperature margins - No allotropic changes, excellent stability - Good behaviour under irradiation in normal and transient conditions - Good compatibility with fuel rod cladding - Good compatibility with water coolant in case of fuel rod failure References: D. Baron, L. Hallstadius, K. Kulacsy, R. Largenton, J. Noirot, Comprehensive Nuclear Materials Second Edition Chap. 2.02 Fuel Performance of Light Water Reactors (Uranium Oxide and MOX), Elsevier, 2020			
Open issues to be investigated			
- Margin to fuel melting: Some progresses are still needed in the characterizations and understanding of thermal properties evolutions under irradiation and at high temperature (thermal conductivity, heat capacity, melting temperature…) - Behaviour of UO_2 fuel in steady state conditions: Some progresses are still needed in the			

characterizations and understanding of microstructure evolution: high burnup structure (HBS) and restructuring occurring in the centre of high burnup UO_2 fuel - Behaviour of UO_2 fuel under transient conditions (Ramp, RIA, LOCA …): Some progresses are still needed in the characterizations and understanding of mechanical properties evolution under irradiation and especially at high temperature: elastic properties and thermal expansion, thermal creep, local (within grain and on the grain boundaries) rupture properties - Some progress is still needed on the modelling of UO_2 fuel under transient conditions (RIA, LOCA …): damage of grain boundaries, fission gas released, Fuel Fragmentation Relocation and Dispersal…. References: - D. Baron, L. Hallstadius, K. Kulacsy, R. Largenton, J. Noirot, Comprehensive Nuclear Materials Second Edition Chap. 2.02 Fuel Performance of Light Water Reactors (Uranium Oxide and MOX), Elsevier, 2020, ISBN- 978-0-08-102866-7 - J. Carbajo, G. Yoder, S. Popov, V. Ivanov, A review of the thermophysical properties of MOX and UO_2 fuels, Journal of Nuclear Materials 299 (2001) 181-198			
Property improvements required or in progress			
- Study new Enhanced Accident Tolerant UO_2 fuels: for example, large grain doped UO_2 fuel in order to reduce internal pressure of UO_2 fuel rods at the End Of Life under steady state conditions and to improve the thermomechanical behaviour under transient conditions - Mitigation of fission gas (xenon, krypton) released and the Fuel Fragmentation Relocation and Dispersal			
Technology readiness level	9	**Likeliness of use**	100%
Other observations			
- Improvements and reduction of uncertainty margins needed to take into account more stringent safety constraints (regulators)			
List of authors of the ID card			
Rodrigue Largenton (EDF), Bruno Michel (CEA)			

4.2 MOX fuels for LWR

Compositions of interest	Pu content between ~4 and 12%, O/M ratio between ~1.98 and 2.01, a few % of Am (between ~0.5 and 2%)		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor	X		PWR, BWR …
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy-cooled fast reactor			
Gas-cooled fast reactor			
(Very) high temperature reactor			
Supercritical water-cooled reactor			
Molten salt reactor			
Other (specify here)			
Fusion			
Type of component	Fuel		
Manufacturing method	Ceramic powder technology (MIMAS…)		
Joining method	N/A		
Target operational environment	Cladding temperature (outer surface) ~260-330°C – Typical operating central temperature inside the fuel ~1000°C (temperature at 30-40 % of the melting point) – Average fuel rod Burn-up targeted ~60-70 GWd/t => 3 or 4 years of residence in core – Middle (example BWR: around ~70-80 bar) or high (example PWR: ~123-160 bar) pressure on outer clad and middle pressure in fuel rods (initial pressure is between ~15 and 20 bar for more recent PWR fuel fabrications and ~2–8 bar for BWR fuels – this pressure increases with the irradiation level) – Flux of neutrons with high energy (around 1.10^{14} n/cm^{-2}/s in the core centre)		

Properties that make the material suitable for the target environment

- High melting point providing large operating temperature margins
- High irradiation and thermal creep rates
- No allotropic changes, excellent stability
- Good behaviour under irradiation in normal and transient conditions
- Excellent behaviour in transient conditions (ramp tests: PCMI), thermal creep rate being much higher than for UO_2 fuel
- Good compatibility with fuel rod cladding
- Good compatibility with water coolant in case of fuel rod failure

References:

- D. Baron, L. Hallstadius, K. Kulacsy, R. Largenton, J. Noirot, Comprehensive Nuclear Materials Second Edition Chap. 2.02 Fuel Performance of Light Water Reactors (Uranium Oxide and MOX), Elsevier, 2020, ISBN- 978-0-08-102866-7
- J-P. Ottaviani, D. Staicu, R. Calabrese, N. Vér, G. Trillon, J. Klousal, A. Fedorov, S. Portier, M. Verwerft, ESNII+ project Deliverable D7.11 State of the art with a literature review of MOX fuel properties, European Commission Preparing ESNII for HORIZON 2020, Grant agreement no: 605172, 2015

Open issues to be investigated[¡Error! Marcador no definido.]

- Margin to fuel melting: Some progresses are still needed in the characterization and understanding of thermal properties evolutions under irradiation and at high temperature (thermal conductivity, heat capacity, melting temperature …)

- Behaviour of MOX fuel under steady state: Some progresses are still needed in the characterizations and understanding of microstructure evolutions (high burnup structure (HBS) and restructuring occurring in the centre of high burnup MOX fuel) - Behaviour of MOX fuel under transient conditions (Ramp, RIA, LOCA…): Some progresses are still needed in the characterizations and understanding of mechanical properties evolutions under irradiation and especially at high temperature: elastic properties and thermal expansion, thermal creep, local (within grain and on the boundaries of it) rupture properties - Some progresses are still needed on modelling of MOX fuel under transient conditions (RIA, LOCA…): damage of grain boundaries, inter-phase cracking, fission gas released, Fuel Fragmentation Relocation and Dispersal. - Optimize fabrication to reduce internal pressure of MOX fuel rods at the End Of Life under steady state conditions and to improve the thermomechanical behaviour under transient conditions (actions related to current MOX fuel rods and to the future possible multirecycling) References - D. Baron, L. Hallstadius, K. Kulacsy, R. Largenton, J. Noirot, Comprehensive Nuclear Materials Second Edition Chap. 2.02 Fuel Performance of Light Water Reactors (Uranium Oxide and MOX), Elsevier, 2020 - J. Carbajo, G. Yoder, S. Popov, V. Ivanov, A review of the thermophysical properties of MOX and UO_2 fuels, Journal of Nuclear Materials 299 (2001) 181-198			
Property improvements required or in progress			
- Optimize fabrication to increase homogeneity and grain size within MOX fuel in order to reduce internal pressure of MOX fuel rods at the End Of Life under steady state conditions and to improve the thermomechanical behaviour under transient conditions - Mitigation of fission gas (xenon, krypton and helium) released and the Fuel Fragmentation Relocation and Dispersal			
Technology readiness level	9	**Likeliness of use**	100%
Other observations			
- Improvements and reduction of uncertainty margins needed to take into account more stringent safety constraints (regulators)			
List of authors of the ID card			
Rodrigue Largenton (EDF), Bruno Michel (CEA)			

4.3 MOX fuels for fast reactors

Compositions of interest	Pu content between 20 and 30%, O/M ratio between 1.93 and 1.99, a few % of Am		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor	X		
Lead alloy-cooled fast reactor	X		
Gas-cooled fast reactor	X		
(Very) high temperature reactor	X		
Supercritical water-cooled reactor			
Molten salt reactor			
Other (specify here)			
Fusion			
Type of component	Fuel		
Manufacturing method	Ceramic powder technology		
Joining method	N/A		
Target operational environment	Cladding temperature around 600-650°C, typical operating temperatures inside the fuel between 30 and 80 % of the melting point, Burn-up targeted: 100-150 GWd/t (~10-15% FIMA), 400-800 days of residence, very intense flux (7.10^{15} n/cm^{-2}/s in the core centre) of neutrons with high energy		
Properties that make the material suitable for the target environment			
High melting point, no allotropic changes, excellent stability, - excellent behaviour under irradiation, in particular, swelling rate much lower than other fuels Refs: Y. Guerin, Fuel Performance of Fast Spectrum Oxide Fuel Main Author(s),chap. 2.21 of Comprehensive Nuclear Materials, Elsevier, 2012			
Open issues to be investigated			
- Margin to fuel melting: evolution of thermal properties under irradiation, thermal conductivity - Knowledge of irradiation defects, evolution of microstructure: fracture, high burnup structure (HBS), evolution of composition: Pu relocation and O/M variation - Fission gas and helium behaviour, transport of non-gaseous fission products - Mechanical fuel-cladding interaction: thermal expansion, creep (thermal and irradiation-induced), Chemical fuel-cladding interaction: cladding corrosion, Joint Oxyde-Gaine (JOG), Reaction Oxyde Gaine (ROG), Fuel/coolant compatibility issues - Optimize fabrication to reduce dust issues in the context of multirecycling, consequence of multi-recycling of Pu and higher concentrations of ^{241}Pu which will lead to higher ^{241}Am contents - Corium composition and properties Ref: D.R. Olander, Fundamental aspects of nuclear reactor fuel elements, Technical information center, 1985; Strategic research agenda of the EERA-JPNM, 2019 - Strategic research agenda of the EERA-JPNM, 2019, http://www.eera-jpnm.eu/filesharer/documents/Materials%20for%20Sustainable%20NuclearEnergy%20-%20SRA%20of%20the%20EERA-JPNM%20-%20web%20version.pdf			
Property improvements required or in progress			
Optimize fabrication to yield more homogeneous material, reduce dust issues in the context			

of multirecycling, mitigation of cladding corrosion			
Technology readiness level	7-8	**Likeliness of use**	90
Other observations			
- MOX used in FR in the past. Reference fuels for ESNII systems. Improvement needed to take into account new reactor designs and more stringent safety constraints			
List of authors of the ID card			
Marjorie Bertolus (CEA), Marco Cologna (JRC)			

