Nuclear Science
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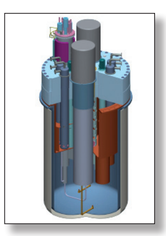
Status Report on Structural Materials for Advanced Nuclear Systems



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Nuclear Science

Status Report on Structural Materials for Advanced Nuclear Systems

NEA No. 6409

NUCLEAR ENERGY AGENCY
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Foreword

Under the guidance of the NEA Nuclear Science Committee (NSC) and under the mandate of the Working Party on Scientific Issues of the Fuel Cycle (WPFC), the objective of the Expert Group on Innovative Structural Materials (EGISM) is to conduct joint and comparative studies to support the development, selection and characterisation of innovative structural materials that can be implemented in advanced nuclear fuel cycles under extreme conditions such as high temperature, high dose rate, corrosive chemical environment and long service lifetime. The objectives of the expert group are: 1) to provide a state-of-the-art assessment of specific areas so as to identify priority areas of research; 2) to determine areas where experimental protocols and standards are needed and where the sharing of available experimental installations could be possible; 3) to identify existing databases and 4) to organise the next in a series of workshops on structural materials for innovative nuclear systems (SMINS).

This report summarises the status of innovative structural materials development in NEA member countries. It assesses specific elements in the identification of priority areas of research: system requirements for advanced reactors, the study of advanced materials to meet system requirements, the readiness level of each of the materials and novel material pathways that could significantly improve materials performance in advanced nuclear systems.

Acknowledgements

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1. Introduction to advanced nuclear systems

Since 2000, most of the advanced nuclear systems have been developing under the framework of the international research and development community, the Generation IV Initiative (GIF), which aims to develop future nuclear systems with high-safety features and economics, enhanced resource-use and a minimum level of waste production. Six systems have been selected for Generation IV consideration: gas-cooled fast reactor (GFR), very-high-temperature reactor (VHTR), sodium-cooled fast reactor (SFR), lead-cooled fast reactor (LFR), molten salt reactor (MSR) and supercritical-water-cooled reactor (SCWR) [1]. In addition to the GIF, another possible future system is the accelerator-driven system (ADS), which uses subcritical core with the spallation target mainly cooled by lead. This system could be deployed at double-strata nuclear fuel cycle as a dedicated burner [2].

1.1 Gas-cooled systems: VHTR and GFR

The thermal spectrum very-high-temperature reactor (VHTR) and the gas-cooled fast reactor (GFR) are two reactors operating with high-temperature helium [3]. Due to high-temperature functioning, these systems may attain high yields for energy production and supply high-temperature process heat that can be used for massive hydrogen production through a water decomposition process. However, the use of these high temperatures raises the question of selection of structural materials. VHTR technology is based on the design of previous high-temperature gas-cooled reactors, for which the primary in-core structures are made of graphite and carbon-based ceramic composites. The key out-of-core structures include the reactor pressure vessel (RPV, made of 2¼ Cr bainitic steel or 9Cr martensitic steel), the cross-vessel component and intermediate heat exchangers (IHXs) that will divert heat from the primary side of the reactor to a hydrogen production plant. Such components place great demands on their materials of construction. Thin sheets are used at up to 850 to 950°C and must sustain a differential pressure of 6 to 7 MPa during off-normal events. Therefore, mechanical strength, creep resistance and corrosion resistance are required for extended lifetime (approximately 20 years). Such temperatures demand the use of Ni-base alloys rich in chromium (about 22 wt%) and strengthened by additions of molybdenum, cobalt and tungsten. Examples include Inconel 617 (Ni-22Cr- 12Co-8Mo-0.1C) and Haynes 230 (Ni-22Cr-14W-2Mo-0.1C).

The GFR core structure is similar in function to a thermal spectrum VHTR, but the challenge to in-core structure in the GFR is unprecedented because fuel elements and other internal components will be exposed to very high temperatures in the range of 400 to 1 000°C and much higher radiation damage than in a thermal spectrum reactor. Moreover, moderator materials, such as graphite, must be minimised in the GFR. After a first phase of selection, fiber SiC/SiC composites were chosen as reference core inert materials. To address the intrinsic permeability of ceramic composites, CVD ceramic or metallic barriers having good chemical compatibility with SiC and U/PuC will be developed for application on the fuel assembly wall. Candidates for core internals are either based on the same composites or metallic materials (Alloy 800, austenitic steel) with special insulation. At this early design stage, materials for other structures, such as RPVs and out-of-core components, are expected to be the same as in a thermal VHTR.

The GFR reference design uses a direct-cycle helium turbine for electricity and can use process heat for thermochemical production of hydrogen. An alternate design also uses helium as the primary coolant, but utilises an indirect Brayton cycle, with supercritical carbon dioxide (S-CO_2) at 550°C and 20 MPa, for power conversion. A third optional design is a S-CO_2-cooled, direct Brayton cycle system. Because of the large temperature difference across any gas-cooled reactor core, materials will experience both low (~450°C) and high (~850°C) temperatures, while also accumulating much higher radiation damage than in a thermal spectrum reactor. Materials, such as graphite, that are traditionally used in thermal spectrum gas-cooled reactors must be minimised in the GFR, as moderators must be limited in fast spectrum systems. The core matrix and supports are the components inside the core barrel.

1.2 Sodium-cooled systems

Sodium-cooled fast breeder reactors (SFRs) can utilise uranium resources almost infinitely and many research and development (R&D) projects are ongoing. Japan and the Russian Federation are currently operating SFRs and many countries, including China, France, India and the Republic of Korea, are developing SFRs [3].

From the viewpoint of materials for core design, the main requirements for SFRs are:

- High burn-up cores are necessary to improve economy of the plant.
- A high-temperature core is also necessary to further improve efficiency of the plant.

Existing SFRs use austenitic stainless steels for fuel pins and wrapper tubes. However, to fulfil the above requirements, intensive research and development is occurring in many countries on ferritic-martensitic steels, particularly oxide dispersed strengthened (ODS) ferritic-martensitic steels.

From the viewpoint of materials for structural design of vessels and piping, etc., the main features of SFRs are:

- Normal maximum operation temperature is around 550°C, which is higher than that of light-water reactors (LWRs) and well in the creep regime of most materials.
- Internal pressure is almost negligible, contrary to LWRs. Main loading is from thermal transients and seismic events. Thermal transient loadings require evaluation of creep-fatigue.
- Service life is getting longer to improve economy and reduce environmental burdens. Sixty years is the most representative number for reactors currently being designed and material strength standards have to be prepared accordingly.

To fulfil these requirements, materials of prime interest are low-carbon nitrogen-added Type 316 stainless steels and 9-12 Cr ferritic steels.

1.3 Lead-cooled systems: LFR and ADS

The LFR systems are lead or lead-bismuth eutectic (LBE)-cooled reactors with a fast-neutron spectrum and closed fuel cycle. Multiple LFR designs have been proposed, including the Russian BREST system, the INEEL/MIT actinide burner and modular systems such as the secure transportable autonomous reactor (STAR) and the encapsulated nuclear heat source (EHNS). Options include a range of plant ratings, such as a long refueling interval transportable system ranging from 50 to 150 MWe, a modular system from 300 to 400 MWe and a large monolithic plant at 1 200 MWe. The designs can generally be classified into low temperature (<550°C average outlet temperature) or high temperature (>550°C and up to 800°C average outlet temperature). The high-temperature option provides for a broader range of energy products, including electricity, potable water and hydrogen. Lead bismuth has a lower melting point than lead and its use simplifies the ability to prevent primary freezing. For a large-scale deployment of nuclear energy, moving away from LBE towards lead reduces cost, primarily due to the elimination of expensive bismuth, and reduces radioactivity levels associated with polonium production from bismuth [4].

The design and development of accelerator-driven systems (ADS) were initiated with the objective of transmuting high-level nuclear waste (e.g. the minor actinides neptunium, americium and curium) and potentially reducing the burden on a final repository [5,6].

In the accelerator-driven system, a reactor with a subcritical core is coupled to a proton beam through a neutron spallation target. The neutron spallation target is normally placed in the middle of the subcritical core [5-7]. The proton beam hits the spallation material (as a reference material, the liquid lead-bismuth eutectic (LBE) has been selected) to generate neutrons. These generated neutrons sustain fission reactions in the subcritical core. The power of the proton beam/neutron spallation source is related to the design of the subcritical core (e.g. core power and k_{eff}).

Structural materials issues to be addressed and solved for ADS are related to both the neutron spallation target and the subcritical core reactor. Both components are in contact with heavy liquid metal (HLM) as LBE or lead. Key issues to be addressed in these environments are corrosion/oxidation resistance and degradation of mechanical properties of the structural and clad materials when in contact with flowing HLM.

In the neutron spallation target liquid, LBE is used as spallation material and cooling medium. The two main design options for the neutron spallation target are either with a beam window or windowless. Both options have advantages and disadvantages. For instance, the presence of a window will potentially allow confinement of LBE and spallation products that are generated. On the other hand, the window will experience extreme conditions due to proton/neutron irradiation (dpa, hydrogen and helium production), high temperature, thermal gradients and high-flow velocity of the liquid metal. Even if the burden of the structural materials of a windowless spallation target is less severe, items related to materials performance near the spallation zone need to be addressed.

One of the key R&D items to be addressed for the subcritical core is related to the uranium-free minor actinide bearing fuel and associated fuel cladding. The fuel pin/bundle is in critical need of a development and qualification programme. In particular, as far as the cladding material is concerned, one can consider that in addition to the materials issues typical for fast reactor cores, corrosion issues need to be taken into account. These issues usually have an important impact on materials selection and reactor design parameters, for example core temperature or thermal gradients.

1.4 Other systems: MSR and SCWR [4]

The MSR is the only Generation IV reactor concept that utilises liquid fuel made by dissolving uranium and thorium fuel in circulating molten salt. Molten fluoride salts emerge as primary candidate salts because of their wide range of solubility for uranium and thorium. These salts also exhibit many other attractive properties – such as low melting point; high boiling point; low vapour pressure; high thermal, chemical and radiation stability and optical transparency – and provide the ability to operate the reactor without pressurisation. In a MSR, the fuel salt flows through the reactor core, where fission occurs within the flowing salt and is then circulated through an IHX and back into the reactor core. The use of homogeneous liquid fuel obviates the need for solid fuel fabrication and the liquid fuel itself can be easily purified from fission products, at least compared to solid fuels.

The SCWR, essentially an advanced LWR, is designed for baseload electricity production with a high thermal efficiency (about 45% versus 33%) and potential for considerable plant simplification compared to the current fleet of commercial LWRs. The base design for the SCWR is a thermal spectrum reactor using conventional LWR-type oxide fuel and employing water rods as a moderator. The SCWR will operate above the critical point of water (374°C, 22.1 MPa); therefore the coolant does not undergo a phase change while passing through the core. The inlet temperature of the reference SCWR is ~280°C, while the outlet temperature is 620°C. The operating pressure of the reactor is 25 MPa. The combination of radiation, high temperatures, pressures and a rather aggressive chemical environment makes the SCWR one of the more challenging reactors from the standpoint of materials selection.

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2. Gas-cooled systems: GFR and VHTR

2.1 France

2.1.1 Introduction

In January 2006, the French President requested design of a Generation IV system that could be in operation by 2020. In June 2006, the French parliament passed a law requiring the options of future nuclear systems to be studied by 2012 and a prototype to be in operation by 2020. Finally, in December 2006, an inter-departmental committee agreed on a technical roadmap for SFR, GFR and fuel cycle studies leading to gathering data to choose future options in 2012. This strategy is based on the necessity to save uranium and reduce ultimate waste in the future. Concerning the GFR development plan, a priority for R&D was identified to support current judgements of pre-feasibility, as well as to update the baseline concept by 2012 with innovative design features. The 2012 milestone is essential, as it marks the end of the GFR viability phase with issuance of a final report containing decisions about detailed studies for construction of "Allegro", a 50 to 100 MWth experimental GFR.

The main requirements on the reactor materials to be used are the following [1-4]:

- 60-year extended lifetime, requiring development of prediction laws and thus knowledge of the various phenomena of material damage;
- fast neutron damage (fuel and core materials):
  - effect of irradiation on microstructure, phase instability, and precipitation;
  - swelling, growth, hardening, embrittlement;
  - effect on tensile properties (yield strength, ultimate tensile stress, elongation, etc.);
  - irradiation creep and creep rupture properties;
  - hydrogen and helium embrittlement.
- high-temperature resistance (VHTR > 850 to 950°C):
  - effect on tensile properties (yield strength, UTS, elongation);
  - high-temperature embrittlement;
  - effect on creep rupture properties;
  - creep-fatigue interaction;
  - fracture toughness;
- corrosion resistance: primary coolant, power conversion, H_2 production.

Other criteria for the materials are their availability, costs to fabricate and assemble, the possibility of in-service inspection (use for example of non-destructive examination techniques) and their composition, which should be optimised to present low-activation (or rapid deactivation) features that facilitate maintenance and disposal.

2.1.2 Generic components of GFR and VHTR

For some components, the materials developments are common for VHTR and GFR systems. Table 1 synthesises the common components and their conditions. Figures 1 and 2 provide images of the locations of components described.