4.4 Carbide fuels

<table>
<tr><td>Compositions of interest</td><td colspan="3">$(U,Pu)C_{1+x}$, Pu ~20% to 70%, minor actinides also possible</td></tr>
<tr><td>Type of reactor</td><td>Large reactor</td><td>SMR (if applicable)</td><td>Notes</td></tr>
<tr><td>Light water reactor</td><td></td><td></td><td></td></tr>
<tr><td>Heavy water reactor</td><td></td><td></td><td></td></tr>
<tr><td>Sodium-cooled fast reactor</td><td>X</td><td>X</td><td></td></tr>
<tr><td>Lead alloy-cooled fast reactor</td><td>X</td><td>X</td><td></td></tr>
<tr><td>Gas-cooled fast reactor</td><td>X</td><td>X</td><td></td></tr>
<tr><td>(Very) high temperature reactor</td><td>X</td><td>X</td><td></td></tr>
<tr><td>Supercritical water-cooled reactor</td><td></td><td></td><td></td></tr>
<tr><td>Molten salt reactor</td><td></td><td></td><td></td></tr>
<tr><td>Other (specify here)</td><td></td><td></td><td></td></tr>
<tr><td>Fusion</td><td></td><td></td><td></td></tr>
<tr><td>Type of component</td><td colspan="3">Fuel</td></tr>
<tr><td>Manufacturing method</td><td colspan="3">Ceramic powder technology, carbothermic reduction, hydriding-dehydriding, direct synthesis by arc-melting, inert atmosphere</td></tr>
<tr><td>Joining method</td><td colspan="3">N/A</td></tr>
<tr><td>Target operational environment</td><td colspan="3">Metal cooled fast reactors: 600°C – 700°C cladding temperature, Burnup limit ~15 at.% or more.
Gas cooled fast reactor: Cladding temperatures up to 1200°C. Burnup limit ~15 at.% or more.</td></tr>
<tr><td colspan="4">Properties that make the material suitable for the target environment</td></tr>
<tr><td colspan="4">- Higher fissile density compared to oxide fuels
- High thermal conductivity (~5 times higher than oxide): higher margin to melting, more efficient heat transfer to coolant
- Chemical compatibility with liquid sodium</td></tr>
<tr><td colspan="4">Open issues to be investigated</td></tr>
<tr><td colspan="4">- Influence of composition, microstructure, stoichiometry and impurities (oxygen and nitrogen primarily) on fundamental and safety relevant thermophysical properties: thermal conductivity, thermal expansion, melting point, high temperature mechanical properties…
- Evolution of properties with irradiation
- High swelling during irradiation
References
A.K.Sengupta et al. „Carbide Fuel", in in Comprehensive Nuclear Materials, Vol. 3, 2012, Pages 55-86.
D. Manara et al. "Thermodynamic and Thermophysical Properties of the Actinide Carbides", in Comprehensive Nuclear Materials, Vol 2, 2012, Pages 87-137
Strategic research agenda of the EERA-JPNM, 2019, http://www.eera-jpnm.eu/filesharer/documents/Materials%20for%20Sustainable%20NuclearEnergy%20-%20SRA%20of%20the%20EERA-JPNM%20-%20web%20version.pdf</td></tr>
<tr><td colspan="4">Property improvements required or in progress</td></tr>
<tr><td colspan="4">- Compatibility tests between fuel, cladding and coolant (Carburization of claddings - material dependent)

Limited Fabrication/irradiation experience, some processing challenges:
- Thermal stability, potential vaporisation of Pu or MA during processing
- Evaluation of pyrophoricity (particularly during fine powder handling) and oxidation behaviour (thermochemical modelling of the reaction with air and moisture, microstructural</td></tr>
</table>

studies of the oxidation mechanisms) - High-purity inert cover gas required for fuel fabrication, maintenance of C/M ratio difficult, difficult to control oxygen impurities - Dissolution not simple (larger presence of insoluble platinum group metal –PGM- particles and possible formation of organic compounds)			
Technology readiness level	5	**Likeliness of use**	50%
Other observations			
- Relative density 80~90%, grain size ~10 µm.			
List of authors of the ID card			
D. Freis (JRC), M. Cologna (JRC), P. Olsson (KTH)			

4.5 Nitride fuels

Composition of interest	UN, (U,Pu)N, (U,Pu,MA)N, UN-U_3Si_2, (U,Zr)N, (Pu, MA,Zr)N, UN-UO_2 (MA = minor actinides Am, Np, Cm)		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor	X	X	UN, UN-U_3Si_2
Heavy water reactor			
Sodium-cooled fast reactor	X	X	Driver fuels : UN, (U,Pu)N, (U,Pu,MA)N
Lead alloy-cooled fast reactor	X	X	
Gas-cooled fast reactor	X		
(Very) high temperature reactor	X		
Supercritical water-cooled reactor	X		
Molten salt reactor			
Other (specify here): ADS, Space applications	X		(Pu,MA,Zr)N UN and (U,Zr)N
Fusion			
Type of component	Fuel		
Manufacturing method	direct nitriding, hydriding–dehydriding–nitriding, carbothermic nitriding, direct ammonolysis of uranium fluorides, spark plasma sintering		
Joining method	N/A		
Target operational environment	Metal cooled fast reactors: 600°C – 700°C cladding temperature, Burnup limit ~15 at.% or more.		

Properties that make the material suitable for the target environment

- Higher fissile density (40% more uranium in UN than in UO_2): enabling one to operate the fuel at a higher linear power, leading to higher conversion ratios and potentially higher burn-ups. May help to reduce the physical size of the core
- Higher thermal conductivity: reduction of the fuel centreline temperature, decrease in the energy stored per unit mass while increasing the margin for fuel melting, and delay the migration of fission products and actinides, which positively affects the fuel swelling
- Good chemical compatibility with most potential cladding materials, as well as irradiation stability.
- Longer fuel cycle time thanks to neutronic behaviour of UN in the core, the fuel cycle could increase from the commonly applied 18 months (standard UO_2) to about 25 months, based on a burnup of 50 GWd/tU. This increase leads to fewer shut downs for reloading, thus being an economical benefit for nitride fuel implementation
- Chemical compatibility with liquid sodium
- The mononitrides of uranium and plutonium are completely miscible
- Good dissolution in nitric acid (HNO_3), making this fuel compatible with the PUREX process, which uses HNO_3 for dissolution of spent fuel. This should lead to an easier reprocessing to reprocess than alternative fuels

References

- Matzke, H. Science of Advanced LMFBR Fuels: Solid State Physics, Chemistry, and Technology of Carbides, Nitrides, and Carbonitrides of Uranium and Plutonium; Elsevier: Amsterdam, Netherlands, 1986.
- Janne , Wallenius, Nitride Fuels, R.J.M. Konings, R.E. Stoller, (Eds.), Comprehensive Nuclear Materials (Second Edition), Elsevier, Oxford, 2020, pp. 88-101
- D. Staicu, R.J.M. Konings, T. Wiss, O. Beneš, C. Guéneau, J. Noirot, Plutonium in Nuclear Fuels, Plutonium Handbook, ISBN: 978-0-89448-201-4, 2019, 2nd Edition, chap. 29, pp. 2312-2321
- C. Ekberg, D. Ribeiro Costa, M. Hedberg, M. Jolkkonen. Nitride fuel for Gen IV nuclear power systems. J Radioanal Nucl Chem 318, 1713–1725 (2018).

Open issues to be investigated			
- Influence of composition, microstructure, stoichiometry and impurities (oxygen and carbon primarily) on fundamental and safety relevant thermophysical properties: thermal conductivity, thermal expansion, melting point, high temperature mechanical properties… - Evolution of properties with irradiation - Compatibility tests between fuel, cladding and coolant - Limited Fabrication/irradiation experience, some processing challenges: - Because the radiotoxic isotope ^{14}C is formed as a result of the $^{14}N(n,p)^{14}C$ reaction during reactor irradiation, the nitride fuel must be fabricated with enriched ^{15}N (natural nitrogen contains 0.37% ^{15}N and 99.63% ^{14}N). Because enrichment is relatively costly and an enrichment of at least 99% is required, the nitrogen must be recycled during fabrication and eventually during reprocessing - Pyrophoric, particularly for fine powders. Needs to be handled in inert gas. - Oxidation behaviour in different conditions may need further study (e.g. thermochemical modelling of the reaction with air and moisture, microstructural studies of the oxidation mechanisms) - High-purity inert cover gas required for fuel fabrication. Impurity levels well controlled for metal hydride-nitride route; not so well-controlled for carbothermic reduction			
Property improvements required or in progress			
- High swelling during irradiation - For deployment of nitrides in LWR environment, improvement of stability of the nitrides in water at 300°C by addition of dopant elements or compounds that can form protective layers around the grains, such as Cr, ZrO_2, Th, into the fuel pellet. Compound UN-UOX fuels are also studied.			
Technology readiness level	8	**Likeliness of use**	90%
Other observations			
Russia is currently the only country that has an advanced programme on nitride fuels. The Russian nuclear fuel manufacturer TVEL develops experimental fuel assemblies made of nitride fuel for the BREST and Beloyarsk nuclear power plant (BN) fast neutron reactors. The construction of a facility to produce high-density U-Pu nitride fuels has been announced.			
List of authors of the ID card			
T. Wiss, JRC, P. Olsson, KTH			

4.6 Uranium silicide fuel for LWR

Compositions of interest	U_3Si_2		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor	X		Accident Tolerant Fuel
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy-cooled fast reactor			
Gas-cooled fast reactor			
(Very) high temperature reactor			
Supercritical water-cooled reactor			
Molten salt reactor			
Other			
Fusion			
Type of component	Fuel		
Manufacturing method	Powder metallurgy techniques: comminution of uranium silicide ingot, pellet pressing and sintering		
Joining method	N/A		
Target operational environment	Cladding: Cr-coated Zr alloys or SiC-SiC composites Uranium silicide (U_3Si_2) has 17% higher uranium density than UO_2 providing significant fuel savings by 10% to 15% fewer assemblies to be loaded for typical 18 month fuel cycles.		

Properties that make the material suitable for the target environment

- High Uranium density: 11.3 (g/cm^3)
- Melting temperature: 1665 °C
- Thermal conductivity: 9-20 W/(m·K) [300-1200 °C]. The higher thermal conductivity than that of UO_2 allows the fuel to respond to transients and prevent centreline melt conditions and significantly lowers the amount of energy stored in the core, which is beneficial in LOCA event.

References:

E. J. Lahoda, F. A. Boylan, Development of LWR Fuels with Enhanced Accident Tolerance ATF Feasibility Analysis and Final Technical Report Deliverable for the Westinghouse Accident Tolerant Fuel Program (Rev. 2). 2019. https://doi.org/10.2172/1511013

State-of-the-Art Report on Light Water Reactor Accident-Tolerant Fuels, Nuclear Energy Agency report No. 7317, 2018

Fabiola Cappia, Post-Irradiation Examinations of the ATF Experiments - 2020 Status, Report INL EXT-20-59619 (2020), https://doi.org/10.2172/1773801

Open issues to be investigated

- Irradiation of the U_3Si_2 with SiC and Cr coated Zr cladding is required to determine the performance aspects of the fuel and cladding
- Key issues are swelling, creep , degradation of thermal conductivity, FGR, in-core oxidation resistance and interactions between the fuel and cladding, especially SiC behaviour in reactor environments
- Development work to allow manufacture of U_3Si_2 directly from UF_6, bypassing the U metal stage to reduce the cost of manufacturing

Property improvements required or in progress

Lead test assemblies (LTA) of Westinghouse EnCore fuel rods containing U_3Si_2 fuel pellets in Cr-coated Zr cladding have been loaded into a commercial nuclear power plant (NPP) Exelon's Byron unit 2 (USA) in September 2019. They will stay in reactor for up to six years and will be examined between each refuelling cycle every 18 to 24 months. The fuel will then be used in transient tests to help determine the safe operating limits. LTAs of U_3Si_2 fuel pellets in SiC ceramic matrix composite cladding could be loaded into a reactor by 2022.			
Technology readiness level	6-7	**Likeliness of use**	70%
Other observations			
List of authors of the ID card			
V. Grišmanovs (NRG)			

4.7 Uranium silicide fuel dispersed in Al matrix for Material Test Reactors

Compositions of interest	U_3Si_2 (Nominal composition) (LEU: 19.75% ^{235}U)		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy-cooled fast reactor			
Gas-cooled fast reactor			
(Very) high temperature reactor			
Supercritical water-cooled reactor			
Molten salt reactor			
Other:			MTR
Fusion			
Type of component	Fuel		
Manufacturing method	Fuel plate manufacturing process		
Joining method	N/A		
Target operational environment	Plate type configuration. Cladding (Al alloy) temperature: 70 - 100 °C, typical operating temperatures inside the fuel: 100 - 150 °C (15-20 % of the melting point). Peak Burnup targeted: 80% ^{235}U (4.2 x 10^{21} f/cm^3(fuel volume), 15% FIMA), Power: 2 kW/gU, 200 days of residence		
Properties that make the material suitable for the target environment			
• High U-density, needed for conversion from HEU (93% ^{235}U) to LEU (~19% ^{235}U) without loss of performance • Excellent irradiation performances at low power (< 140 W/cm^2) up to high burnups, which resulted in fuel qualification for use in low power research reactors in 1978 (U.S. NRC NUREG-1313) • Good heat transfer properties, mostly given by the Al matrix in which the U_3Si_2 particles are dispersed • Reprocessing of spent fuel available at industrial scale Refs: A. Leenaers, et al., U-Si Based Fuel System, chap. 5.15 of Comprehensive Nuclear Materials (2nd Edition), Elsevier, 2020, ISBN 9780081028667. Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for use in Non-Power Reactors. NUREG-1313. U.S Office of Nuclear Reactor Regulation, 1988.			
Open issues to be investigated			
• Fuel behaviour at high power (and temperature) and high loading (volume fraction of fuel phase in Al matrix), including the evolution of fission gas bubbles and the U_3Si_2-Al interaction; • Evolution of fuel plate's thermal conductivity with burnup.			
Property improvements required or in progress			
High power irradiations of high loaded U_3Si_2 dispersed fuel plates and post irradiation examinations are currently underway at SCK.CEN. Depending on their outcome, property improvements will be evaluated			
Technology readiness level	7-8	**Likeliness of use**	95%
Other observations			
• Fuel nominal composition is U_3Si_2, but, during fabrication and irradiation, other phases (USi,			

U_3Si, U,…) may form • Fuel amorphization occurs soon after the irradiation starts at the operational temperatures of MTRs (100 - 150 °C).
List of authors of the ID card
D. Salvato (SCK.CEN)

4.8 Uranium-Molybdenum fuel dispersed in Al matrix for Material Test Reactors

Compositions of interest	U 93 wt.% - Mo 7 wt.% (LEU: 19.75% ^{235}U)		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy-cooled fast reactor			
Gas-cooled fast reactor			
(Very) high temperature reactor			
Supercritical water-cooled reactor			
Molten salt reactor			
Other:			MTR
Fusion			
Type of component	Fuel		
Manufacturing method	Fuel plate manufacturing process		
Joining method	N/A		
Target operational environment	Plate type configuration. Cladding (Al alloy) temperature: 70 - 100 °C, typical operating temperatures inside the fuel: 100 - 150 °C (25-30 % of the melting point). Peak Burnup targeted: 80% ^{235}U (6.2×10^{21} f/cm^3(fuel volume), 15% FIMA), Power: 2 kW/gU, 200 days of residence		

Properties that make the material suitable for the target environment

- High U-density, needed for conversion from HEU (93% ^{235}U) to LEU (~19% ^{235}U) without loss of performance;
- Good heat transfer properties, mostly given by the Al matrix in which the UMo particles are dispersed;
- Alloying with Mo stabilizes, upon quenching, the γ-U phase at low temperature, which has better irradiation performance compared to α-U;
- Fuel crystallinity preserved during irradiation;
- Predictable fission gas behaviour, with an ordered array of nanobubbles forming at low fission density (unprecedented in a nuclear fuel);
- Excellent irradiation performances at low/medium power (< 300 W/cm^2) up to high burnups

References

A. Leenaers, et al., U-Mo Based Fuel System, chap. 5.16 of Comprehensive Nuclear Materials (2nd Edition), Elsevier, 2020, ISBN 9780081028667

Open issues to be investigated

- Comprehension of all contributions to fuel plate swelling at high power (> 300 W/cm^2) so that it can be better predicted;
- Minimization of the interaction between the UMo particles and the Al matrix, which impacts the mechanical integrity of the irradiated fuel plate at high power and high burnup;
- Evolution of fuel plate's thermal conductivity with burnup
- Origin of the nanobubble lattice at low burnups is still unknown. Some progresses achieved recently. Specifically designed neutron and/or ion irradiations are required to shed more light onto it.

D. Salvato et al., The initial formation stages of a nanobubble lattice in neutron irradiated U(Mo), Journal of Nuclear Materials 529 (2020) 151947

Property improvements required or in progress

<table>
<tr><td colspan="4">• Coating of UMo with inert diffusion barrier (ZrN) and/or adding of Si to the Al matrix to reduce interaction between UMo particles and Al matrix;
• Heat treatment of the UMo particles to delay fuel recrystallization at intermediate-high burnups and, as such, delay the resulting acceleration in fuel swelling rate;
• Development of computer codes to predict fuel plate swelling.
A. Leenaers, et al., ZrN coating as diffusion barrier in UMo dispersion fuel systems, Journal of Nuclear Materials 552 (2021) 153000</td></tr>
<tr><td>Technology readiness level</td><td>6-7</td><td>Likeliness of use</td><td>80%</td></tr>
<tr><td colspan="4">Other observations</td></tr>
<tr><td colspan="4"></td></tr>
<tr><td colspan="4">List of authors of the ID card</td></tr>
<tr><td colspan="4">D. Salvato (SCK.CEN)</td></tr>
</table>

4.9 Liquid fuels: molten salts

Compositions of interest	Fluorides of alkali metals and alkaline earth metals optionally mixed with fluorides of actinides. The Li as one of the considered cation is usually enriched in ^{7}Li to minimize the neutron captures on ^{6}Li, whose capture cross-section is large. Detailed composition depends on the foreseen application and for fuel salt search for balance between neutronic performance, liquidus temperature and sufficient actinides solubility. Typical examples: $LiF\text{-}BeF_2$: eutectic composition with low melting temperature. Applicable as coolant of MSR or HTR. $AnF_4\text{-}LiF\text{-}BeF_2$: typical fuel or blanket salt for thermal breeder in Th-U cycle. Be has certain moderation power. Is therefore avoided in non-thermal molten salt reactors (MSRs). $AnF_4\text{-}LiF$: typical fuel or blanket salt for fast MSR operated in Th-U cycle. Li, with strong scattering cross-section resonance however suppresses fast neutron spectrum. Alternative cations with slightly higher neutron capture probability are Na and K. These cannot be used in thermal breeder.

Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy-cooled fast reactor			
Gas-cooled fast reactor			
(Very) high temperature reactor		X	
Supercritical water-cooled reactor			
Molten salt reactor	X	X	sizeable
Other (specify here)			
Fusion	X		^{6}LiF can be considered as tritium source in fusion blanket

Type of component	Fuel or coolant of thermal and fast spectrum reactors. Applicability to closed U-Pu cycle is limited because of F and Li scattering resonances in fast spectrum area and because of limited solubility of trivalent fluorides, e.g. PuF_3. This solubility limit relates also achievable burnup, because many soluble FPs are also trivalent.
Manufacturing method	Synthesis/purification of pure components by fluorination (HF, F2, NF3 …) powder mixing in solid state
Joining method	N/A
Target operational environment	Molten Fluoride Reactors: 600°C – 800°C fuel temperature. Lower melting temperature for $LiF\text{-}BeF_2$ coolant. Burnup is limited by solubility of respective FPs fluorides and is generally lower than in chlorides.