Table 1: Common components and conditions of the GFR and the VHTR

Component	Normal temperature (°C)	Abnormal temperature	Pressure	Fluence ($n.cm^{-2}$) (E > 0.1 MeV)
RPV	400 to 500°C	650 °C for 50h	5 to 8 MPa	$3x10^{18}$
Operating control rod	400 to 1100°C	Up to 1500 °C within 100h	5 to 8 MPa	$4.3x10^{21}$
IHX	Primary: 450 to 1000°C Secondary: 400 to 950°C	-100°C/min -0.4 MPa/min	primary: 5 to 8 MPa secondary: 5 MPa	0

Figure 1: The ANTARES project – HTR (AREVA NP)

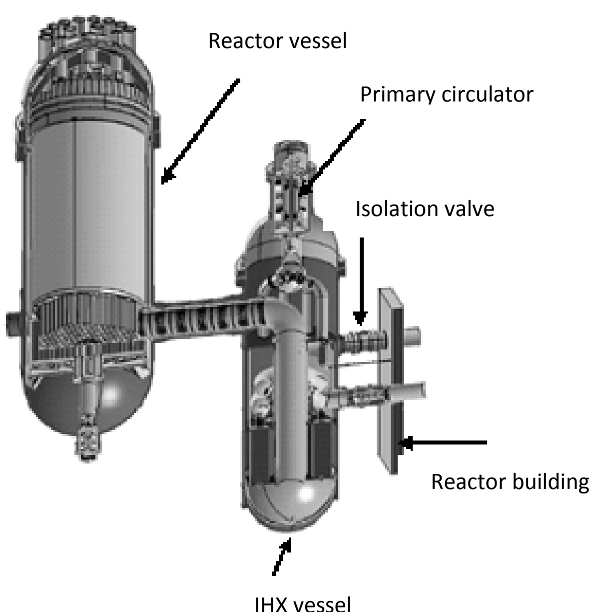
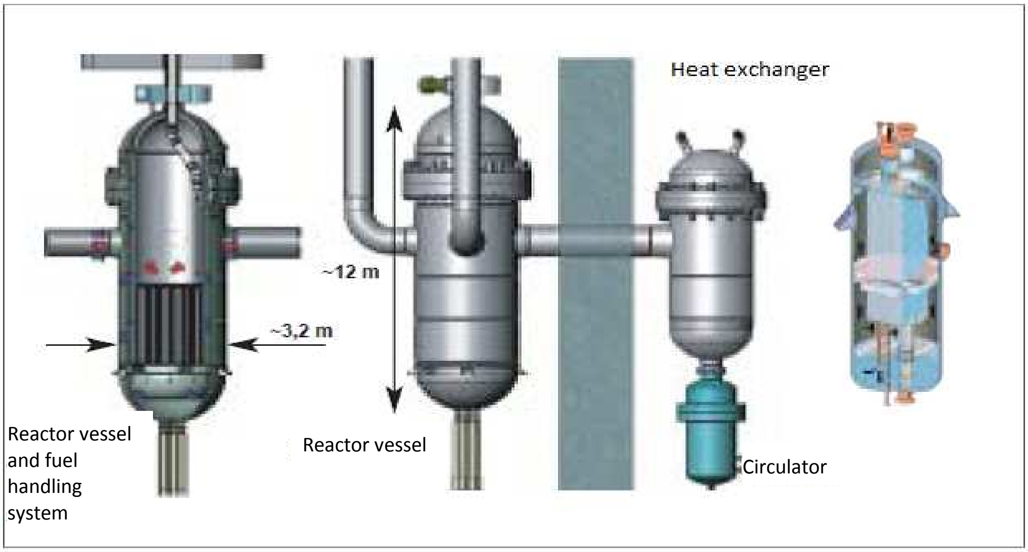


Figure 2: Gas-cooled fast reactor: The ALLEGRO primary system overview [7]



2.1.3 Pressure vessel of gas-cooled reactors

Compared to a pressurised water reactor (PWR), the main differences are the following:

- high-normal/off-normal service temperatures:
  - up to 450 to 500°C at 5 to 8 MPa;
  - up to $1x10^{19}$ $n.cm^{-2}$ fluence;
- very large vessel size requires the scale-up of ring forging and on-site joining;
- irradiation resistance has to be demonstrated for licensing.

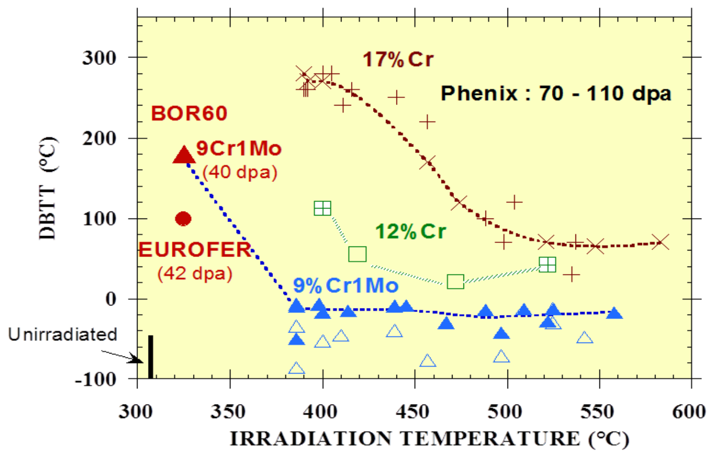
Requirements on the materials are thus:

- mechanical strength at low temperature and up to 500°C;
- creep properties in normal operating conditions: negligible creep regime, need for a creep-fatigue interaction diagram;
- chemical compatibility with impure helium: no materials degradation by corrosion should occur for 60 operating years;
- irradiation effects: no irradiation-induced embrittlement for 60 years;
- industrial feasibility and qualification of large forged rings and weldments.

In these conditions, 9Cr steels are promising candidate materials for the RPV of high-temperature gas-cooled reactors. In fact, the chemical composition of these materials has been progressively optimised by adding elements, allowing the formation of carbides (notably T91, 9Cr-1Mo VNb or T92, and 9Cr-0.5Mo WVNb). These materials have a martensitic structure. The metallurgy of these steels allows a good compromise between creep properties and toughness. Indeed, these steels have an excellent irradiation behaviour, the microstructure of the steels remains stable and they exhibit toughness in typical irradiation conditions of rapid reactors: 400 to 550°C and up to 100 dpa for durations of a few years. In these conditions the ductile-brittle transition temperature of

9Cr1Mo steels is nearly not affected by irradiation, compared to 12Cr steels or 9Cr-2Mo-VNb ferritic-martensitic (F/M) steels (Figure 3) [5].

Figure 3: Ductile to brittle transition temperature for F/M steels with different Cr contents as a function of irradiation temperature [5]



At higher and lower temperatures, the results on these steels show great stability under thermal ageing, with uncertainties remaining for very long durations (up to 60 years).

In the low-temperature range (below 450°C), the main in-service issue remains the effect of the α/α' unmixing on the mechanical properties. Preliminary results under mixed and fast neutron spectrum show that the α/α' demixing should allow this type of material to keep reasonable ductility and fracture toughness.

For temperatures above 500°C, in addition to coalescence of precipitates and matrix restoration that lead to softening of the material, precipitation of intermetallic phases occurs, preferentially localised at the different interfaces of the tempered martensite structure. The precipitation phenomena, mainly due to molybdenum (and/or tungsten), have, however, limited consequences on the residual ductility/toughness of the aged material as long as the molybdenum (or tungsten) concentration remains below 0.5 to 1 wt%.

All these positive aspects have to be moderated by the fact that few data are available for very long durations and that one should be very careful with the simple extrapolation of short-duration results. If 9Cr steel should be chosen for the RPV, a specific surveillance programme should be established. For all aspects, the 60-year lifetime of the reactor requires thorough knowledge of the evolution of the material submitted to the various constraints (irradiation, thermal, mechanical, environmental) and modelling of its behaviour to guarantee stability over the reactor life.

Finally, the elaboration of large components of 9Cr martensitic steels remains to be validated and research is being performed on:

- Large-dimension 9Cr steel elaboration: evaluation of microstructure heterogeneity, distribution of properties through the thickness of the material and their evolution under thermal ageing. Creep properties will have to be carefully studied.
- Assembling processes: welding processes have to be developed for thick materials and the welding parameters must be optimised to suppress hot cracking.
- Evaluation of in-service functioning, for both normal and abnormal conditions, requires the control of thermal ageing, notably for very long durations. This last point is essential, as temperatures encountered in the vessel and its 60-year lifetime could lead to mechanical properties evolution, notably creep properties.

2.1.4 Control rods

Material conditions for the control rods are summarised in Table 1. For this application, ceramic materials are considered, specifically C/C composites.

C/C composites are made of carbon fibers randomly disposed or woven, constituting a pre-form of a carbon matrix. Their properties depend on these two components, as well as on the fiber/matrix interphase. These materials are thus planned to be used for control rod cladding (start-up/shut-down control rods and operating control rods) and core support plates because of their high specific resistance and good mechanical behaviour at temperatures above 1 100°C. Unfortunately, these materials are sensitive to oxidation at such temperatures and their behaviour under irradiation is badly known; this behaviour is particularly critical for operating control rods. Some industrial solutions exist to improve the oxidation resistance (protective coating, SiC impregnation, self-healing matrix, use of SiC/SiC), but irradiation behaviour might be a real concern (risk of swelling, drop in thermal conductivity and mechanical properties, failure).

Fiber architecture is specific to the component, so that the fiber can bear the thermal-mechanical loading. The fiber nature, architecture (1D, 2D, 3D), matrix nature, heat treatment and geometry of the components have to be adjusted to meet the specific requirements of control rods. Finally, composite materials are inhomogeneous and most of the time anisotropic. Moreover, there is a wide dispersion during mechanical characterisation, which necessitates a large quantity of tests. The tensile property and dimensional stability of this type of composite has been proven satisfactory only up to moderate doses of around 10 dpa under mixed spectrum.

The main issues are:

- long-term stability of dimension and physical properties;
- detrimental irradiation effect on the interphase and its capability of deviating cracks and thus providing reasonable fracture toughness;
- required high creep strength of the fiber to bear thermal-mechanical loading in long-term service under high temperature and neutron flux;
- type of mechanical damage under irradiation and creep.

The behaviour of these materials under coupled irradiation and mechanical stress needs to be assessed. There is a real need for irradiation data, for better dimensioning the composite materials. Oxidation behaviour has to be evaluated as well, to ensure integrity of the control rods in case of accidental air ingress. Data on thermo-mechanical properties are also needed. Testing components that are as representative of the final structure as possible is also necessary.

Finally, modelling of composite behaviour will be essential for assessing the lifetime of components (for control rods, typically three to five years lifetime is considered).

2.1.5 Intermediate heat exchanger

Identified technologies of compact IHXs are:

- plate-machined, stamped or chemically-etched heat exchanger assembled by diffusion bonding;
- plate-fin heat exchanger;
- corrugated plate heat exchanger.

The conditions are summarised in Table 1.

Key issues to be addressed are:

- materials development: Ni-base alloys (Inconel 617, Haynes 230), ODS;
- design: compactness, high thermomechanical resistance, high thermal efficiency (95%), low pressure drop, no leakage, inspectability and lifetime (20 years is the desired lifetime);
- properties required at high temperature: tensile, long-term creep, fatigue, creep-fatigue, corrosion resistance, fabrication and joining techniques.

Ni-base alloys are the reference materials for gas-cooled primary system components, such as the IHX. This component should work under very severe conditions (very high temperature of 800 to 1 000°C and high pressure of 5 to 8 MPa). The main topic of these heat exchangers concerns the coupling between ideal geometries coming from the optimised design and the ability of the Ni-base materials to be formed and assembled. Finally, the guarantee of material integrity over the life duration of the exchanger has to be validated with robust modelling.

The material for the primary circuit must exhibit very good thermal stability for long operating time, moderate creep strength and well-established metal working and welding techniques. The candidates will thus be selected within the class of Ni-base, solid solution strengthened superalloys. In the past, exhaustive work has been performed on Inconel 617 and Hastelloy X. Inconel 617 exhibits the best creep properties, but its high cobalt content may lead to potential radioactive contamination problems. Hastelloy X and its variants exhibit good stability in VHTR environments, but do not perform as well in terms of creep resistance. Commissariat à l'énergie atomique (CEA) has selected Haynes 230 as a very promising candidate material for this application. Indeed, this material has been developed to withstand aggressive environments and exhibits good creep properties.

Experimental programmes have been launched to qualify Haynes 230. It includes long thermal exposure treatments, mechanical characterisation (tensile, impact toughness, and creep under air and vacuum) and corrosion studies (static exposure tests and creep tests under impure helium in benign conditions). Both the bulk material and the welds are tested. Results of creep tests at 750 to 950°C under vacuum and oxidising helium showed no significant influence of the tested environments and no significant difference between Inconel 617 and Haynes 230.

2.1.6 Specific core materials of the VHTR

Thermal reactors need a neutronic moderator, which should withstand the ion core conditions of the reactor (high-temperature helium environment and irradiation). Graphite appears to be the best candidate due to its properties:

- low neutron absorption;
- refractory material;
- relatively corrosion resistant;
- low cost;
- industrial knowledge.

The requirements for nuclear graphite are:

- good mechanical properties and thus high density;
- dimensional stability under irradiation, thus satisfying isotropy;

- low neutron absorption;
- low absorbing or minimising impurities with significant activation cross-sections, in order to produce a waste with minimised activity.

These points need to be studied and validated.