	Gaseous and volatile FPs may be separated from the fuel salt during operation. Carrier salt and actinides fluorides have unlimited life span under irradiation. Accordingly, is also the soluble FPs would be separated, they have unlimited use.
Properties that make the material suitable for the target environment	
- Low melting points (melting point is a function of compositions and its minimization may result in lower density) - Low vapour pressure combined with possibility to remove components with high volatility, e.g. ZrF_4. - Reasonable solubility for tetravalent actinide fluorides. - No thermal decomposition of fuel matrix at high T, ongoing recombination. - Chemical compatibility with structural materials, e.g. Hastelloy, which however requires redox potential control. - Low viscosity - Possibility to convect the heat from active core due to the liquid nature	
Open issues to be investigated	
- Impact of long term corrosion of salt towards structural materials - Fundamental properties of selected fuel salt composition - coupling chemistry with multi-physics - Properties change with burn-up - Trivalent actinide fluorides solubility. - FPs solubility limits at high burnups. - Tritium management in case of LiF-based matrix use - Reprocessing scheme and related safeguards. - Redox potential control and online monitoring.	
Property improvements required or in progress	
Fuel composition optimizations are ongoing as MSR's allow variety of salt variations based on targeted concept (burner, breeder, fast, thermal, etc.)	

Technology readiness level	3	**Likeliness of use**	70
Other observations			
List of authors of the ID card			
O. Benes (JRC), J. Krepel (PSI)			

Compositions of interest	Chlorides of actinides, chlorides of alkali metals and alkaline earth metals, and mixture of previous. The chlorine is usually enriched in ^{37}Cl to minimize the neutron captures on ^{35}Cl, whose capture cross-section is large; at the same time under revision. Composition depends on the foreseen fuel cycle type and usually targets balance between neutronic performance, liquidus temperature and chemical stability. Reference salt: $NaCl-UCl_3-PuCl_3$: high neutronic performance, density competing with liquidus temperature, which is minimal around 65% mol of NaCl. Alternative additive cations with slightly higher neutron capture probability are Mg, K, and Ca. These are usually not foreseen for a breeder or breed-and-burn reactor.

Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**

Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor			
Lead alloy-cooled fast reactor			
Gas-cooled fast reactor			
(Very) high temperature reactor			
Supercritical water-cooled reactor			
Molten salt reactor	X	X	sizeable
Other (specify here)			
Fusion			

Type of component	Fuel or coolant of fast spectrum reactor. Chlorides are not foreseen for thermal spectrum because of their higher neutron capture probability. From the same reason Li and Be are avoided, because of their moderation power (Be) and scattering resonances in fast spectrum (Li). Applicability of chlorides to closed Th-U cycle is limited and may result in very bulky cores.
Manufacturing method	Synthesis/purification of pure components by chlorination (Cl_2, CCl_4 …) for chlorides; powder mixing in solid state
Joining method	N/A
Target operational environment	Molten Chloride Reactors: 500°C – 800°C fuel temperature, burnup is limited by solubility of respective FPs chlorides and is generally higher than in fluorides. Gaseous and volatile FPs may be separated from the fuel salt during operation. Carrier salt and actinides chlorides have unlimited life span under irradiation. Accordingly, is also the soluble FPs would be separated, they have unlimited use.

Properties that make the material suitable for the target environment
- Low melting points (melting point is a function of composition and its minimization may result in lower density and actinide content) - Low vapour pressure combined with possibility to remove components with high volatility. - Reasonable solubility for actinide chlorides - No thermal decomposition of fuel matrix at high T, ongoing recombination. - Chemical compatibility with structural materials, e.g. stainless steel, which however requires redox potential control. - Low viscosity - Possibility to convect the heat from active core due to the liquid nature
Open issues to be investigated
- Impact of long term corrosion of salt towards structural materials - Fundamental properties of selected fuel salt composition - Properties change with burn-up - FPs solubility limits at high burnups. - Reprocessing scheme and related safeguards. - Redox potential control and online monitoring.
Property improvements required or underway
Fuel composition optimizations are ongoing as MSR's allow variety of salt variations based on targeted concept (burner, breeder, fast, thermal, etc.)

Technology readiness level	3	**Likeliness of use**	70
Other observations			
List of authors of the ID card			
O. Benes (JRC), J. Krepel (PSI)			

4.10 Minor actinide bearing driver fuels for fast reactors

Composition of interest	$(U,Pu,Am,Np)O_2$ As-fabricated: Pu content between 20 and 30%; O/M ratio between 1.93 and 1.99; Am, Np < 5%		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor	X	Potentially	Scarce exp. data currently available, no experiences in SMRs.
Lead alloy-cooled fast reactor	X	Potentially	
Gas-cooled fast reactor	X	Potentially	
(Very) high temperature reactor	X	Potentially	
Supercritical water-cooled reactor			
Molten salt reactor			
Other (specify here)			
Fusion			
Type of component	Fuel		
Manufacturing method	Ceramic powder technology		
Joining method	N/A		
Target operational environment	Same of fast-reactor MOX: cladding temperature ~ 600-650°C, fuel operating temperatures ~ 80% of the melting point, target burn-up ~ 150 GWd/t (~ 20% FIMA), 400-800 days of residence, intense flux (~ 10^{16} n/cm^{-2} s^{-1}, core centre) of high-energy neutrons (> MeV).		

Properties that make the material suitable for the target environment

Common with fast-reactor MOX
- high melting point (degrading with burn-up: down to ~ 2950 K at ~ 100 GWd/t)
- no allotropic changes, high micro-structural stability
- behaviour under irradiation: lower swelling rate than other fuels

References:

- Y. Guerin, "Fuel Performance of Fast Spectrum Oxide Fuel", in: Comprehensive Nuclear Materials, Chap. 2.21, Elsevier, 2012, ISBN 978-0-08-056027-4
- F. Delage et al., "Actinide-Bearing Fuels and Transmutation Targets", in: Comprehensive Nuclear Materials (Second Edition), Chap. 5.19, Elsevier, 2020, ISBN 978-0-08-102866-7

Open issues to be investigated

- Margin to fuel melting: evolution of thermal properties under irradiation (thermal conductivity, melting point, impact of Am, Np contents besides O/M effect)
- Knowledge of irradiation defects: evolution with fuel burn-up
- Evolution of composition: Pu relocation and O/M variation
- Evolution of microstructure: high burnup structure (HBS)?
- Fission gas and helium behaviour
- Transport of non-gaseous fission products
- Mechanical fuel-cladding interaction: thermal expansion, creep (thermal and irradiation-induced)
- Chemical fuel-cladding interaction: inner cladding corrosion, Joint Oxyde-Gaine (JOG), Reaction Oxyde-Gaine (ROG)
- Fuel/coolant compatibility issues
- Optimize fabrication processes (multi-recycling context)
- Consequence of multi-recycling of Pu and higher concentrations of ^{241}Pu which will lead to higher contents of ^{241}Am

- Corium composition and properties References: - D.R. Olander, "Fundamental aspects of nuclear reactor fuel elements", Technical information center, Energy Research and Development Administration, 1985, doi:10.2172/7343826 - Strategic research agenda of the EERA-JPNM, 2019, http://www.eera-jpnm.eu/filesharer/documents/Materials%20for%20Sustainable%20NuclearEnergy%20-%20SRA%20of%20the%20EERA-JPNM%20-%20web%20version.pdf
Property improvements required or underway¡Error! Marcador no definido.
- Optimize fabrication to yield the highest material homogeneity, optimize multi-recycling process (in common with fast-reactor MOX) - Investigation and mitigation of cladding corrosion in case of interaction fuel-cladding (in common with fast-reactor MOX) - Further assess impact of homogeneous minor actinides < 5%: really negligible? (on thermal properties: thermal conductivity, melting temperature)

Technology readiness level	3	**Likeliness of use**	30%

Other observations
- Likeliness of use: low, due to current limited plans for MA-MOX fuel use as driver fuel (e.g., in Gen-IV concepts). Possibly in MYRRHA for transmutation studies, but first steps should employ U-Pu MOX, then Am-MOX pins in experimental positions (not as driver fuel). - Effects of homogeneous minor actinides (Am, Np < 5%) currently deemed limited, hence material properties and behaviour assumed equal to U-Pu MOX (same Pu content). - Current knowledge from MOX irradiated in past fast reactors / experimental facilities. Improvement needed to take into account new reactor designs (Gen-IV) and more stringent safety constraints. - MA-MOX irradiations reported in literature: - SUPERFACT-1 (Phénix): Am-MOX and Np-MOX, homogeneous (Am, Np ~ 2%) and heterogeneous (Am, Np ~ 50%) - SPHERE (HFR): homogeneous Am-MOX (Am ~ 3%) - MARINE (HFR): heterogeneous Am-MOX (Am ~ 15%) Reference: - J. Babelot, N. Chauvin, "Joint CEA/JRC Synthesis Report of the Experiment SUPERFACT 1", Report JRC-ITU-TN-99/03, 1999 - E. D'Agata et al., "SPHERE: Irradiation of sphere-pac fuel of $UPuO_{2-x}$ containing 3% Americium", Nuclear Engineering and Design 275 (2014) 300-311 - E. D'Agata et al., "The MARINE experiment: Irradiation of sphere-pac fuel and pellets of UO_{2-x} for americium breading blanket concept", Nuclear Engineering and Design 311 (2017) 131-141 - A. Gallais-During et al., "Outcomes of the PELGRIMM project on Am-bearing fuel in pelletized and sphere-pac forms", Journal of Nuclear Materials 512 (2018) 214-226
List of authors of the ID card
Alessandro Del Nevo (ENEA), Alessio Magni (Polimi, ENEA)

4.11 Minor actinide bearing blanket fuels

Composition of interest	$(U,Am)O_2$. As-fabricated: Am content up to 30%; O/M ratio between 1.93 and 1.99		
Type of reactor	**Large reactor**	**SMR (if applicable)**	**Notes**
Light water reactor			
Heavy water reactor			
Sodium-cooled fast reactor	X	Potentially	Scarce exp. data currently available, no experiences in SMRs.
Lead alloy-cooled fast reactor	X	Potentially	
Gas-cooled fast reactor	X	Potentially	
(Very) high temperature reactor			
Supercritical water-cooled reactor			
Molten salt reactor			
Other: ADS			MYRRHA
Fusion			
Type of component	Americium Bearing Blanket		
Manufacturing method	Powder metallurgy, sol-gel and ion exchange Weak Acid Resin (WAR) technologies		
Joining method	N/A		
Target operational environment			

Properties that make the material suitable for the target environment

Common with fast-reactor MOX and homogeneous MA transmutation fuel
- high melting point
- no allotropic changes, micro-structural stability

References:

- F. Delage, L. Ramond, A. Gallais-During, S. Pillon, 5.19 - Actinide-Bearing Fuels and Transmutation Targets, in: R.J.M. Konings, R.E. Stoller (Eds.), Comprehensive Nuclear Materials (Second Edition), Elsevier, Oxford, 2020: pp. 645–683. https://doi.org/10.1016/B978-0-12-803581-8.12049-1.
- E. Epifano et al., Melting behaviour of uranium-americium mixed oxides under different atmospheres, The Journal of Chemical Thermodynamics. 140 (2020) 105896. https://doi.org/10/gh4tvr.
- E. Epifano, Study of the U-Am-O ternary phase diagram, Ph.D. thesis, Université Paris-Saclay, 2017. https://pastel.archives-ouvertes.fr/tel-01852208/document .
- T. Kooyman, Current state of partitioning and transmutation studies for advanced nuclear fuel cycles, Annals of Nuclear Energy. 157 (2021) 108239. https://doi.org/10/gm82cc.

Open issues to be investigated¡Error! Marcador no definido.

- Knowledge of irradiation defects: evolution with fuel burn-up
- Evolution of composition: O/M variation
- Evolution of microstructure : open and closed porosity
- Helium behaviour
- Mechanical fuel-cladding interaction: thermal expansion, creep (thermal and irradiation-induced)
- Better knowledge of the U-Am-O phase diagram => O/M control and assessment
- lack of data on thermal properties
- self-irradiation (prior irradiation)

References:

- Strategic research agenda of the EERA-JPNM, 2019, http://www.eera-jpnm.eu/filesharer/documents/Materials%20for%20Sustainable%20NuclearEnergy%20-%20SRA%20of%20the%20EERA-JPNM%20-%20web%20version.pdf

Property improvements required or underway¡Error! Marcador no definido.

- Optimize fabrication process to have a better control of O/M ratio = synthesis of $(U,Am)O_2$

with O/M < 2.00 with Am < 30 % - microstructure optimization (open vs close porosity and elemental distribution homogeneity) - Helium behaviour for various Am contents - Evaluation and investigation of cladding corrosion in case of interaction fuel-cladding - Thermal properties: thermal conductivity, Emissivity, melting temperature, heat capacity - Alpha self-irradiation effect			
Technology readiness level	3	**Likeliness of use**	?
Other observations			
- Am Bearing Blanket irradiations reported in literature: - MARIOS (HFR): $Am_{0.15}U_{0.85}O_{1.94}$ (dense 92.5% and open porosity 88%) [1,2,3] - DIAMINO (OSIRIS) : $Am_{0.15}U_{0.85}O_{1.94}$ (dense 95.7% and open porosity 82%) [4] - SUPERFACT 1 : $U_{0.60}Np_{0.20}Am_{0.20}$, $U_{0.586}Np_{0.19}Am_{0.19}O_{1.93}$ [5,6,7] - SPHERE (HFR): $(U_{0.9}Am_{0.1})O_{2-x}$ [8] - MARINE (HFR) : $U_{0.87}Am_{0.13}O_{1.935}$ & $U_{0.86}Am_{0.14}O_{1.93}$ [9] References: [1] E. D'Agata et al, MARIOS: Irradiation of UO2 containing 15% americium at well-defined temperature, Nuclear Engineering and Design. 242 (2012) 413–419. https://doi.org/10/ddgqb7. [2] E. D'Agata et al, The results of the irradiation experiment MARIOS on americium transmutation, Annals of Nuclear Energy. 62 (2013) 40–49. https://doi.org/10.1016/j.anucene.2013.05.043. [3] D. Prieur, et al., B. Philippe, Fabrication and characterisation of U0.85Am0.15O2−x discs for MARIOS irradiation program, Journal of Nuclear Materials. 414 (2011) 503–507. https://doi.org/10.1016/j.jnucmat.2011.05.036. [4] S. Bejaoui et al., Description and thermal simulation of the DIAMINO irradiation experiment of transmutation fuel in the OSIRIS reactor, Progress in Nuclear Energy. 113 (2019) 28–44. https://doi.org/10/gm8wdk. [5] C. Prunier et al., Some specific aspects of homogeneous Am and Np based fuels transmutation through the outcomes of the Superfact experiment in Phenix fast reactor, in: Global '93: International Conference and Technology Exhibition, Seattle, 1993. [6] J. Babelot, N. Chauvin, "Joint CEA/JRC Synthesis Report of the Experiment SUPERFACT 1", Report JRC-ITU-TN-99/03, 1999 [7] L. Luzzi et al., Assessment of three European fuel performance codes against the SUPERFACT-1 fast reactor irradiation experiment, Nuclear Engineering and Technology. 53 (2021) 3367–3378. https://doi.org/10/gm82d8. [8] A. Gallais-During et al., Outcomes of the PELGRIMM project on Am-bearing fuel in pelletized and sphere-pac forms, Journal of Nuclear Materials. 512 (2018) 214–226. https://doi.org/10/gm82dp. [9] E. D'Agata et al., The MARINE experiment: Irradiation of sphere-pac fuel and pellets of UO2−x for americium breading blanket concept, Nuclear Engineering and Design. 311 (2017) 131–141. https://doi.org/10/f9nsqh.			
List of authors of the ID card			
Philippe MARTIN (CEA), Thierry WISS (JRC)			

5. Concrete

Coordinator: Miguel Ferreira, VTT

Concrete is a heterogeneous material composed of cement binder, fine aggregates (sand) and coarse aggregates mixed with water which hardens with time. There is an extremely large variety of compositions depending on the types of cement and aggregates, as well as their proportions. Furthermore, certain admixtures can be added to the mixing process that enhance certain fresh and/or hardened concrete properties (e.g. plasticizer for workability in the fresh state; air entrainment for resistance of hardened concrete in freezing environments).

Reinforced concrete structures in NPPs are composed of several constituents, including concrete, conventional steel reinforcement, pre-stressed steel, steel liner plates, and structural steel. While unique in application, they share many physical characteristics with conventional concrete structures. Experience shows that ageing degradation of reinforced concrete structures can be a result of exposure to aggressive environments, excessive structural loads, accidental conditions, use of unsuitable materials, poor material and construction quality, and the lack of or inadequate maintenance. As concrete ages, changes in its properties will occur naturally as a result of continuous microstructural changes (e.g., due to cement hydration, crystallization of amorphous constituents, reactions between cement pore solution and aggregates, etc.), as well as environmental interaction leading to adverse performance of the cement paste matrix and aggregates under physical or chemical attack (e.g., internal expansion, cracking, leaching etc.).

The cards in this section detail for each ageing or degradation mechanism affecting concrete the practical consequences of this mechanism and the R&D efforts made and needed on materials aspects monitoring and structural modelling.

5.1 Temperature effects on delayed deformation and damage

Concrete structures types	Importance level			
	Insignificant	Slight	Medium	High
Containment with metallic liner				X
Containment without metallic liner				X
Cooling towers			X	
Spent fuel pools	X			
Water intake/outtake structures	X			
Concrete pipe	X			
radioactive waste package				X
Structural anchorages	X			
Foundations, piles and underground structures	X			
Pedestal of reactor vessel		X		

Practical consequences for Long Term Operation

At early age, containment buildings and radioactive packages experience restrained shrinkage or thermal expansion which can lead to cracking or initiate it. In the case of containment buildings, prestressing tendons are put after concrete hardening. Early age cracks are closed under the compressive effects of prestressing tendons. During ageing phase, containment building experience drying, shrinkage and creep at moderate temperature too (40°C in intrados). Delayed strain (shrinkage and creep) induces prestress loss and leakage increasing (when there are no metallic liner) in containment building. The loss of prestressing force increases at elevated temperatures due to increases in the relaxation of steel and volumetric changes of concrete (i.e. creep and shrinkage).

Temperature has significant effect on creep. Creep and shrinkage are accelerated when temperature is increased, both if drying is prevented (basic creep) and also if drying is allowed (desiccation shrinkage and desiccation creep). In the case where drying is allowed, a significant part of this kinetics increase is due to the acceleration of drying at higher temperature. When heating is applied shortly after loading, additional strains occur compared to the case where heating is applied and equilibrated before loading. These additional strains are called thermal transient strains.

For moderate temperature, efficient and well validated empirical models for finite element computations are available. These models, however, need to be calibrated on laboratory data (which is almost never available for existing containment buildings), on monitoring data (which is often available for containment buildings), or on predictions of the material behaviour of concrete (for which some models exist but are insufficiently validated).

For accidental phases and particular operational situations involving exposition of concrete to temperatures up to 150°C, experimental data lacks and the material modelling is less mature.

References:

- D. Jacques et al., Overview of state-of-the-art knowledge for the quantitative assessment of the ageing/ deterioration of concrete in nuclear power plant systems, structures and components. Deliverable D1.1 H2020 ACES Project (2021)
- IAEA Nuclear Energy Series, Ageing Management of Concrete Structures in Nuclear Power Plants, No. NP-T-3.5
- Z. P. Bažant et M. Jirásek, Creep and hygrothermal effects in concrete structures, Dordrecht, The Netherlands: Springer, 2018.
- Laurent Granger, Comportement différé du béton dans les enceintes de centrales nucléaires : Analyse et modélisation (in French), PhD Thesis ENPC, 1995
- A. Neville, W. Dilger et J. Brooks, Creep of plain and structural concrete, London: Construction press, Longman group, 1983.