In VHTR reactors, graphite will be irradiated at temperatures between 500 and 1 200°C, depending on the components considered. Phenomena that need to be studied under irradiation are:

- Dimensional variations: the fast neutrons flux generates carbon atom displacement in interstitial positions and vacancies between the graphite planes. These defects generate modifications of the graphite dimension that depend on:
  - irradiation temperature (Between 300 and 700°C, a contraction occurs in the two directions of polycrystalline graphite: perpendicular and parallel to the graphene planes. In this temperature range, the higher the temperature, the lower the dimension variations. Above 700°C, a contraction also occurs, but in that temperature range, the higher the temperature the higher the deformation rates.);
  - crystallite size (the higher the crystallite size, the better the graphite dimensional stability);
  - graphite isotropy.
- Thermal conductivity: under irradiation, degradation of the thermal conductivity of polycrystalline graphites occurs for very low neutron fluences. For a given temperature, the thermal conductivity monotonously decreases with fluence. For a given fluence, thermal conductivity decreases with the increase in irradiation temperature.
- Young modulus: under irradiation the Young modulus of polycrystalline graphite increases considerably, which can lead to embrittlement of the material. This phenomenon appears for low fluences and is more important for lower irradiation temperatures (< 300°C).
- Irradiation creep: irradiation creep appears significant for temperatures as low as 100°C and can lead to important deformations.
- Corrosion: graphite is extremely sensitive to the presence of oxidising species in helium. Graphite oxidation leads to the formation of CO, CO_2, H_2..., depending on the oxidising gas, and can lead to strong degradation of the material, which, in extreme cases, can impact the safety of the system. The oxidation process of graphite is strongly dependent on temperature and the corroding species. Controlled tests should be performed to understand precisely the behaviour of graphite and be able to predict its behaviour in normal and abnormal conditions (air ingress).

No irradiation test is presently scheduled in France.

Finally, from all these results, technical files for codification of design standards should be established.

2.1.7 Specific clad materials of the GFR

Neutronic requirements and operating conditions of the GRF core components are:

- neutron transparency and low activity that disqualify all purely refractory metals, such as tungsten, rhenium, tantalum, molybdenum and niobium bases for core structural materials;
- normal operating temperature range of the core materials: 400 to 1 000°C;
- conventional accidental transients of the fuel element, up to 1 600°C with extreme accidental situations implying temperatures as high as 2 000°C;
- irradiation conditions: E > 0.1 MeV, $2x10^{27}$ $n.cm^{-2}$, 80 dpa (three years).

Requirements of the GFR fuel cladding are:

- leak-tightness barrier to fission products;
- good mechanical behaviour up to 1 600°C (integrity up to 2 000°C): ductility, fracture toughness;
- thermal conductivity (> 10 $W.m^{-1}.K^{-1}$);
- chemical compatibility with fuel and cooling gas (helium).

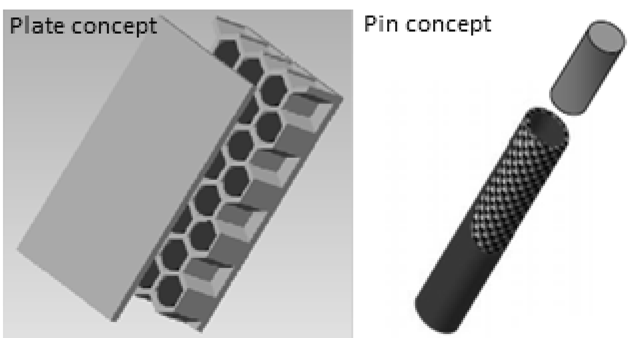
Regarding all of these ambitious target characteristics, the development of refractory core materials has been identified as a major critical issue considering the feasibility of the GFR concept and thus as a first priority for the GFR R&D that has been focused on the definition of the GFR element.

After a first phase of selection of different candidate materials, fiber SiC/SiC composites have been chosen for reference core inert materials to reach the very ambitious goals assigned to the cladding of the fuel element.

In parallel to the development of such an innovative material, CEA has launched an exploratory study about the possible use of semi-refractory metallic structural materials as V-base alloys. Obviously, such a back-up solution would be used in conditions where the maximum operating temperature of the reactor could be lowered without significantly degrading performance of the GFR system.

Initially, the different concepts envisioned for the fuel element are the SiC/SiC pin reference concept and the SiC/SiC, then metallic plate back-up concept (Figure 4). Today, taking into account the fabrication and joining difficulties of the fine honeycomb structures, the plate concept has been abandoned to the pin concept.

Figure 4: View of GFR fuel



Ongoing research on SiC/SiC composites concerns:

- composite elaboration process development and optimisation;
- definition of a dedicated joining process;
- interphase stability under high dose and high temperature;
- SiC/SiC deformation capability and thermal conductivity under irradiation: indeed, SiC undergoes swelling under irradiation, which decreases with increase in temperature up to 1 000°C and increases again above 1 000°C. The thermal conductivity of SiC decreases under irradiation and seems to remain constant above some dpa at room temperature, or above 1 000°C;
- effect of the impure helium environment on corrosion and erosion resistance of SiC/SiC;
- composite development for obtaining good leak tightness to the fission products: development of CVD ceramic or metallic barriers having good chemical compatibility with SiC and U/PuC;
- irradiation creep behaviour at high temperature: if the material is under constraint during irradiation, irradiation creep leads to deformation that increases with flux, fluence and temperature. In the case of SiC, it is not important up to 900°C, but increases above this temperature. For this point, a rig was designed for in situ creep tests under tensile stress for mini composites (CEDRIC) in OSIRIS: T = 600 to 1 000°C, 150 MPa, 6 dpa/yr at online monitoring of temperature and tensile stress;
- simulation of irradiation effects via charged particles in the Saclay JANNUS facility (three beams irradiation device);
- multi-scale modelling of thermomechanical behaviour;
- codification work adapted on composites for nuclear use, focused on the definition of damageing modes, criteria, dedicated testing and associated data bank.

2.1.8 References

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2.2 Japan

The Japan Atomic Energy Agency (JAEA) VHTR is similar to the JAEA high-temperature test reactor (HTTR) reactor, but the VHTR is designed for a longer reactor lifetime and larger scale. Operation temperature of the VHTR is the same as the HTTR, so that almost the same structural materials can be used in the VHTR, but some new materials are considered for lifetime extension.

2.2.1 Structural materials for the JAEA VHTR

All of the structural materials for the JAEA VHTR are used in helium environments containing impurities, such as H_2O and CO_2. Table 2 summarises structural materials for the JAEA VHTR, compared with the HTTR.

IG-110 or IG-430

The graphite core and reflector will be used at a temperature range between 400 and 1 000°C, and neutron dose is to be $10^{26}m^{-2}$. Lifetime of these components is assumed to be six years. The material (IG-110) used in the HTTR is also the candidate material for the VHTR. Material coding is under consideration in the ASME and Atomic Energy Society of Japan (AESJ). The R&D issue of the graphite core is lifetime extension. The most important issue on graphite material is irradiation resistance (dimension stability, irradiation creep). Degradation of the material during service should be confirmed. IG-430, which has more resistance against thermal shock and neutron irradiation, is considered as replacement material. Another advanced material for these components is SiC/SiC composite; however, the material database of SiC/SiC composite is quite limited.

2.25 Cr-Mo

2.25Cr-Mo steel is considered for use for RPV and high-temperature coaxial tubes. These components will be used for 40 to 60 years. Commercial-grade material can be applied for these components. There is no R&D issue for this material. 9Cr steel is also being investigated for the capability to extend component lifetime.

Hastelloy XR

Hastelloy XR will be used for high-temperature coaxial tube liners and IHX heat pipes. Service temperature is 900 to 950°C, with a lifetime of 40 to 60 years. Because Hastelloy XR is also used in the HTTR, a material database, including long-term creep properties, has been prepared. However, the VHTR is expected to have a longer lifetime than the HTTR, thus the database should be extended. Since these components will not be exposed to neutrons, no irradiation data is needed. Ni-Cr-W steel is also being investigated for availability. Basic composition and some properties of Ni-Cr-W steel have been determined. Another candidate material is Inconel 617.

Alloy 800H

Alloy 800H is used for control rod sleeves in the HTTR. This material can also be used in a VHTR, but only below 900°C for times less than five years. C/C composite and SiC/SiC composite are considered as materials for control rod sleeves to improve reactor performance; however, not enough data have been obtained to evaluate the viability of these materials.

Superplastic ceramics

The upper shield and core barrel are made of graphite and metallic materials in the HTTR. Instead of these materials, superplastic ceramics, such as stabilised zirconia, are being investigated to modify fabricability for the complex shape of the component.

Table 2: Components of JAEA's VHTR and R&D

Component	HTTR	VHTR	R&D subject
Fuel component reflector	IG-110	IG-110 or IG-430 SiC/SiC composite	Database expansion life extension
Control rod sleeve	Alloy 800H	C/C composite SiC/SiC composite	Database expansion design code development
Upper shield core barrel	Graphite metallic materials	Superplastic ceramics (fine grain stabilised zirconia, etc.)	Database expansion design code development
RPV IHX casing high-temperature coaxial tube	2.25Cr-1Mo steel 4.8 MPa at 440°C	Mn-Mo steel (w/ helium cooling) 2.25Cr-1Mo, 9Cr steel (without helium cooling)	Database expansion for the design code development, including irradiation effects
High-temperature coaxial tube liner IHX heat pipe	Hastelloy XR 0.29 MPa at 955°C	Hastelloy XR (< 950°C) Ni-Cr-W steel (< 1 000°C) Inconel 617	Database expansion in Hastelloy XR long-term properties database expansion for the design code development for Ni-Cr-W steel

2.3 Republic of Korea

VHTR technology can provide the high-temperature heat needed for hydrogen production. The goals are to achieve a temperature of 950°C for high efficiency in hydrogen production by the Sulfur-Iodine (SI) process; achieve a high-pressure difference between the reactor system and hydrogen production system; and develop corrosion-resistant materials for sulfuric acid, hydroiodide, and helium environments for a long component lifetime. The reference reactor concept has a 200 MWth helium-cooled core based on either prismatic block fuel or pebble fuel. The VHTR system has a coolant outlet temperature of 950°C.

Our scope for the materials R&D work is as follows: (1) material screening/selection and qualifying for a RPV, IHX, core structures, and process heat exchanger materials for a SI system based on their high-temperature properties, irradiation behaviours, corrosion resistance and manufacturability; (2) codifications of relevant high-temperature structural design rules to extend the American Society of Mechanical Engineers (ASME) and Korean Electric Power Industry Code (KEPIC) to the very-high-temperature region and support licensing of a system design; (3) material characterisations and database establishment; (4) alloy modifications and developments; (5) non-destructive evaluations; and (6) tests and optimisation of high-temperature components.

Candidate materials of each component are summarised in Table 3. Among them, our current work is focusing on the following materials: (1) SA 508/533 and 9Cr-1Mo for a RPV, (2) Inconel 617 for an IHX and hot gas duct, (3) C_f/C composite for a control rod, and (4) graphite for a reflector and support structures in the core region. Modified 9Cr-1Mo steel is being considered as candidate material for a RPV. Some mechanical properties, such as tensile/yield strength, fracture toughness, hardness, etc., were measured and the thermal ageing effects at 600°C were analysed. Creep and fatigue tests of base metals and weldments are being performed.

For an IHX and a hot gas duct, screening tests of Ni-base alloys and evaluations of high-temperature properties are being performed. The requirements of an IHX are to achieve an operating temperature of 950°C and an output power of 200 MWth, using helium. The creep tests of Inconel 617 under 18 to 35 MPa at 950°C in air were terminated and are now being carried out at 975°C. Effects of a helium environment on microstructural evolution and degradation in mechanical properties are being estimated. A creep stress of Inconel 617 was predicted by the Larson-Miller parameter method, with the single C-value method and multi C-values method at 950°C for 10^5 hours. Using the multi C-values method, higher reliable prediction data could be obtained. The microstructure and mechanical properties of Inconel 617 were also investigated after exposure at 950 to 1 050°C in air and helium. Compression and hardness tests of Inconel 617 were carried out after ageing at 1 050°C.

Table 3: Summary of candidate materials for each VHTR component in the Republic of Korea

Component	Standard material	Type	T (°C)	Code status
RPV	SA 508/533	Low alloy steel	380	ASME
	Mod.9Cr-1Mo	F/M steel	593	ASME 2004
IHX/Hot gas duct	Hastelloy X (22Cr-18Fe-9Mo)	Ni-base alloy	900	Section II 2004
	Inconel 617 (22Cr-9Mo-12Co)	Ni-base alloy	982	ASME Draft Code Case
	Haynes 230 (22Cr-14W-5Co)	Ni-base alloy	900	Section II 2004
Composite for control rod, core internal	X8CrNiMoNb 1616 (16Cr-16Ni-2Mo-1Nb)	Austenitic steel	650	Irradiation service
	C_f/C, SiC_f/SiC composite	Fiber reinforced composite	~1 000	-
Graphite for reflector and core support structure	Graphite	Nuclear grade	600	ASME draft

Ceramics and composites are being considered as candidate materials for insulating structures, control rod components and other internal structures, such as core restraints, belts, core barrel and tie-rods. Ceramics and composites are also of interest for control rod components. Increase in operating temperature of a high-temperature gas-cooled reactor up to 950°C requires ceramic composites for the control rod cladding and guide tubes, instead of metallic materials such as Alloy 800H. C_f/C composites are being considered as the structural components of the control rod in a short-term option and SiC_f/SiC composites for longer-term application. Evaluations of high-temperature properties, such as thermal conductivities in the range of room temperature to 1 200°C and the flexural strength of C_f/C composites, have been carried out. The oxidation behaviours are being evaluated.