- H. Cagnon, T. Vidal, A. Sellier et J. M. Torrenti, «Transient thermal creep at moderate temperature,» *Key Engineering Materials,* vol. 711, 2016. - CANADIAN STANDARDS ASSOCIATION, Concrete Materials and Methods of Concrete Construction/Test Methods and Standard Practices for Concrete, CSA A23.1-09/A23.2-09, CSA, Toronto (2009). - ANDERSON, P., Thirty Years of Measured Prestress at Swedish Nuclear Reactor Containments, Nucl. Eng. Des. 235 (2005). - LUNDQVIST, P., Nuclear Reactor Containments: Evaluation of Prestress Losses and Prediction Models, 3rd FIB Int. Congress Inc. the PCI Annual Conv. and Bridge Conf., Precast Prestressed Concrete Institute, Washington, DC (2010).
R&D focusing on materials aspects (modelling, characterization, novel binders and mix designs)
- Development of micromechanical models for prediction of concrete properties (drying, shrinkage, creep and damage) from concrete mix proportions at moderate temperature to overcome the lack of data on real concrete containment buildings for phenomenological models calibration - Experimental investigation of concrete behaviour (drying, shrinkage, creep and damage) at elevated temperature (up to 150°C) - Development of calibrated and validated phenomenological models for concrete behaviour (drying, shrinkage, creep and damage) at elevated temperature (up to 150°C) - Experimental investigation at structural scale of massive structure under temperature and pressure effects (150°C and 5.2 bars for Containment building and 50°C/30% for radioactive waste package)
R&D focusing on Non-destructive testing, monitoring, structural assessment
- Development of distributed extensometry techniques for temperature and strain measurement - Development of water content sensor for structural application - Development of Non-destructive testing **(**NDT) for evaluation of prestressing in tendons injected with cement grout and / or in concrete - Development of digital image correlation technique for large structure application
R&D focusing on structural modelling
- Development of modelling approach for damage of massive structure at early age for getting initial stress state and initial crack - Development of modelling approach for damage of structure in accidental situation - Development of modelling approach for air-vapour leakage of containment building in accidental situation - Development of digital twin approach for Ageing Management and optimization of Long Term Operation, to ensure that the concrete component withstand and to estimate air-vapour leakage in severe accident situation
Other observations
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List of authors of the ID card
Jean-Luc ADIA (EDF), Charles TOULEMONDE (EDF), Nhu-Cuong TRAN (EDF)

5.2 Corrosion

Concrete structures types	Risk level			
	Insignificant	Slight risk	Medium risk	High risk
Containment with metallic liner				X
Containment without metallic liner	X			
Cooling towers				X
Spent fuel pools			X	
Water intake/outtake structures			X	
Concrete pipe				X
radioactive waste package			X	
Structural anchorages			X	
Foundations, piles and underground structures		X		
Pedestal of reactor vessel		X		

Practical consequences for Long Term Operation

Corrosion is the main degradation mechanism for concrete components within NPPs (containment structure, cooling towers, concrete pipe). Concrete has limited capacity for plastic deformation and absorption of mechanical energy. Reinforcing bar (rebar) is typically installed in locations where tensile stresses are anticipated to address this. Fortunately, rebar and concrete are mutually compatible. They have similar coefficients of thermal expansion, and the relatively high pH of the concrete pore water (pH~12.5–13.6) contributes to the formation of an oxide film that passivates steel against corrosion. In some cases (containments or pipes for instance), metallic liner is embedded in concrete to enhance the leak tightness function of the structure. Disruption of the passive film can, however, occur due to leaching of alkaline substances by water or carbonation (reduction of pH) or chloride intrusion (destabilization of passive layer).

When metallic iron is transformed into ferric oxide (rust) by corrosion, its volume increases and can initiate cover concrete cracking. Because corrosion is fairly uniform, such cracking usually occurs prior to a particular structural cross-section becoming excessively weak, thus giving a visual warning of deterioration. Occasionally, however, cover spalling occurs before any surface signs of deterioration are visible. Structural strength and serviceability are only reduced and jeopardized when rebar corrosion causes a significant loss of steel cross-section, or there occurs a loss of bond between steel and concrete. For the particular case of liner plates, localized corrosion leading to a loss of leak tightness is of most concern.

References:
- IAEA Nuclear Energy Series, Ageing Management of Concrete Structures in Nuclear Power Plants, No. NP-T-3.5
- ERLIN, B., VERBECK, G.J., Corrosion of Metals in Concrete – Needed Research, Corrosion of Metals in Concrete, SP-49, American Concrete Institute, Farmington Hills, MI (1975).
- MEHTA, P.K., GERWICK, J.B.C., Cracking-corrosion interaction in concrete exposed to marine environment, Concr. Int. 4 10 (1982) 45–51.
- BUILDING RESEARCH STATION, The durability of steel in concrete: Part 2 – Diagnosis and assessment of corrosion-cracked concrete, Build. Res. Establishment Dig. 26 (1982).
- RASHEEDUZZAFAR, AL-SAADOUN, S.S., AL-GAHTANI, A.S., Corrosion cracking in relation to bar diameter, cover, and concrete quality, J. Mater. Civ. Eng 4 4 (1992) 327–341.

R&D focusing on materials aspects (modelling, characterization, novel binders and mix designs)

In the case of corrosion by carbonation, it is very important to know when the carbonation front has reached the reinforcement bed or the liner, how much time is available before the corrosion begins. Experimental work and models allowing one to bring elements are desired.

Further understanding is required with regard to which materials may play a role in mitigating

this corrosion, for both application to maintain performance of existing structures as well as application to ensure performance in new or planned structures (e.g., chemical treatments vs the influence of new concrete raw materials).
R&D focusing on Non-destructive testing, monitoring, structural assessment
- Improved monitoring techniques (e.g., drone image analysis) are needed to facilitate acquisition of adequate field data to support material and modelling aspects. - Improved non-destructive testing (NDT) techniques are needed to facilitate correlation between laboratory results (e.g., particularly accelerated testing) and actual material performance, particularly for processes such as corrosion where damage may not be readily evident from inspections of the external surface alone.
R&D focusing on structural modelling
- Improved models for each type of corrosion (e.g., crack, crevice, etc.) are required to improve the link between fundamental understanding of corrosion mechanisms and structural performance. - Improved models for concrete to incorporate the impacts of corrosion and coupling with other processes for quantitative, holistic damage assessments (e.g. impact of corrosion on leak tightness of containment)
Other observations
List of authors of the ID card
Jean-Luc ADIA (EDF), Charles TOULEMONDE (EDF), Nhu-Cuong TRAN (EDF), Tandré OEY (VTT)

5.3 Endogenous reactions

Concrete structures types	**Risk level**			
	Insignificant	**Slight risk**	**Medium risk**	**High risk**
Containment with metallic liner			X	
Containment without metallic liner			X	
Cooling towers		X		
Spent fuel pools	X			
Water intake/outtake structures	X			
Concrete pipe		X		
radioactive waste package	X			
Structural anchorages	X			
Foundations, piles and underground structures				X
Pedestal of reactor vessel			X	

Practical consequences for Long Term Operation

Delayed Ettringite Formation (DEF) is a special case of internal sulphate attack. It occurs when internal or external sulphates react with anhydrous or hydrated calcium aluminates and has an expansive character. Such an expansion is not a concern in fresh concrete but can induce cracking in hardened concrete and also increase the risk of secondary forms of deterioration such as reinforcement corrosion. DEF is a result of high early temperatures (above 70°C) in concrete which prevents the normal formation of ettringite (a normal product of early cement hydration) or the decomposition of ettringite that has been already formed. If structures susceptible to DEF are later exposed to water, ettringite can be formed again in the cement paste, causing expansive forces that result in cracking. Elevated temperatures also increase potential for damage due to DEF. DEF process leads to degradation of mechanical properties such as compressive strength and can promote increased permeability.

Some siliceous and dolomitic aggregates can react with the alkali hydroxides in concrete, causing expansion and cracking over a period of many years. This alkali-aggregate reaction has two forms: Alkali-Silica Reaction (ASR) and Alkali-Carbonate Reaction (ACR). In an ASR, alkali ions form a calcium alkali-silicate gel. This gel takes up pore solution water due to the attractive forces between polar water molecules and alkali-silicate gel. ASRs have been observed in NPP concrete structures in Belgium, Canada, USA and Japan.

References

IAEA Nuclear Energy Series, Ageing Management of Concrete Structures in Nuclear Power Plants, No. NP-T-3.5

R&D focusing on materials aspects (modelling, characterization, novel binders and mix designs)

- Extension of basic understanding of each process to conditions relevant to nuclear materials is needed, especially in the case of Alkali Aggregate Reactions (AAR) - due to prolonged radiation exposure aggregate susceptibility to AAR might increase.
- Adaptation of current understandings of cement and concrete chemistry to new raw materials, especially novel binders other than traditional cement, is required to both improve the sustainability of the nuclear structural materials and take advantage of the notable benefits of such alternate materials (e.g., reduced permeability, potentially improved stability under irradiation, etc.).
- Development of data base on DEF and ASR at room and high temperature
- Development of micromechanical models to understanding damage due to expansion induced by DEF or ASR

- Development of calibrated and validated phenomenological models for DEF or ASR kinetics and damage induced to feed structural application
R&D focusing on Non-destructive testing, monitoring, structural assessment
- Improved NDT monitoring, for a wide range of concrete properties, especially length change and water content, is needed to support both research on material/structural aspects and modelling. - Improved frequency of monitoring, and/or use of automated (drone) monitoring paired with machine learning for image analysis, is needed to improve relevance of laboratory and accelerated tests for field performance, particularly in the case of deterioration mechanisms that are evidenced by visible surface cracking (e.g., alkali-aggregate reaction). - Development of distributed extensometry techniques for temperature and strain measurement - Development of NDT for evaluation of prestressing in tendons injected with cement grout and / or in concrete - Development of digital image correlation technique for large structure application
R&D focusing on structural modelling
- Development of modelling approaches for damage induced by DEF or ASR are required, to establish the links between chemical effects and mechanical deterioration. - Extension of basic models to structural scale, both in terms of environmental and loading effects, is required to accurately model such processes in the nuclear context. - Extension of basic models to couple with concurrent deterioration processes, at the necessary scales, are also required. - Synergy with monitoring and material aspects will be of the utmost importance to achieve practically relevant results.
Other observations
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List of authors of the ID card
Jean-Luc ADIA (EDF), Charles TOULEMONDE (EDF), Nhu-Cuong TRAN (EDF), Tandré OEY (VTT)

5.4 Irradiation

Concrete structures types	**Importance level**			
	insignificant	**Slight**	**Medium**	**High**
Containment with metallic liner		X		
Containment without metallic liner		X		
Cooling towers	X			
Spent fuel pools	X			
Water intake/outtake structures	X			
Concrete pipe	X			
Radioactive waste package				X
Structural anchorages	X			
Foundations, piles and underground structures	X			
Pedestal of reactor vessel				X

Practical consequences for Long Term Operation

Changes in concrete properties due to irradiation appear to depend primarily on the aggregates used, the volume of which is affected by irradiation. Fast neutrons are mainly responsible for expansion caused by atomic displacements in certain aggregates. Quartz aggregates containing crystals with covalent bonds are more affected by radiation than calcareous aggregates containing crystals with ionic bonds. When nuclear radiation is attenuated or absorbed in the concrete, almost all absorbed radiation is converted into heat, impacting concrete physical, mechanical and nuclear properties. Irradiation effects on concrete are:

- Decrease in tensile and compressive strengths and modulus of elasticity,
- Resistance of concrete to neutron radiation depends on mix proportions, type of cement and type of aggregate,
- Deterioration of concrete properties associated with a temperature rise resulting from irradiation is relatively minor,
- Irradiated concrete's coefficients of thermal expansion and conductivity differ little from those of temperature exposed concrete.
- When exposed to neutron irradiation, concrete's elasticity modulus decreases with increasing neutron fluence.
- Concrete creep is not affected by low level radiation exposure, but for high levels of exposure, creep would probably increase because of the effects of irradiation on tensile and compressive strength.
- Gamma rays produce radiolysis of water in cement paste that can affect concrete's creep and shrinkage behaviour to a limited extend and also result in the evolution of gas.

References:

IAEA Nuclear Energy Series, Ageing Management of Concrete Structures in Nuclear Power Plants, No. NP-T-3.5

R&D focusing on materials aspects (modelling, characterization, novel binders and mix designs)

- Refinement of micromechanical models for understanding aggregate expansion induced by irradiation effects on concrete is needed to adequately incorporate these processes in basic design of nuclear concretes.
- Improved understanding is needed of the effects of both relatively lower dose yet longer term radiation exposures to extend fundamental understanding of radiation interactions to the meso- and macro-scale impacts they have on both chemical and physical degradation processes.
- Improved data management is needed to create a standardized, easily accessible database of relevant concrete and environmental data, to address shortfalls in such data available to

modelling efforts, particularly with regard to the evolving state of knowledge surrounding low-dose radiation exposure, gamma radiation, and sustained exposure in waste packages.
R&D focusing on Non-destructive testing, monitoring, structural assessment
- Improved NDT monitoring, for a wide range of concrete properties, especially length change and water content, is needed to support both research on material aspects and modelling. - Improved frequency of monitoring, and/or use of automated (drone) monitoring paired with machine learning for image analysis, is needed to improve relevance of laboratory and accelerated tests to field performance, particularly in the case of deterioration mechanisms that are evidenced by visible surface cracking (e.g., alkali-aggregate reaction).Development of digital image correlation technique for large structure application
R&D focusing on structural modelling
- Development of a modelling approach to incorporate the wide range of relevant irradiation-induced deterioration mechanisms for massive structures is needed (e.g., thermal stress, chemical alteration, etc.). - Development of models for the changing penetration rate of radiation in the deteriorating concrete is needed to adequately capture the time-dependence of deterioration under relevant field conditions.
Other observations
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List of authors of the ID card
Jean-Luc ADIA (EDF), Charles TOULEMONDE (EDF), Nhu-Cuong TRAN (EDF), Tandré OEY (VTT)

5.5 Reactive transport processes in concrete

Concrete structures types	**Importance level**			
	Insignificant	**Slight**	**Medium**	**High**
Containment with metallic liner	X			
Containment without metallic liner	X			
Cooling towers			X	
Spent fuel pools	X			
Water intake/outtake structures	X			
Concrete pipe	X			
radioactive waste package				X
Structural anchorages	X			
Foundations, piles and underground structures			X	
Pedestal of reactor vessel	X			

Practical consequences for Long Term Operation

The majority of the mechanisms affecting concrete durability involve reactive transport phenomena, i.e. they involve the transport of chemical species and their reaction with the hydrated phases of the cement. The main consequences are mineralogical and microstructural modifications that can induce deformations and therefore cracking which in turn can impact on transport. Among the most usually encountered degradation processes are:

- Carbonation: reaction between atmospheric CO_2 and the Ca ions leading to the precipitation of $CaCO_3$ and dissolution of Ca-bearing phases. In addition to rebar corrosion, carbonation induces significant changes in mineralogy and microstructure and generates shrinkage and cracking
- Ca-leaching: when in contact with water, Ca is washed away and leached from the concrete surface. This induces the dissolution of Ca-bearing phases (portlandite, C-S-H). The main consequences are increase in porosity, coarsening of the pore-structure, significant strength loss and eventually decalcification shrinkage and cracking
- Multi-ionic attack: when the leaching water contains ions, these can react with the hydrates to impact on the mineralogy and microstructure. For example, contact with water containing chlorine and magnesium can form chlorinated phases (Kuzel's/Friedel's salt, Ca oxychloride) or M-S-H (whose impact on strains and strength remains an open issue)
- External sulphate attack: Similar to DEF, external sulphate attack corresponds to the reaction between sulphur (S) and alumina to precipitate expansive phases but S comes from the environment (usually from the ground or underground water)

These processes have some common features:

- They all involve chemical reactions/processes that occur at the nanoscale
- They are very much influenced by the amount of water that is present in the pores: water conditions the transport properties (and then the rate at which the species are transported) and water is the reaction medium (chemical reactions can only occur in water). This stress the need of accurate descriptions for water transport and accurate descriptions of the effect of saturation on transport and chemical reactivity.
- In most cases transport is the step limiting the process (the degradation rate is imposed by transport); the description of the link between mineralogy, microstructure and transport is then of first importance.
- Similarly, the description of cracking induced by the considered process as well as the consequences on the transport properties are also of paramount importance.

References:

- Kangni-Foli et al. (2021) "Carbonation of model cement pastes: the mineralogical origin of microstructural changes and shrinkage" Cem. Concr. Res. 144, 106446
R&D focusing on materials aspects (modelling, characterization, novel binders and mix designs)
- Need of approaches/models that describe the mechanisms at the nanoscale + Need for multiscale approaches/models to evaluate the consequences of the nanoscale mechanisms at the mesoscale (material properties) - Use of machine learning (ML) to assess the durability properties of the materials based on their composition and/or propose the most suitable concrete composition for the considered degradation scenario (need for exhaustive compilations of experimental data)
R&D focusing on Non-destructive testing, monitoring, structural assessment
- Development of durability-oriented embedded instrumentation/sensors and NDT (…)
R&D focusing on structural modelling
- Need for multi-scale approaches/models to compute the structural response induced by the nanoscale mechanisms and/or the materials durability properties (accounting for the retroaction of the mechanical loading/response of the structure on the material properties and/or on the nanoscale mechanisms) - Use of ML techniques for structural computations
Other observations
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List of authors of the ID card
Stéphane POYET (CEA)

5.6 Voids and defects in steel-concrete structures

Concrete structures types	Importance level			
	Insignificant	Slight	Medium	High
Containment with metallic liner			X	
Containment without metallic liner			X	
Cooling towers			X	
Spent fuel pools			X	
Water intake/outtake structures			X	
Concrete pipe			X	
radioactive waste package			X	
Structural anchorages			X	
Foundations, piles and underground structures			X	
Pedestal of reactor vessel			X	

Practical consequences for Long Term Operation

In some cases, concreting may be difficult to achieve and/or to verify properly. Fresh concrete flow may be hindered by the presence of rebars or studs (for steel-concrete) and generate voids, holes, honeycombs and other defects that can impact the concrete properties and affect the structure usability, performance and durability. The quality of concrete placing may also be difficult to control because of the concrete thickness and/or due to the presence steel liner/forms that prevent access to the concrete (steel-concrete). The case of steel-concrete is a good example of what should be required:

- Concrete formulation: flowable but stable fresh concrete to fill all the volume between the steel plates
- Because concrete cannot be observed directly: need of durability-oriented instrumentation (concrete drying, temperature, carbonation, steel corrosion…) and NDT for structural health monitoring
- Models and specific approaches to describe the local interaction between studs and concrete and suitable models for structural computation.

References:

- Burgan, B., Hoang Tung, V., Chryssanthopoulos, M., et al., SC for industrial, energy and nuclear construction efficiency (SCIENCE) : final report, Publications Office, 2018, https://data.europa.eu/doi/10.2777/248496

R&D focusing on materials aspects (modelling, characterization, novel binders and mix designs)

- For novel binders - adapt mix design and concrete practice for suitable concrete properties at the fresh and hardened and suitable properties
- Need for scale-one mock-ups

R&D focusing on Non-destructive testing, monitoring, structural assessment

- Development of embedded instrumentation/sensors for long-term durability monitoring (deformation, drying, cracking, carbonation, and steel corrosion…)
- Development of NDTs for defects localization and/or structural health monitoring (inputs for digital twin approaches…)

R&D focusing on structural modelling

- Development of reliable & cost-effective models/tools to describe the mechanical behaviour of steel-concrete
- Development of digital twin approach for Ageing Management and optimization of Long-Term Operation
- Use of ML techniques for structural computations (surrogate models…)

Other observations
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List of authors of the ID card
Stéphane POYET (CEA), Eric PHILIP (CEA)

5.7 Microbial activity

Concrete structures types	Importance level			
	Insignificant	**Slight**	**Medium**	**High**
Containment with metallic liner			X	
Containment without metallic liner			X	
Cooling towers			X	
Spent fuel pools				
Water intake/outtake structures				
Concrete pipe			X	
radioactive waste package				X
Structural anchorages				
Foundations, piles and underground structures			X	
Pedestal of reactor vessel				

Practical consequences for Long Term Operation

Microorganisms (e.g. bacteria, fungi) are present everywhere in the environment and it has been known for long that their activity can have a significant impact on cementitious (and metallic) materials and can induce bio-deterioration and threaten the durability of materials and potentially impair structural integrity. The most usual degradation pathway is the production of mineral or organic acids that alter/degrade the concrete components and, like concrete leaching, generate strength loss, porosity increase, shrinkage and cracking… In the same way, some bacteria are able to oxidize iron and other metals and lead to iron bio-deterioration. Alternatively, microbial activity is also known to affect the redox potential and then modify the speciation and mobility of radionuclides within concretes.