2.4 Switzerland

2.4.1 Introduction

Switzerland, as a full member of the Gen-IV initiative, has decided to actively contribute to different projects with its main emphasis on gas-cooled reactors (VHTR, GFR). After the temperatures given for the VHTR in the roadmap were considerably reduced over the past five years, currently two concepts are being considered. Short-term (type NGNP): gas outlet temperature 750°C and advanced: gas outlet temperature 900 to 920 ^{0}C. In the case of the GFR, design of the core and particularly the fuel are very important. Even with reduced gas outlet temperatures and a cold vessel option the VHTR bears considerable design challenges that need a sound scientific background. Typical questions (among others) are:

- long-term creep properties (extrapolation), including eventual diffusion-controlled creep;

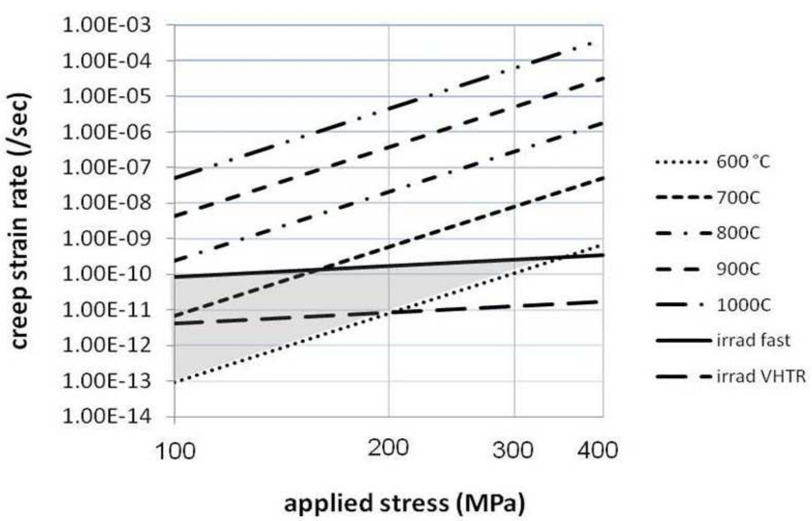
- creep-fatigue interactions;
- irradiation damage and irradiation creep;
- negligible creep criteria, including “low temperature creep” effects;
- condition-based monitoring;
- high-temperature corrosion.

Materials of interest are conventional RPV steels, Grade 91, Inconel 617, Alloy 800H and others. With respect to the high-temperature option, more advanced materials like nano-structured F/M steels, TiAl or fiber-reinforced ceramics and ODS materials are also considered as advanced cladding materials in SFRs or as a back-up-solution for GFR pellet-type fuel.

2.4.2 Typical research performed at HTMAT

Thermal creep properties of the ferritic ODS steel PM2000 were investigated and compared with Inconel 617 and an intermetallic titanium aluminide. For temperatures up to 1 000°C, it is expected that TiAl has about a factor of 2 (in stress) better stress rupture behaviour than Inconel 617. Its main drawback comes from very low ductility at room temperature. From 950°C up to higher temperatures, there is a clear, visible advantage of ODS steel. Expectedly, the stress rupture strength of ODS steels increases with decreasing particle size. Irradiation of materials leads to different types of damage. Swelling under ion irradiation is one type of damage that is investigated. Applying constant load under irradiation at temperatures up to 650°C leads to irreversible deformation, usually called irradiation creep. Figure 5 shows a thermal creep/irradiation creep map for TiAl, as an example. Interesting results were gained for the irradiation creep behaviour of ferritic ODS materials. It turned out that dispersoids had almost no effect and irradiation creep behaviour was the same as for similar materials without dispersoids. This is in contrast to thermal creep, where dispersoids are responsible for improved creep properties. The shaded area of Figure 5 represents conditions of possible interactions between thermal creep and irradiation creep. At temperatures above about 750°C, irradiation creep cannot take place due to annealing of the defects created.

Figure 5: Thermal and irradiation creep rates of TiAl as a function of stress and temperature



Another important research field concerns damage characterisation with miniaturised samples. This could become an important option for condition-based monitoring of components. Small samples could be removed from components and tested with techniques like micro-pillar compression or nano-indentation. Together with microstructural investigations in transmission electron microscopy (TEM) or in beamlines, improved damage characterisation and residual life assessments should be possible.

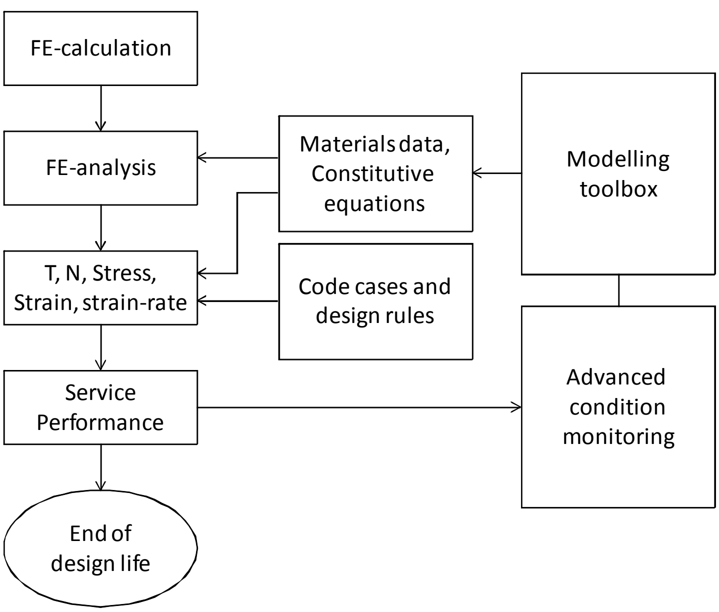
The experimental work is supported by multi-scale modelling. Ab-initio, molecular dynamics and dislocation dynamics modelling techniques are used to provide a sound understanding of damage. Table 4 shows the possibilities of using modelling for the analysis of technical problems and ways in which the models can be validated.

Figure 6 indicates ways in which these techniques could be used for improved lifetime assessments of advanced nuclear power plants.

Table 4: Modelling techniques and validation tools for analysis of technical problems

Technical problem	Modelling tools	Validation tools
Phase diagrams and microstructural stability	Ab initio, MD, thermodynamic modelling (kMC, rate theory, external support)	Co-ordination chemistry, phases, magnetic issues (TEM, beamlines)
Mechanical properties (strength, toughness, etc.)	Dislocation dynamics (dislocation-obstacle, interactions, dislocation patterning), MD (TEM contrast)	Micro (mechanical tests), dislocation arrangements, dislocation mechanisms (TEM)
Oxidation/corrosion	Ab initio, thermodynamic modelling, kMC	Chemistry, phase formation, (HR)TEM, EELS, beamlines
Fracture	Advanced FE, dislocation dynamics	Convectional samples

Figure 6: The possible role of modelling and advanced condition monitoring for residual life assessment in future nuclear power plants



2.5 United States

2.5.1 *Overview*

Material needs, as defined for the US VHTR systems [also known as the Next Generation Nuclear Plant (NGNP)] have been detailed in [1]. The summary below is derived from details in that report.

Planning assumes a reactor outlet temperature of 900 to 950°C and an operating pressure of 5 to 9 MPa. Due to a lack of maturity of materials for very-high-temperature systems, the impact of reducing the reactor outlet temperature from 950°C to 750 to 800°C is also being studied.

2.5.2 *Vendors*

Three vendors have contributed design information to the NGNP project. AREVA proposed a reactor outlet temperature of 900°C connected to an indirect steam cycle. General Atomics proposed a plant with two primary power conversion loops (i.e. two 300 MWth maximum steam generators (SGs) and two 6 MW maximum primary helium circulators). The General Atomics concept uses a cooled RPV to allow LWR steel (i.e. SA 508/533) to be used. Westinghouse proposes a 500 MWth system at 950°C reactor outlet temperature connected to a 50 MWth hydrogen production loop and a separate 470 MWth Rankine cycle SG. All three vendors propose helium as the primary coolant.

2.5.3 *Reactor pressure vessel*

Two different materials are being considered for the RPV, Grade 91 and SA 508 Grade 3, which are currently used as a LWR vessel material. To use SA 508 requires a vessel cooling system to limit the maximum temperature seen by the vessel. Grade 91 still needs to be codified through the ASME for use above 371°C. Additionally, welding procedures for fabricating large sections in representative geometries of Grade 91 need to be developed. SA 508 Grade 3 still only has limited data on long-term thermal and environmental ageing effects in an impure helium environment.

2.5.4 *Reactor internals*

Reactor internals support and maintain the reactor core geometry within the vessel, provide thermal and radiological shielding and transport heat. These include the top plenum, upper core restraint, side reflector, and core outlet plenum. Materials being considered for reactor internals include metallic, graphite, and composite components. Materials for metallic components (typically Alloy 800H) tend to be mature without large development needs. Using components constructed of graphite requires qualification of a new graphite to replace the H451 graphite used in older reactors. Use of composite components (SiC-SiC or C-C) would require significant development prior to use.

2.5.5 *Core and structures*

Two different core designs have been proposed. AREVA and General Atomics have proposed a prismatic design (hexagonal columnar blocks of fuel, moderator, or reflector material placed in a columnar array), while Westinghouse has proposed a pebble-bed design (fuel elements are spheres rather than prismatic columnar blocks). For the prismatic design, the mechanical stability of the graphite blocks during loading and unloading, as well as during seismic events, must be determined. These mechanical properties must be known for both un-irradiated and irradiated materials. As described in the section on reactor internals, qualification of new grades of graphite, including irradiation testing, must be conducted to verify properties. This includes understanding tensile and creep properties.

2.5.6 Heat transfer system

The heat transfer system for the NGNP uses helium to transfer heat from the primary heat transfer system (PHTS) to the secondary heat transfer system (SHTS) through an IHX. Five different materials are being considered for the IHX vessel: Haynes 230, Alloy 800H, Inconel 617, Hastelloy X and ceramics. Six different configurations are being considered for the heat exchanger: shell and tube, involute, capillary tube, helical tube, plate and fin and printed circuit. In addition to helium, three other options are being considered for the secondary cooling fluid: nitrogen/helium mixtures, molten salts and steam.

In selecting material for IHXs, many material properties are being considered, including creep-fatigue, embrittlement, thermal expansion, thermal fatigue, crack resistance, oxidation resistance, carburisation resistance and thermal conductivity, all for varying maximum operating temperatures. Additionally, fabricability, licensing, availability, erosion, corrosion performance in an impure helium environment (or salt or steam depending on final coolant decisions) and cost are also critical. Two final parameters of importance include dust susceptibility and tritium migration for each material being considered.

2.5.7 Cross-vessel piping

Cross-vessel piping connects the RPV and IHX and consists of four main components: the support structure, hot and cold ducts, insulating materials and the insulation liner. Hot helium coolant (900°C) is transferred via the hot duct from the core, while cold helium coolant (500°C) is returned via the cold duct to the core. Two designs have been suggested. One separates the hot and cold ducts such that an insulation liner is not required. An alternative design has the hot and cold ducts collocated and separated by an insulation liner. This second design may enhance heat transfer and serve as a pre-heater to fluid entering the reactor.

Support structure material options for the cross-vessel piping are SA 508/533 and Grade 91 steels, corresponding to the RPV material options. Insulation material options are Kaowool, refractories and ceramic fiber (alumina). Liner material options are Alloy 800H, Haynes 230, Inconel 617, Hastelloy X, and ceramics.

2.5.8 Reference

[1] Idaho National Laboratory Technical Report (2009), “Next Generation Nuclear Plant Project Technology Development Roadmaps: The Technical Path Forward”, INL/EXT-08-15148, Revision 0.

3. Sodium-cooled systems

3.1 France

3.1.1 Introduction

In January 2006, the French President requested design of a Generation IV system that could be in operation by 2020. In June 2006, the French parliament passed a law requiring the options of future nuclear systems to be studied by 2012 and a prototype to be in operation by 2020. Finally, in December 2006, an inter-departmental committee agreed on a technical roadmap for SFR, GFR and fuel cycle studies leading to gathering data to choose future options in 2012. This strategy is based on the necessity to save uranium and reduce ultimate waste in the future.

Therefore, an important programme has been launched in France by three partners, CEA, AREVA NP and EDF, to develop an innovative SFR concept. The main milestones of the programme launched in France are as follows:

- 2009 for the selection of main tracks for R&D and studies;
- 2012 for the selection of the main design options;
- 2020 for construction of a prototype;
- 2040 to 2050 for the industrial deployment of a new generation of SFRs.

The materials candidates for SFRs have been evaluated through past successive projects: Phénix, Superphénix, Superphénix 2 and European fast reactor.

Table 5 synthesises the main components and materials used in the past and those envisioned for the new reactors.

The main requirements of the reactor materials to be used are the following:

- the materials need to exhibit dimensional stability during irradiation, both under stress (irradiation creep or relaxation) and without stress (swelling);
- the mechanical properties of the materials (tensile strength, ductility, creep resistance, fracture toughness, resilience) have to remain acceptable;
- the materials must retain their properties in response to corrosive environments such as liquid sodium, fuel cladding chemical interaction, water vapour, eventual alternative fluids;
- the clad materials have to be compatible with the reprocessing process in use;
- life duration: the behaviour of materials has to be guaranteed at least up to 60 years.

Table 5: Material choices for SFRs

Component	Specificities	Materials used in the past (PX & SPX)	Materials for new SFRs
Sub-assembly	T = 400-650°C > 150 dpa	Clad : Austenitic steels: Ti-stabilised Type 316 stainless steel, 15-15Ti, duct : Ti-stabilised type 316, 9Cr-1Mo martensitic steel	Clad : ODS F/M steels, advanced austenitic steels, duct 9Cr martensitic steel or low activation martensitic
Structures in contact with sodium	Internals and hot circuits T ≤ 550°C	316 L (low-carbon) 316LN (controlled nitrogen) stainless steel	Same austenitic steels
	Auxiliary piping, little loaded T < 400°C	Z2 CN 18-10 or 304L	
	Main vessel T ~400°C	Z2CND17-2 with controlled nitrogen or Type 316LN stainless steel	
	IHX T ≤ 550°C	Z2CND17-2 with controlled nitrogen or Type 316LN stainless steel	Same austenitic steels or 9Cr-1Mo steel
Upper closure of vessel	"box" structure design	A42	A42 or 16MND5 if design in thick plate
Steam generator and vapour pipes	Helicoidal tubes vapour collector T < 490°C	Z5NCTA33-21 or Alloy 800 2.25Cr1Mo	Same or modified 9Cr1Mo if straight tubes design

Other criteria for the materials are their availability and their costs to fabricate and assemble. Moreover, their composition should be optimised to present low-activation (or rapid deactivation) features that facilitate maintenance and disposal.

3.1.2 Sub-assembly materials

Indeed, the will to reach high burn-ups, to optimise the use of resources and minimise waste requires materials with much lower swelling than that of the materials currently used. The maximum hoop deformation of austenitic steels increases rapidly after an incubation period, while that of F/M steels remains low even at displacement doses close to 150 dpa. This also shows the need to be thorough in the material characterisation, since no problem seems to exist until the displacement dose reaches approximately 70 dpa; at that dose, a swelling mechanism gets triggered for some grades of austenitic alloys. Also, implementing low swelling materials may allow reducing coolant channel thickness, which is one of the paths to cope with the coolant void reactivity coefficient, a major safety issue for SFRs.

Normal operating conditions of the SFR are:

- dose : higher than 150 dpa;
- stress at the end-of-life : 100 MPa;
- T: 400 to 650°C.

Concerning the cladding material, the reference choice considers ODS F/M steels, while austenitic steels such as 15-15Ti or advanced austenitic steels are kept as back-up. 15-15Ti steel will be the cladding material for the first cores of ASTRID.

The main benefits of ODS F/M steels are that, due to their F/M matrix, they show very good behaviour with respect to swelling at high doses. Moreover, the nanosized dispersoids of yttrium oxide provide these alloys with good creep resistance at high temperatures. The ODS grades currently developed in the framework of the SFR or fusion reactor development programmes contain 9 to 12% Cr. However, these alloys could show some limitations in terms of internal corrosion (oxide clad reaction), behaviour in the reprocessing process and temperature (phase transition around 1 075 K). Therefore, ferritic steels with around 14% and more Cr have been developed and could be used up to 1 175 K. Although irradiation data are scarce, the Body-Centered Cubic (BCC) crystalline structure should present an excellent resistance to swelling. The main in-service issue in the low-temperature range remains the effect of unmixing on the mechanical properties. In the high-temperature domain, this issue is the required stability of oxide dispersion to maintain the improved creep resistance of this material and the absence of heavy intermetallic phase precipitation that could degrade the toughness of the cladding. Preliminary results under mixed and fast neutron spectrum show that demixing should allow this type of material to keep reasonable ductility and fracture toughness. The stability of oxide dispersoids under irradiation is an open issue to be settled. The action of the oxide dispersoids on irradiation creep remains to be understood.

For ODS steels, the ongoing research programme includes the following tasks:

- Elaboration:
  - mastering the industrial elaboration process (atomisation, mechanical alloying in an attrition mill, consolidation of the powders,...) to obtain a homogeneous product with the desired dispersoid size and reproducible properties;
  - characterisation of the elaborated materials: chemical composition, microstructure, morphology, mechanical properties;
  - alternative elaboration processes: direct hot extrusion (JAEA), ODS production via roll-bonding (e.g. the Generation-IV and transmutation materials (GETMAT) project working with the company OCAS);
  - fine analysis of the reinforcement mechanism between nano-scale particles and dislocations to define an optimal microstructure and develop mechanical behaviour models.
- Welding:
  - state-of-the-art on processes already used without fusion: friction stir welding, resistance welding, spark plasma sintering and hot isostatic pressing;
  - analysis of welding processes already applied for the welding of experimental sub-assemblies in Phénix;
  - elaboration of ODS steel samples welded in a representative geometry and study of their creep behaviour.
- Mechanical behaviour and irradiation effects:
  - mechanical properties determination (in particular, determination of the limiting temperature for use of the materials), study of the global matrix + nanoparticles stability of actual and new ODS after irradiation (Supernova, Matrix 1 and 2) and to be irradiated in future programmes to be considered in the following reactors: Osiris, RJH, JOYO, BOR 60, BN600;
  - simulation with charged particles (JANNUS, PSI, ….) and parametric study of nanoparticle stability and of mechanical properties in a large temperature range;
  - modelling of observed phenomena and prediction of the behaviour in thermal ageing conditions.

- FCCI: analysis of ferritic clads irradiated in Phénix.
- Corrosion: corrosion tests on un-irradiated materials are going to be performed in liquid sodium in a dedicated device (named CORRONA) with controlled chemistry (O, C) and hydrodynamics. These tests will allow the determination of the corrosion behaviour of different materials and understanding of corrosion mechanisms. This insight will be used to develop corrosion models to guarantee the 60-year integrity of plant components.
- Reprocessing: corrosion tests in nitric acid of the developed ODS steels and tests in nitric acid with irradiated clads, with and without fuel in Atalante.

Concerning austenitic steels, the present optimised austenitic steel, 15-15 Ti AIM1, will be confirmed to reach 100 to 120 dpa using recent irradiations in Phénix. Advanced austenitic steels are also considered in the objective to reach higher doses; evaluations are being performed notably on 15Cr-15/25Ni steels irradiated in the Supernova experiment in Phénix. First results (swelling) show some satisfying results for specific compositions.

For systems with lower operating temperatures of 400 to 550°C, 9Cr1Mo (W) martensitic steel is considered as the reference material for duct material. For this component, the fundamental requirements that have to be fulfilled are:

- swelling resistance;
- irradiation and thermal creep;
- embrittlement under irradiation, toughness, ductile-brittle transition temperature (DBTT);
- weldability;
- behaviour in Na environment.

The 9Cr martensitic steels, with their low-activation variants for fusion, are foreseen to be used for temperatures up to 550°C. A large set of data issued from the various fast neutron reactors programmes exists for the classical 9-12 Cr martensitic steels for doses around 100 dpa in the range of 400 to 550°C. The resistance to swelling is excellent due to the BCC crystalline structure and the high density of sinks in the martensitic microstructure. The hardening and DBTT shift are acceptable when irradiation occurs in the range of 400 to 550°C. In this irradiation temperature range, the DBTT of 9Cr-1Mo steels as EM 10 remains lower than room temperature, even at high doses.

3.1.3 Structural materials

For the structural materials, austenitic and F/M steels are considered. F/M steels appear promising due to the following properties:

- high thermal conductivity;
- lower thermal dilatation than austenitic steels;
- lower cost than austenitic steels.

Nevertheless, at this stage the actual materials choices for the SFR design are the following:

- Below core structures and primary circuit (cold structures – 400°C, no deformation, low irradiation): the reference choice remains Type 316LN (low-carbon and controlled nitrogen) stainless steel.
- Above core structures (ACS; hot structures – 550°C, creep, weld joint coefficient, low irradiation): the replacement of Type 316LN stainless steel is considered due to

degradation of creep and creep-fatigue properties and other materials are examined. For these structures, either an evolution of Type 316 stainless steel is considered or the use of a Ni-base alloy.

- Secondary circuit, IHX and SG: for these structures, the use of F/M steels (reference: Grade 91 steel) is strongly considered due to the potential gain of a much more compact circuit (due to their satisfying mechanical and thermal properties), thus making the system more competitive economically. For these parts of the reactor, the choices of materials also depend on the design and the technical choices (e.g. use of a S-CO_2 conversion energy system).

For F/M 9-12 Cr steels, the ongoing research programme includes the following tasks:

- The main developments have been performed on F/M 9-12 Cr steels to improve creep behaviour. For this aspect, CEA is working on the development of a new, optimised 9-12 Cr steel by adjusting optimal chromium concentration and the reinforcement mode (W + carbides…) to obtain the best compromise between mechanical characteristics, compatibility with various environments and assembly. Ongoing developments are:
  - Grade 92: microstructural characterisation of this type of steel, creep behaviour study;
  - new alloys Fe-9Cr-xW-V-N: development of new martensitic steels for creep properties optimisation.
- Other mechanical properties: creep, fatigue, and creep-fatigue properties are studied for these materials, especially extrapolations of laboratory fatigue life data (months) for the prediction of very long-term in-service lifetime. This requires understanding mechanisms responsible for damage and fracture. This work is also performed on welded zones, HAZ.
- Weldability: specific studies are being performed on Grades 91 and 92 concerning hot cracking sensitivity during welding.
- Corrosion: corrosion tests are going to be performed in liquid sodium, in a dedicated device and with controlled chemistry to determine corrosion mechanisms and establish models to predict long-term behaviour of the materials (other phenomena, such as carburisation/decarburisation and oxygen impact, will also be studied). Other corrosion tests will be performed for various environments considered in the reactor: alternative fluid in the secondary circuit, S-CO_2 energy conversion system, water vapour, etc.
- Ageing: life duration of the materials has to be guaranteed up to 60 years, requiring modelling of various phenomena encountered by the materials.

For the austenitic steels, the main on-going work concerns:

- re-examination of existing data to determine a limit temperature for the use of Type 316 stainless steels for the hot primary circuit and to propose:
  - either evolution of the specifications for this type of steel;
  - or the use of an advanced austenitic steel.

CEA also contributes to the European Creep Collaboration Committee (ECCC), which works on the optimisation of austenitic steels concerning creep behaviour at high temperature. Optimised austenitic steels could thus be considered in replacement of Type 316LN stainless steel. The weldability of these new steels will have to be validated.

3.1.4 Reactor vessel

As previously discussed, Type 316LN stainless steel is the reference material for the vessel and internal components, such as ACS, internal vessel, diagrid and strongback. There is good confidence in this material due to a large available database.

Studies are mainly focused on determining the mechanical properties of this steel and modelling for extension of lifetime at least up to 60 years.

3.1.5 Internal components

As for the vessels, the preferred material is Type 316 LN stainless steel and studies are focused on lifetime extension. Nevertheless, few internal components, diagrid and ACS are submitted to neutron flux, close to 2 dpa. For those two structures, a very low content of activation and transmutation products as cobalt will be specified at the procurement stage. The influence of these doses on mechanical properties will be investigated.

3.1.6 Steam generators

The choice of material for SGs is complex because various aspects must be considered. In addition to the criteria noted in the introduction, it should have:

- a good resistance to water and water vapour oxidation;
- a good resistance to wastage;
- ability for inspection and in-situ intervention, in particular for cleaning operations.

The options include existing materials such as Alloy 800, used for Superphénix SG tubes, Alloy 800H, 2.25Cr1Mo, modified 9Cr1Mo and improved F/M grades such as P92 ASME grade, P122 and NF12 alloy. In a first selection, the prime candidate for SG material is the modified 9Cr1Mo, the advantage being better thermal performance, which allows significant economical gains compared, for example, to the better corrosion resistant Alloy 800. 2.25Cr1Mo steel has not been retained, as it is less creep and corrosion resistant. Concerning modified 9Cr1Mo, studies performed concern mainly:

- fabrication processes, to have fine control of the final microstructure to avoid degradation of the alloy's high-temperature properties;
- welding processes;
- thermal ageing consequences on the base metal and welded joints;
- cyclic behaviour at high temperature;
- oxidation behaviour in water vapour;
- interaction between fatigue, creep and oxidation.

Finally, concerning all materials of SFR, it will be essential to examine and characterise Phénix materials after their definitive end, to gain knowledge of the long-term behaviour of SFR structural materials.

3.1.7 Codification

The fourth edition of the *Règles de Conception et de Construction des Matériels Mécaniques des Îlots Nucléaires RNR (RCC-MR)* was edited in October 2007. Developments have to be performed linked to the change of materials (the RCC-MR was developed for austenitic steels). A new edition of the RCC-MR is foreseen in 2011. The following points are being examined to get dimensioning rules:

- 9Cr steels creep and creep-fatigue data: validity of the creep-fatigue rule of these materials:

- creep joint coefficients for new materials;
- tensile strength;
- toughness;
- ductility for base metal and weldings at end-of-life for Type 316LN stainless steels and 9Cr steels;
- rupture mechanics data.

Behaviour laws for new materials of the fuel assembly, ODS and F/M steels, irradiated and un-irradiated, will have to be established, as well as for new circuit materials. Behaviour laws already established for austenitic steels, Type 316LN stainless steels and 15-15Ti for example, will have to be re-examined to integrate recent results, notably under irradiation.

3.1.8 References

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3.2 Japan

3.2.1 Introduction – FaCT project

In Japan, the R&D programme on materials for fast breeder reactors is integrated in the Fast Reactor Cycle Technology Development Project (FaCT Project) led by the JAEA. The FaCT project is for the commercialisation of a Japanese sodium-cooled fast breeder reactor (JSFR) in around 2050 and operation of a demonstration reactor of JSFR, which is scheduled for around 2025.

3.2.2 Core materials

Fuel pin cladding and duct tubes of the JSFR fuels will be irradiated to a neutron dose of 250 dpa and exposed to a burn-up of 250 GWd/t in peak. Design studies for large- and medium-scale reactors set target discharge average burn-up to be 150 GWd/t. To improve thermal efficiency of the plants, maximum outlet coolant temperature at the RPV is determined to be 823 K, the corresponding maximum (hot-spot) temperature of the cladding tube is 973 K and that of the duct tube is 853 K. Since BCC structure is more resistant against radiation damage than face cantered cubic (FCC) structure, ferritic steels are indispensable for both tubes. It is necessary for the cladding tubes to be ODS ferritic steels, because PH ferritic steels will rapidly lose their strength over 923 K. On the other hand, PH steels can be applied to the duct tubes. So, ODS and PH steels have been selected as the most prospective candidate materials for cladding and duct tubes, respectively. Manufacturing technology developments will be extended from large-scale to commercial-scale. The material strength standard for ODS and PH steels will be established based on future out-of-pile and in-pile tests in the FaCT project. In addition to material irradiation tests that began in the 1990s, ODS steel clad fuel pins will be newly irradiated in JOYO to prove their irradiation performance. Depending on the progress of R&D, ODS steel clad fuel pin bundles with PH steel duct tubes will be irradiated in JOYO; irradiation in the prototype fast breeder reactor MONJU is also envisioned.