Microbial activity is almost always possible although it requires some conditions related to the micro-organisms themselves (synergy between different organisms within biofilms for instance), the materials (porosity, chemical composition, pH, etc.) and the environment (temperature, water and nutrient supply, etc.) to be met.

Alternatively, the ability of micro-organisms to produce substances that react with cementitious materials may be an advantage: the activity of bacteria may lead to the precipitation of calcium carbonate and thus seal porosity or cracks in curative or preventive approaches.

References:

- Turick & Berry (2016) “Review of concrete biodegradation in relation to nuclear waste” Journal of Environmental Radioactivity 151, 12-21
- Zhang et al. (2019) “Use of genetically modified bacteria to repair cracks in concrete” Materials 12, 3912
- Van Tittelboom et al. (2010) “Use of bacteria to repair cracks in concrete” Cement & Concrete Research 40, 157-166

R&D focusing on materials aspects (modelling, characterization, novel binders and mix designs)

- Alternative binders for increased resistance to bio-deterioration (including supplementary cementing materials, geopolymers, alkali-activated materials…)
- THCBM – Thermo-Hydro-Chemo-Bio-Mechanics: understanding, description and modelling of the interaction(s) between heat and mass (including water) transport, chemical reactions, bacteria population growth and mechanics

R&D focusing on Non-destructive testing, monitoring, structural assessment

- Development of experimental setups that integrate the specify of the biological conditions (heterogeneous within the biofilms)
- Miniaturisation of sensors for monitoring the local chemical conditions (pH, redox for instance) and the local changes in chemical-physical properties.

R&D focusing on structural modelling

- Upscaling approaches (micro-macro) based on the description of the microbial activity (describing the population growth) and evaluation/description of the disorders at the structure/structural element scale
Other observations
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List of authors of the ID card
Alexandra BERTRON (INSA Toulouse), Stéphane POYET (CEA)

5.8 Concrete in LOCA conditions

Concrete structures types	Importance level			
	Insignificant	Slight	Medium	High
Containment with metallic liner		X		
Containment without metallic liner				X
Cooling towers	X			
Spent fuel pools	X			
Water intake/outtake structures	X			
Concrete pipe	X			
radioactive waste package	X			
Structural anchorages	X			
Foundations, piles and underground structures	X			
Pedestal of reactor vessel				X

Practical consequences for Long Term Operation

In accidental conditions, concrete might have to cope with very unusual conditions In case of a loss of the primary coolant (LOCA), the vaporization of water is expected to induce a significant increase in temperature, total pressure and relative humidity within the reactor containment building (approx. 150°C, 5-6 bar). These specific conditions are expected to induce water and heat transfer in the concrete wall and generate strains, stresses and possibly cracking. The description of these coupled phenomena remains difficult today due to the lack reliable and consolidated data (mainly concrete properties and associated mechanisms such as transient thermal strains or creep for instance) to feed models and numerical tools (that still remain to be improved and validated). In the absence of a metallic liner, the presence of cracks is expected to generate a pathway for fission products dissemination in the environment. However, water condensation within the cracks (due to temperature differences between hot vapour and temperate/cold concrete), crack tortuosity and constrictively are expected to promote the retention of RNs in the concrete (through physical and chemical processes) and then limit the dissemination.

In the event of a fuel meltdown (creation of corium), if the corium exits the reactor vessel, the concrete would come into contact with the liquid corium. In addition to the chemical interactions and associated degradation, the concrete would then be subjected to Thermo-Hydro- Mechanics loading which could generate cracking and promote the transport of liquid corium.

R&D focusing on materials aspects (modelling, characterization, novel binders and mix designs)

- Need of reliable and consolidated data and models to describe the materials properties and deformations in temperature (including cracking)
- Use of ML to assess the materials properties based on their composition and/or propose the most suitable concrete composition for the

R&D focusing on Non-destructive testing, monitoring, structural assessment

- Development of embedded instrumentation/sensors and NDT for cracking (detection, monitoring, quantification…
- Development of NDTs for cracks localization and/or structural health monitoring (inputs for digital twin approaches…)

R&D focusing on structural modelling

- Need for tests/mock-ups at meso/macro scales (from the centimetre to the structure)
- Use of machine learning techniques for structural computations (surrogate models…)

Other observations

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List of authors of the ID card
Stéphane POYET (CEA)

5.9 Seismic performance and consequence of impacts

Concrete structures types	**Importance level**			
	Insignificant	**Slight**	**Medium**	**High**
Containment with metallic liner			X	
Containment without metallic liner				X
Cooling towers			X	
Spent fuel pools	X			
Water intake/outtake structures	X			
Concrete pipe		X		
radioactive waste package				X
Structural anchorages				X
Foundations, piles and underground structures			X	
Pedestal of reactor vessel				X

Practical consequences for Long Term Operation

Traditionally, it is intended that NPP concrete structures respond in the range of small strains to design basis (DB) seismic loads. For impact loads, especially hard-missile impact, localised damage of the concrete is inevitable, but besides this local zone the structures are also assumed to respond in range of small strains. Hence, NPP structural studies have traditionally been focusing on the quasi-elastic ranges of responses of reinforced concrete elements.

Recently, there is a growing interest, partly triggered by the Fukushima accident, in design extension conditions (DEC), or beyond design basis events. The DEC earthquake scenarios defined by regulators are significantly more demanding compared to earlier DB scenarios. Fulfilling them requires exploiting more of the non-linear stress responses of material and components (e.g. shear walls). Computer modelling capabilities are also increasing significantly allowing for multiscale modelling, whereas information from micro scale models are utilized and integrated into larger scale models, including at structural scale. This sophistication, both in terms of geometry and material models, of modelling as tool for safety analysis for NPPs, creates opportunities, benefits, but also risks when implemented without appropriate calibration to empirical observations.

Another aspect that needs to be considered when LTO of concrete structures is under investigation is the effect of physical, chemical and seismic ageing on the mechanical properties of reinforced concrete. Recurrent small amplitude vibration combined with the corrosive effect of the environmental loads can result in a premature drop of concrete strength and stiffness properties. As a result, analyses done with linear-elastic material assumptions and virgin concrete material parameters may no longer be valid. Further theoretical investigations validated against benchmark experiments is necessary to assess the effect of concrete ageing on the mechanical behaviour of the structure.

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R&D focusing on materials aspects (modelling, characterization, novel binders and mix designs)

- Selection/development of multi-scale material modelling techniques for extended non-linear response ranges (i.e. cracking, damage etc.) of large-scale NPP specific concrete structures (e.g. shear walls); In particular, the topics of anisotropy of damaged concrete and the mechanisms of frictional dissipation in cracked concrete should be addressed.
- Collecting or developing material testing methods to supply multi-scale modelling with reliable input parameters at different scales. If novel binders, e.g. geopolymers or Alkali Activated Materials are considered, a thorough experimentally validated theoretical study of the mechanical properties has to be carried out.

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R&D focusing on Non-destructive testing, monitoring, structural assessment

Sophisticated concrete material models are reliable only if properly calibrated. Therefore, novel experimental methods for concrete mechanical parameter testing need to be considered. An important development topic is the use of soundwave velocity speed using the pitch-catch method to determine the stiffness properties of damaged concrete.

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R&D focusing on structural modelling

- A hot topic in structural modelling and simulation of impact loaded structures is the modelling of fragmentation. Standard finite element formulation needs to be extended in order to materialize macroscopic crack growth and fragmentation, which is necessary to correctly model the multibody simulation beyond initial failure. Other related topics include the correct modelling of reinforcement to concrete bond slip as well as contact of pre-stress cables against ducts in the neighbourhood of an impact. It also has to be acknowledged that solutions used for quasi-static simulations may not be effective in high-speed dynamic simulations.

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Other observations

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List of authors of the ID card

Stéphane POYET (CEA), Ludovic Fülöp (VTT), Alexis FEDEROFF (VTT)

5.10 Steel-concrete systems performance

Concrete structures types	**Importance level**			
	Insignificant	**Slight**	**Medium**	**High**
Containment with metallic liner				X
Containment without metallic liner		X		
Cooling towers		X		
Spent fuel pools				X
Water intake/outtake structures			X	
Concrete pipe		X		
radioactive waste package				X
Structural anchorages				X
Foundations, piles and underground structures			X	
Pedestal of reactor vessel			X	

Practical consequences for Long Term Operation

Steel-concrete (SC) is a structural typology and construction method in which continuous steel plates are used on the surfaces of concrete walls or slabs. The steel plates have the roles of both formwork and tensile reinforcement; they replace the longitudinal reinforcement. The continuous steel-plates on the sides of a construction element bring many benefits, achieving high levels of prefabrication and modularization leading to improved economy by shorter construction times; but it hinders the use of many traditional NDT techniques to assess the condition of concrete. SC elements are employed in the Westinghouse AP1000 plant design with units under construction, or entering operations, in China and the US. SC is also a potential answer for the needs, in terms of structural innovation, of SMRs. European standardization of SC design is ongoing in EN and AFCEN.
While the design solutions for future SC NPPs, are being worked out, there is need for lifetime performance estimation, efficient inspection solutions, monitoring techniques (especially NDT), and end-of-life scenario models.

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R&D focusing on materials aspects (modelling, characterization, novel binders and mix designs)

Develop modelling solutions for performance of SC in extreme loading conditions (especially cracking, delamination, post-linear response, etc.)

R&D focusing on Non-destructive testing, monitoring, structural assessment

- Development of standardization on specialized issue of performance (e.g. in-plane shear, high velocity missile impact, etc.)
- Selection and development of suitable NDT techniques for inspection and monitoring of CS structural systems.

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R&D focusing on structural modelling
Confirmatory modelling of large-scale CS structures, for model-to-measurements stress and strain distributions in connections.
Other observations
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List of authors of the ID card
Stéphane POYET (CEA), Ludovic Fülöp (VTT)

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