3.2.3 Structural materials

For structural materials for the JSFR, Type 316FR (fast reactor-grade) stainless steel and modified 9Cr-1Mo steel are applied. Both materials are subjected to 823 K for 60 years, which is the design life of the JSFR. Type 316FR stainless steel is a low-carbon and nitrogen-added stainless steel and has been developed in Japan to improve high-temperature creep properties. Chemical compositions of the steel are optimised within specifications of Type 316 stainless steel in the Japanese Industrial Standard. A unique feature regarding the chemical composition is that phosphorus is also added. This material has been adapted to the IHX of the experimental fast reactor JOYO located in Oarai, Japan and would be used for RPVs and internal structures of the JSFR. Modified 9Cr-1Mo steel is to be used for primary and secondary coolant systems, IHXs and SGs, to take advantage of low thermal expansion and good elevated temperature properties of this material. Alloy development for these steels is nearly complete, but there are some important issues to be addressed; the first is acquisition of long-term data that will form the basis for 60-year design and development methods to evaluate material performance, such as very long-term creep-fatigue. The second issue is the development of manufacturing technology to produce components required for the JSFR, such as the large diameter forged ring for the RPV (Type 316FR stainless steel), very thick forged plate and very long and thin double-walled heat exchanger pipes (modified 9Cr-1Mo steel). These are necessary for the economic advantage of the JSFR. The final point to be noted is that these steels have not been registered in current Japanese nuclear codes and standards published by the Japan Society of Mechanical Engineers (JSME) that will be used for the design and construction of the JSFR. Therefore, it is necessary to register the steels in the JSME code and to standardise the allowable stresses required for the JSFR structural design at elevated temperature. The JSME code, 2016 Edition, is scheduled to be used for the licensing process of the JSFR demonstration plant, which is envisioned to begin in around 2018.

The material selection for fuel assemblies and major components are shown in Figures 7 and 8.

Figure 7: Material selection for fuel assemblies of the JSFR

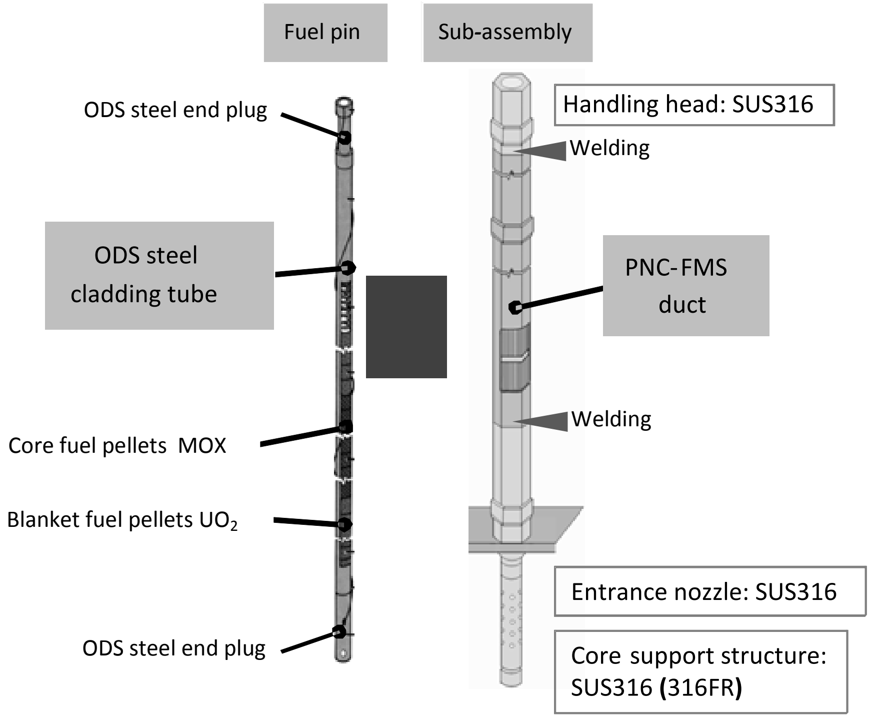
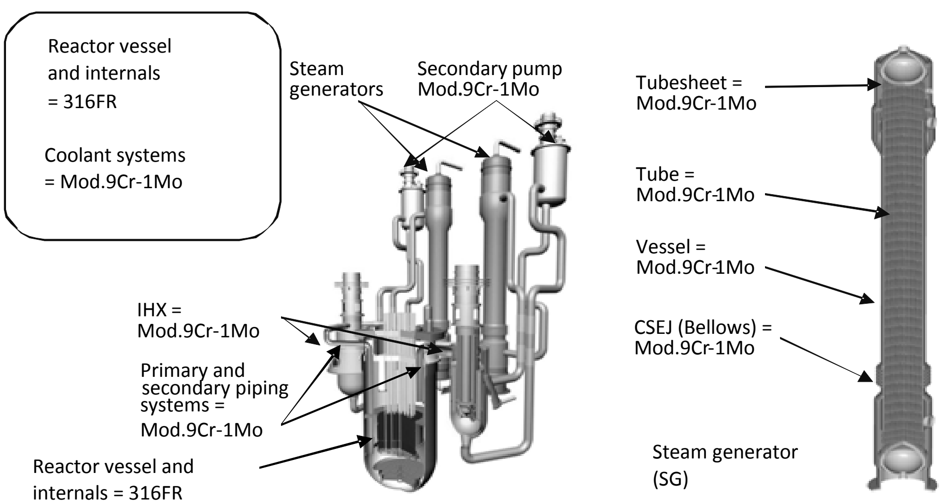


Figure 8: Material selection for major components of the JSFR



3.3 Republic of Korea

The conceptual design of a demo SFR in Korea was established in 2007. The demo SFR is designed to have a coolant outlet temperature of 545°C and a power of 600 MWe. Candidate materials for each component of the SFR are given in Table 6. HT9 steel is being considered as the candidate material for fuel cladding for the demo SFR.

Table 6: Candidate materials for each component of a demo SFR in the Republic of Korea

Reactor	RPV	IHX	IHTS* Piping	SG	Cladding	Remarks
KALIMER-600	Type 316 stainless steel	Mod. 9Cr-1Mo	Mod. 9Cr-1Mo	Mod. 9Cr-1Mo	HT9	Conceptual design

*Intermediate heat transfer system.

The design requirement for the fuel cladding is given in Table 7. Maximum allowable temperature and maximum fluence are designed to be 650°C and 250 dpa, respectively. Here, there is no limitation of cladding temperature by eutectic melting, due to the barrier to the inner surface of the cladding tube.

Table 7: Design requirements for fuel cladding

Maximum allowable temperature	650°C
Maximum fluence	250 dpa
Thermal strain	< 1%
Total strain	< 3%
Swelling	< 5%
Limit of cladding temperature by eutectic melting	None (Barrier to inner cladding tube)

Present activities are focusing on fabrication of HT9 cladding tubes and their performance verification through out-of-pile and in-pile tests. Fabrication of HT9 cladding tubes is scheduled by the end of 2013. The forthcoming activities are scheduled to verify the performance of HT9 cladding tubes, including in-pile tests in an irradiation facility. The manufacturing technology for the duct, wire and assembly part will be developed by 2025. After these verifications and developments, the fuel assembly is planned to load into the Korean demo SFR.

New F/M and F/M ODS steels are being developed, with a target of applying these steels to cladding materials of the commercial SFR in the future. Basic studies on developing new F/M steels were carried out in the first stage of the SFR project, from 2007 to 2009. Some of the newly designed F/M steels revealed considerably improved creep rupture strength compared with Grade 92 as a reference steel and a patent application was submitted. The performance evaluation under the in-pile condition is planned to be performed in the Korean demo SFR.

Fabrication processing parameters of the cladding tubes are recognised to be as important as alloy design, because they also affect the performance of F/M steels. The fabrication processes of the F/M steel cladding are mainly composed of melting, hot forging, hot extrusion, cold pilgering and heat-treatment. The optimised conditions at each stage are being studied. The purpose of this study is to form fine and uniform precipitates in a martensite matrix.

3.4 United States – clad and duct materials

3.4.1 Requirements for SFR core materials

One of the primary goals of the Fuel Cycle R&D (FCRD) programme is to use fast neutron spectrum to transmute minor actinides extracted from spent LWR fuel. One of the options for providing this fast spectrum is to use a SFR. Fuels in these fast reactors need to be irradiated to at least 20% burn-up before discharge and recycling for the process to be economically feasible. Presently, achieving a burn-up of 20% in fast reactor fuels is a significant challenge. The cladding materials used in such high-dose irradiations will experience at least 200 dpa of exposure. The requirements are that the cladding must contain the fuel until discharge and the duct must maintain ductility and mechanical strength to direct the sodium coolant and allow handling of the fuel sub-assembly during and after discharge. The material property needs for cladding are the following in order of importance:

- Radiation resistance (physical and dimensional changes): The cladding dimensional and physical properties must be resistant to irradiation-induced degradation. In the core midplane, where flux is the highest, the cladding must be resistant up to at least 200 dpa. Corresponding dose at the top and bottom of the core will be substantially lower, but irradiation temperature at these locations often poses additional challenges in maintaining irradiation resistance, especially at the bottom of the core, where irradiation temperatures typically drop below 400°C.
- Mechanical properties (before and after irradiation): Tensile properties (strength, ductility), creep resistance, fracture toughness and fatigue resistance must all be sufficient to contain the fuel during irradiation, discharge and other fuel handling operations.
- Fuel clad chemical interaction (FCCI): The cladding must be resistant to chemical interaction with the fuel during irradiation, to prevent clad failure.
- Fabrication and joining: The material must be able to be fabricated into component parts and joined to contain the fuel during irradiation and discharge.
- Corrosion resistance: The material must be tolerant of the sodium coolant, with minimal corrosion rates.
- Thermal properties: High-thermal conductivity and low-thermal expansion are important to maintain desired fuel and clad temperatures during irradiation.
- Neutronic properties: Low neutron absorption is required.
- Cladding material availability: Sufficient raw material must be readily available.
- Economics: The cost of the cladding material must be considered, but is not a significant driver.

Material performance needs for the duct are similar, but with some changes to the ranking of desired material properties. The purpose of the duct material is to control coolant flow distribution through the core, provide a means to insert and remove fuel pins, prevent propagation of fuel failures from one sub-assembly to others in the core and provide another layer of protection from fuel pin breach during fuel handling operations. The duct will operate at lower temperature than the fuel and thus the temperature envelope for the duct will be lower than the fuel. On the fuel pins stresses can build due to fission gas release, while stresses on the duct material will be due primarily to structural support of the fuel pins, interaction of the fuel assemblies with the core restraint system due to axial and radial thermal gradients in the core and to handling of the duct. Tensile strength and toughness will be the properties of highest importance, while creep will become less important. High-thermal conductivity, low-thermal expansion, low neutron absorption cross-section and high swelling resistance are still important. FCCI and reprocessing requirements are, of course, a non-issue with the duct.

3.4.2 Candidate materials in the United States for sodium fast reactor clad and duct applications

Presently, cladding material that has been widely used to the highest exposure in US fast reactors in coupon and component form is HT9 (a 12Cr F/M steel). Swelling was measured to be less than or equal to 2% after irradiating to 200 dpa at 390°C. Swelling at higher temperatures was less. Creep measurements have also been performed after irradiation to 200 dpa at 400 to 600°C, showing stable creep response to temperatures up to 550°C. At 600°C an increase in the magnitude of the transient creep response was observed and creep rate per unit stress began to strongly increase. One of the longest running fuel experiments (the ACO-3 experimental fuel sub-assembly and core demonstration experiments) in the Fast Flux Test Facility (FFTF) consisted of HT9 clad and duct with oxide and metal fuel. This ACO-3 sub-assembly ran without issue to a peak dose on the duct of 155 dpa and the plan was to continue to a higher burn-up. Laboratory measurements of post-irradiation toughness show a significant reduction for irradiation temperatures below 400°C, but there have been no instances of fracture of HT9 clad and duct during post-irradiation handling operations at FFTF. Thus, the near-term primary candidate clad and duct material for FCRD is HT9. The two main limitations of this material are a reduction in toughness at irradiation temperatures below 400°C and a reduction in creep strength as irradiation temperatures approach 600°C. Thus, advanced materials being considered include next generation F/M steels, ODS-strengthened ferritic steels and other advanced alloys that have the potential to offer better high-temperature irradiation creep resistance while maintaining adequate fracture toughness at lower irradiation temperatures. The next generation F/M steels show significant increases in creep resistance at temperatures up to 650°C. These alloys include Mod 9Cr-1Mo, HCM12A and NF616 (also being investigated as an advanced reactor core structural material). ODS-strengthened ferritic steels offer further improvements in creep resistance beyond 650°C, and also may provide significant radiation tolerance and helium management from the nanometer-scale precipitate dispersion. Alloys being investigated include MA-957, 12YWT and 14YWT, along with international collaborations investigating the leading ODS alloys being produced in Europe and Japan. All of these materials still require a great deal of further investigation, as they have yet to undergo significant irradiation testing to understand the effect of irradiation on fracture toughness and other key properties.

3.4.3 Existing databases

One of the most comprehensive existing databases for core materials for fast reactors is the Nuclear Systems Materials Handbook (NSMH), which contains irradiation data on HT9 to doses up to 100 dpa. This database is labelled applied technology and presently only US citizens are allowed access to the handbook. A materials handbook is being revised under the FCRD programme, includes data from the NSMH and will include additional data on HT9 to doses up to 200 dpa, as well as new data on advanced alloys.

3.4.4 Current data needs

HT9

- Radiation resistance (physical and dimensional changes): Irradiation tests have been performed on HT9 in coupon form to greater than 200 dpa and in component form to 155 dpa. Testing of these materials is in progress.
- Mechanical properties (before and after irradiation): The tensile properties (strength, ductility), creep resistance, fracture toughness and fatigue resistance have been tested before irradiation. Presently, HT9 is not available commercially, so some additional testing will be required on new heats. Irradiation data are presently available to 100 dpa. These data will be extended to 200 dpa after testing is completed in the next two years.
- Fuel clad chemical interaction: FCCI tests using HT9 as cladding have been performed with U-10Zr and MOX fuel to burn-ups approaching 20%. More testing is needed for transmutation fuels containing plutonium, americium, neptunium and curium.
- Fabrication and joining: Fabrication of tubing and duct materials has been successfully performed by industry. Welding parameters are well established and irradiation data are available on welded joints.
- Corrosion resistance: Significant data exist showing excellent resistance to corrosion when oxygen levels are controlled to below 10 ppm.
- Thermal properties: Significant changes in thermal conductivity are not observed until swelling levels exceed 1%. The coefficient of thermal expansion is nearly independent of irradiation dose/damage.

Given this level of understanding for HT9, technical readiness level (TRL) is around 4 (component validation in a lab environment) or 5 (component validation in a relevant environment) (where TRL 9 is proven success through full-scale demonstration) because of the need for data in contact with transmutation fuels, as well as the need to establish the production again by industry.

Advanced F/M and ODS-strengthened ferritic alloys

- Radiation resistance (physical and dimensional changes): Previous irradiations were performed in the FFTF Materials Open Test Assembly (MOTA) experiments on T91 and MA-957 to doses up to 110 dpa. These samples are being tested. Additional samples were irradiated in the Phénix MATRIX irradiations to doses up to 70 dpa and include T91, HCM12A, NF616, MA-957, 12YWT, and 14YWT at 400 and 500°C. This will provide some needed data on a coupon-scale, but additional data on component- as well as coupon-scale are still required to test irradiation resistance up to and greater than 200 dpa.
- Mechanical properties (before and after irradiation): The tensile properties (strength, ductility), creep resistance, fracture toughness and fatigue resistance have been tested for T91 before irradiation. Post-irradiation data on T91 are limited to a small number of tensile, fracture toughness, and creep tests from specimens, with most of the specimens irradiated to less than 100 dpa. There is a small but growing database on the properties of un-irradiated ODS-strengthened steels (MA-957, 12YWT, 14YWT). There is a very limited database on the mechanical properties of irradiated ODS steels. Further pre-irradiation and post-irradiation testing of these ODS steels is either planned or in progress.

- Fuel clad chemical interaction: Some coupon-level tests have been performed to analyse FCCI interactions with T91, HCM12A and some ODS alloys. More testing at coupon level is needed, followed by component-level testing.
- Fabrication and joining: It is possible to fabricate and join tubes and ducts of T91 and advanced ferritic steels. MA-957 has been produced in tube form and weld methods to seal tubes with end caps have been developed. Large heats of advanced ODS alloys, such as 14YWT are being developed.
- Corrosion resistance: Some testing has been performed on the corrosion resistance of T91 in sodium, showing some decarburisation at temperatures above 600°C. Additional sodium corrosion data are needed on advanced F/M steels and ODS alloys.
- Reprocessing: Data show that reprocessing becomes more difficult with lower chrome alloys. More data are needed on these advanced alloys.
- Thermal conductivity is expected to be stable as long as swelling levels greater than 1% are observed. Thermal expansion is nearly independent of irradiation dose/damage.

Thus, given the lack of data from component testing and the lack of high-dose data, the TRL level for T91 is 3 (experimental demonstration of critical function) to 4, while the TRL level for the advanced F/M alloys is 3 because of the lack of irradiation data and the advanced ODS alloys, which lack high dose irradiation data and presently lack the ability of making large-scale components, is at a TRL of 2 to 3.

3.5 United States – structural materials

Improved structural material performance is one way to improve economics of fast reactors, by potentially allowing both higher operating temperatures (and thus, higher thermal efficiency and power output) and longer lifetimes (reduced replacement costs). Improved material reliability could also result in reduced down time. Superior structural materials will also spur improvements in high-temperature design methodology, allowing for more flexibility in construction and operation. Advanced materials can have a significant impact on capital construction, even if raw material costs are higher than the reference Type 316 stainless steel. Improved materials performance also impacts safety through improved reliability and greater design margins.

3.5.1 Requirements for SFR structural materials

There are many requirements for all nuclear reactor structural materials, regardless of the exact design or purpose. All requirements for a material's use in an advanced fast reactor system must be considered and carefully weighed. The requirements for use as structural material components in SFRs might include:

- Material availability: enough raw material must be readily available.
- Fabrication and joining: the material must be readily made into components.
- Economics: the cost of the material, as well as fabrication costs, must be considered.
- Long-term stability: in this application, the components must be stable throughout the reactor lifetime of 60 years at temperatures up to 550°C.
- Mechanical properties: strength, ductility, creep resistance, fracture toughness and fatigue resistance must all be considered. Any advanced material must be at least

as strong as Type 316L (low-carbon) stainless steel at all proposed operating temperatures.

- Thermal properties: thermal conductivity and thermal expansion are important properties.
- Neutronic properties: low neutron absorption is desired.
- Corrosion resistance: the material must be tolerant of the reactor coolant, with minimal corrosion rate.
- Radiation resistance: the material must be tolerant of radiation damage up to 5 to 10 dpa, at temperatures ranging from 400 to 550°C.
- Code qualification status: the material must be able to pass Nuclear Regulatory Commission (NRC) licensing considerations.

In addition to these reactor-driven requirements, other factors must be considered during the selection process. For example:

- Depth of database: the degree of industrial experience with a material will determine the amount of development and qualification work required.
- Potential for improvement: a commercial alloy that has possibility for further improvement is more desirable than an optimised alloy with similar performance.
- Use in other reactors: the use (or considered use) of an alloy in a different reactor system will provide opportunities for collaboration and reduce the burden to any one development programme.

Only through careful evaluation of all factors and a thorough trade analysis will the most promising candidate materials be chosen for further development. It is important to note though, that there is no ideal material for each of the considerations listed. Indeed, all candidate materials have advantages and limitations. The most promising alloys, which allow the best performance, are also the least technically mature and will require the most substantial effort. These trade-offs must all be weighed carefully.

3.5.2 Materials selected for development in the United States for SFR applications

All classes of materials have been considered for further development by a group of materials experts from five different national laboratories and five leading universities. The key criteria for alloy selection were improved strength and creep performance, as these properties will have the largest impact on reactor performance. Other materials properties were also considered, as were historical reactor use, interest in other development programmes and extent of development required for fast reactor service. Composite materials, refractory metal alloys, ODS steels, and superalloys were all considered for further development. None of these material classes were selected for further development due to issues with inappropriate environments in SFRs, joining and manufacturing difficulties, irradiation effects, and/or economic reasons.

Austenitic stainless steels were also discussed and chosen for further consideration. These steels are well-established and proven fast-reactor structural materials. Advanced austenitics, such as Ti-stabilised 316 stainless steel, D9 (15Cr-15Ni-Ti), HT-UPS, and NF709, all offer improved performance (strength and creep) over Type 316 stainless steel. Limitations include greater swelling rates than F/M steels at high fluences, although this may not be a critical factor for most structural applications. More advanced austenitics have greatly improved mechanical performance over traditional materials, making them an attractive option.

F/M steels are another well-established and proven fast reactor material. These steels offer very low swelling and better thermal properties when compared to austenitic steels.

They are of high interest in all reactor applications. Disadvantages include a lower strength at elevated temperatures than precipitation-hardened (PH) or austenitic alloys, although more advanced steels are comparable.

Four alloys were identified for further development as part of this programme. These include two F/M steels (NF616 and NF616 with special thermo-mechanical treatments) and two austenitic stainless steel alloys (HT-UPS and NF709). Common F/M and austenitic stainless steels, such as HT9 and Type 316 stainless steel, respectively, are traditional and proven materials for SFRs. All the selected alloys offer considerable improvements in strength and creep resistance over these more mature steels, and yet maintain other critical properties at the same level.

3.5.3 Development needs for advanced alloys

While each of the candidate alloys offers improvements over traditional materials, there are distinct development needs for each of the candidate alloys. Each of these areas will require research and testing to validate these advanced alloys for use in nuclear reactor applications. Development activities have been initiated in many of these areas. The research needs can be categorised into four major groups. These include:

Base development

The first and most basic step of alloy development is the procurement of alloys for testing and an initial evaluation of their properties. There is already an industrial base for the four candidate alloys selected, so testing will be confirmatory rather than exploratory in nature. As a result, basic alloy development is of critical importance and is underway. This initial effort will include procurement of test heats for the NF616 and HT-UPS candidate alloys. This task will also provide optimisation of thermo-mechanical treatments and application to the test alloys. Finally, this task will include assessment of basic physical and mechanical properties of both candidate alloys.

Mechanical testing

The evaluation of mechanical properties is a key step in validating improved materials for reactor service. Given that each alloy has some industrial base, mechanical testing requirements are not as high as for alloys being developed from first principles. As noted above for each alloy, fracture toughness, impact, and creep-fatigue testing are the most significant needs. This testing is now being initiated.

Environmental effects

One of the most complex issues facing qualification of structural materials for service in fast reactor applications is degradation due to environmental effects. The long lifetimes expected of structural components at the elevated temperatures require extraordinary stability. Further, interactions of the steel with the reactor coolant (in this case sodium) and neutron field can have significant and negative effects on material performance. Key environmental testing has been initiated and may include evaluations for thermal ageing and neutron irradiation of both base alloys and weldments. Coolant compatibility testing will be initiated as part of an open competition award that will be co-ordinated with this effort.

Code qualification

Past NRC reviews of the Clinch River Breeder Reactor (CRBR) and PRISM designs have identified concerns in over 20 reactor areas and nine key issues in materials and design methodology. Today, code qualification and licensing concerns remain for traditional reactor materials (such as Type 316 stainless steel), which must be resolved for NRC approval. New advanced materials must also be qualified. Past experience has indicated that material test programmes have often concentrated on simple tests, neglecting more complicated issues such as environmental effects, ageing, discontinuity, and complex loading. Time-dependent effects of environments (sodium and neutron irradiation), thermal ageing, creep, creep-fatigue, and creep-ratcheting can be life-limiting factors in component design. For newer, advanced alloys with improved strength properties and creep resistance, these time-dependent material issues are more pronounced as operational experience is reduced. Increased design life also requires significant efforts in developing models and predictive capabilities. The development of a mechanistic understanding of critical material issues and the development of predictive models will not only optimise/minimise testing efforts, but also facilitate sound structural design and analysis. A co-ordinated development programme has been initiated to address both material needs and the need for high-temperature design methodology.

3.5.4 Final remarks and conclusions

The materials needs for US SFRs have been summarised into core materials (clad and duct) and structural materials. For core materials, the critical needs are centered on developing and testing materials with radiation tolerance to greater than 200 dpa, while being resistant to FCCI and corrosion and still being weldable and able to be processed into component form. Prime candidate materials are F/M steels (HT9, T91 and NF616) and ODS-strengthened ferritic materials. For structural materials, the primary driver is to improve the economics of the reactor. Thus, improved materials can decrease the costs of a reactor through enabling higher operating temperatures and increasing the lifetime of components. Many factors must be considered when selecting new materials, including availability, fabricability, cost, mechanical/corrosion/thermal/ neutronic properties, radiation resistance and code qualification status. Primary materials are F/M steels (leading candidate is NF616) and austenitic steels (leading candidate is HT-UPS).

4. Lead-cooled systems: LFR and ADS

4.1 Belgium

4.1.1 Introduction

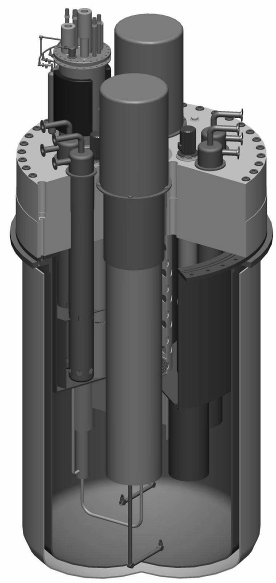
The deployment of innovative nuclear systems relies primarily on the development of structural materials able to withstand operational conditions in normal, abnormal and accidental circumstances, without jeopardising safety. One such system, the MYRRHA project [1], is under extensive investigation at SCK•CEN Mol in Belgium. MYRRHA is an ADS under development at the SCK•CEN to provide protons and neutrons for various R&D applications. It also serves as a basis for design of the European XT-ADS and Fast Spectrum Transmutation Experimental Facility (FASTEF). The MYRRHA project started in 1997, aiming to be operational around 2020.

MYRRHA aims to offer similar capabilities as other materials test reactors (such as the BR2), but also enables transmutation studies. Moreover, the high fast neutron and proton flux allow other experimental conditions that are important for both Generation IV and fusion reactors.

The main characteristics of MYRRHA, in particular the accelerator and reactor core (shown schematically in Figure 9), are:

- proton beam energy: 600 MeV;
- accelerator current: 2.5 mA;
- spallation target: LBE;
- core neutron multiplication factor $k_{eff} = 0.95$;
- core thermal power: 57 to 85 MWth;
- fuel: $(U\text{-}Pu)O_2$ MOX;
- coolant: LBE;
- coolant temperature:
  - inlet: 270 to 300°C;
  - outlet: 400°C.

Figure 9: MYRRHA reactor core design



4.1.2 Selection of candidate materials

For the selection of candidate materials for MYRRHA, the option has been taken to consider industrially available and qualified materials, rather than develop and optimise innovative materials to fulfill the plan for starting operation of MYRRHA around 2020.

Although MYRRHA is conceptually designed to an advanced level, the final choice for structural materials is not yet fully decided. Some structural materials are, however, under extensive investigation and their performance will have to be confirmed. Continuous interaction between designers and material scientists is required to better assess all potential effects that might lead to component failure and that will be part of an overall safety assessment approach. Moreover, this interaction is important to identify missing data required to estimate the lifetime of these components.

One of the critical challenges of the MYRRHA structural materials is their compatibility with the coolant, namely with the liquid metal LBE under intense neutron irradiation (see Table 8). Because of the limited experience on the compatibility of structural materials with heavy liquid metals such as LBE, an extensive R&D programme is being launched at SCK•CEN to provide not only reliable data to the designer, but also support the choices that will be made. The experimental programmes include irradiation of structural materials in LBE and further characterisation under various conditions.

At this stage, three candidate materials were selected as potential structural materials for critical parts of the MYRRHA components. These include austenitic stainless steel (Type 316L stainless steel), Ti–modified 15%Cr-15%Ni austenitic stainless steel (such as DIN 1.4970, D9, AIM1), and the modified F/M 9%Cr-1%Mo steel (T91).

Table 8: The candidate structural materials for MYRRHA

Components	Replaceability	Maximum temperature (normal operational conditions)	Maximum damage dose	Candidate material
Beam window (optional design)	Yes	500°C (Tentative)	40 dpa 5-15 appm He/dpa	T91
Fuel cladding	Yes	500°C	50 dpa	15-15Ti (first core)
				T91
In reactor components	Yes/No	430°C	up to 35 dpa/year	T91, Type 316L stainless steel, 15-15Ti
Reactor vessel	No	370°C	<0.1	Type 316L stainless steel

The materials qualification programme for MYRRHA shows many commonalities with corresponding R&D activities for other nuclear systems, such as the GEN IV LFR [2]. The structural materials issues to be considered are related to the design and safety requirements, where gaps and open questions concerning material performance under representative conditions (e.g. high temperature, high radiation damage doses and aggressive liquid metal environment) are identified. These issues can be classified into six main areas:

- workability and fabricability of materials and components;
- mitigation of liquid metal corrosion;
- degradation of mechanical properties as a result of the interaction of materials with the environment (LME, corrosion fatigue, corrosion creep, SCC, etc.);
- irradiation effects (embrittlement, creep, swelling, etc.);
- long-term effects combining thermal, mechanical, physico-chemical and irradiation conditions representative for the considered machine;
- transferability of the experimental results (laboratory-scale data) to the actual MYRRHA machine.

4.1.3 Workability and fabricability of materials and components

The fuel cladding material is of critical importance, both from economic and safety viewpoints. In the DEMETRA (DM4) domain of the Euratom FP6 integrated project EUROTRANS, studies have been performed to select suitable cladding materials and demonstrate the feasibility of fabricating and qualifying the fuel cladding for MYRRHA/XT-ADS [3]. The fabrication of tubes with relevant dimensions for fuel cladding from commercially available wrought products has been demonstrated in collaboration with the Institute of Physics and Power Engineering (IPPE) in Obninsk, Russian Federation.

The development and qualification of suitable joining and welding technologies of reference F/M steels is another crucial issue, especially for cladding tubes. The investigation of different weld techniques of F/M steels, including dissimilar metal welds for different applications, is a key issue for the applicability of these materials. A first assessment of conventional welding methods, such as tungsten inert gas (TIG) and electron beam (EB) has demonstrated that these could be the weakest part of the structure if not done appropriately [4-5]. The activities envisaged are welding of the claddings and of components made of F/M steel. Fusion welding technologies, such as EB, TIG and laser welding, will be investigated for their applicability to F/M steel dissimilar welds.

4.1.4 Mitigation of liquid metal corrosion

Corrosion of metallic materials by liquid lead-bismuth alloys introduces several challenges for the design and operation of MYRRHA. Oxidation of steel by dissolved oxygen limits the component lifetime through wall thickness reduction and consequently heat transfer degradation (in case of oxide layer formation), especially for cladding and heat exchanger tubes. Eventual dissolution of steel components by the liquid metal typically reduces the component's lifetime. Indeed, dissolution occurs in the absence of an oxide layer at higher rates than oxidation. On the other hand, oxidation forms a diffusion barrier and slows down the corrosion process. The dissolution mode of liquid metal corrosion potentially causes circulation problems by deposition of corrosion products at cold spots.

Considering the operation conditions of MYRRHA, corrosion behaviour of the selected materials may be acceptable, provided that an active oxygen control is ensured to stabilise the passive film on the materials. Although some data on corrosion rates in heavy metals are available in literature, together with some models [6-9], they should be completely assessed and verified in representative conditions to design and optimise any corrosion control system for complex installations.

4.1.5 Degradation of the mechanical properties by interaction with the environment

Regarding environmental effects on the mechanical behaviour of materials, several studies on fracture toughness, fatigue, creep, SCC and liquid metal embrittlement in lead-bismuth environments, have been carried out under the past and ongoing Euratom framework programme projects [10,11] and in support of the MEGAPIE international initiative; the results are summarised in the recently published Handbook by the OECD [12]. Unfortunately, the available information is limited and often not suitable for design purposes.

The vast majority of experimental data on liquid metal embrittlement studies of T91 in contact with LBE concerns tensile test results. The lack of reproducibility and wide scatter of these results are attributed to variability in specimen preparation, more specifically the wetting or intimate contact between liquid LBE and base metal and to the testing conditions (strain rate, oxygen concentration).

Further work will need to consider the effect of various parameters on fracture behaviour (both time independent and time dependent fracture modes) of steels, especially F/M steels in a more systematic way. Preliminary results indicate that the reduction of fracture resistance is not dramatic in LBE, but should be taken into account to preserve required safety margins on components throughout their service life.

4.1.6 Irradiation effects

A number of components in MYRRHA will be exposed to high neutron flux while in contact with liquid LBE. In particular, structural materials close to the spallation target (cladding, diaphragm, core support plate, etc.) will be exposed to both thermo-mechanical loading and neutron and proton irradiation. It is therefore important to evaluate both their dimensional stability and structural integrity.

Few irradiation programmes have been launched at SCK•CEN to examine the post-irradiation behaviour of the selected materials. In the BR2 reactor, specimens of T91 and Type 316L stainless steels were irradiated in the presence of stagnant LBE, as a part of the TWIN-ASTIR programme [13]. The first test campaign, including mainly slow strain rate tests of T91, Type 316L stainless steel, and a few T91/T91 and T91/Type 316L stainless steel joints in both LBE and argon environments, were successfully achieved. The test campaign was temporarily interrupted to resolve the safety issue related to polonium.

Since the initial question of the TWIN-ASTIR experiment (whether irradiation hardening would increase the structural material's susceptibility to liquid metal embrittlement) is still unresolved due to a complete lack of data on material irradiated in contact with LBE at temperatures between 200 and 350°C, it is clear that an irradiation campaign in contact with LBE at low temperature is of the utmost importance. Since the liquid metal embrittlement effect is expected to be maximised in a region of temperature between 200 and 350°C, while the corrosion rate increases with increasing temperature, the most interesting irradiation temperature in contact with LBE would still be 350°C. Within the FP7-GETMAT programme, the LEXUR II irradiation experiment is planned. The experiment will be in a fast neutron spectrum in the BOR-60 reactor, specifically irradiating T91, Type 316L stainless steel and 15-15Ti in the presence of stagnant LBE up to 8 and 16 dpa at 350°C.

4.1.7 Long-term effects of combined thermal, mechanical, physico-chemical and irradiation conditions representative for the considered machine

In addition to irradiation-induced modifications of dimensional and mechanical properties, the synergistic effects of irradiation, mechanical conditions and corrosion due to LBE coolant are a major concern. Since the information concerning these effects is missing, the development of relevant test methods is foreseen. These test methods will be used for further reactor experiments.

4.1.8 Transferability of experimental results to the actual MYRRHA machine

The number of variables in a complex system such as MYRRHA is large and all possible combinations cannot be addressed only by experimental programmes (due not only to lack of time and funding, but also of appropriate experimental facilities capable of reproducing the anticipated conditions in MYRRHA).

There are major differences between a laboratory test specimen and a real nuclear component. In particular, testing conditions are far away from operational ones. Therefore, it is important to develop interpretation tools capable of translating laboratory test results into information relevant to the component. Such interpretation tools can be developed only if basic understanding of the phenomena is available. Therefore, modelling plays a central role in the perspective of accurately assessing the nuclear component lifetime.

Furthermore, numerical tools for simulating behaviour of materials under irradiation and in contact with liquid metals are being developed, for example in the framework of the FP7-GETMAT. These models are based on a correct and quantitative understanding of the fundamental physical mechanisms leading to changes that materials undergo during operation. Thus, although their full development is a long-term objective, they are already of use to guide interpretation of partial experimental results and eventually help to safely extrapolate them to real complex operation conditions, such as those expected for MYRRHA, thereby supporting designers and safety authorities. The continuous improvement of these tools is therefore regarded as part of the effort of qualification of materials intended for use in extreme conditions.

4.1.9 Closing remarks

Examination of the test results, as well as other experimental data from the references show that irradiation and testing conditions have a significant effect on test results. In particular, in the absence of testing guidelines and procedures, it is difficult to provide a reliable database. Therefore, an extensive experimental programme in the un-irradiated condition is being launched to identify critical parameters and provide testing and evaluation guidelines that will be systematically used in the future. It is important to know that there are no standards for the kind of tests performed worldwide and as a matter of fact, specimen size and configuration, specimen preparation, testing conditions (loading rate for example), control of the environment (e.g. oxygen content in LBE), and result evaluation and interpretation differ from one author to another. It became obvious and critical to definitely examine the validity of tests performed, in order for them to be transferable to nuclear components and allow structural integrity calculations.

Finally, to summarise, the materials scientists should provide designers with reliable databases of potential structural materials and a physically-based understanding of underlying degradation mechanisms, as well as interpretation tools that can be used to integrate such information into a more global approach related to the safety assessment of nuclear systems.

4.1.10 References

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4.2 Italy

4.2.1 Operational conditions

Table 9 shows anticipated operating conditions for a lead-cooled fast reactor – European Lead-cooled SYstem (ELSY).

Table 9: Operational conditions of LFR

Coolant	Pure lead
Core outlet coolant temperature	480°C
Fuel residence time	5 years
Maximum burn-up	100 MWd/Kg (10%)
Maximum allowed coolant velocity	2 m/s
Maximum allowed clad temperature	560°C
Peak clad damage	100 dpa
Maximum allowed Von Mises stress	146 MPa

4.2.2 Candidate cladding materials

The maximum burn-up is mainly limited by cladding resistance to the pressure of released fission gas and the fuel cladding mechanical and chemical interaction. At the first stage of the ELSY design, the goal is to fix the realistic limit of the aimed discharge burn-up at 100 MWd/Kg, to achieve the hottest assembly. Possible candidate materials are F/M steels and austenitic steels.

F/M steels show a lower swelling rate and embrittlement under irradiation at T > 350°C, and higher resistance to dissolution in the oxygen-free lead and LBE, compared to austenitic steels. However, they have a higher corrosion rate in the presence of oxygen. Certain Russian steels alloyed with 1 to 2 wt% silicon show very good corrosion resistance in LBE and lead coolant flow under optimised oxygen controlled conditions.

Existing experience with operation of liquid metal-cooled fast reactors (LMFR) and the performed irradiation studies of cladding materials for fast spectrum reactors demonstrate that optimised AUS, known as AIM1 (15-15 Ti mod Si) in French standard, can withstand typical LMFR operation conditions (sodium flow up to 7 m/s, temperature of 400 to 550°C) up to the damage dose of 115 dpa. At these temperatures, F/M steels with 8 to 12% chromium (such as T91, EM-10, HT9, F82H, etc.) are even more resistant; they show the same creep rate at about 200 dpa. At this stage, T91 is proposed as the first candidate for the cladding material of ELSY, taking into account its better irradiation resistance and ongoing R&D on the technology of its protection against corrosion.

4.2.3 The corrosion problem: coated materials

T91, as well as austenitic steels, is known to form a protective oxide layer in lead and lead-bismuth when exposed to these coolants at controlled oxygen content (approximately 10^{-6} weight percent). Recent tests in pure lead and former experiments in lead-bismuth demonstrated that at a temperature of 550°C the oxide layer becomes brittle and its protective action is lost. This fact seems to prevent the use of T91 without any surface treatment for fuel claddings. It was demonstrated that diffusion of aluminum on the surface favours the formation of a stable oxide layer on T91, even at high temperatures. Specific tests in lead are presently underway on T91 specimens, coated by different techniques. Corrosion tests are running in lead at 550°C and 10^{-6} weight percent oxygen.

4.2.4 Other candidates: oxide-dispersed steels

An innovative alternative to prevent corrosion of lead at high temperature seems to be represented by the selection of ODS steels. These steels, produced by powders metallurgy, thanks to their content in chromium and yttrium oxides, are able to form a very stable oxide barrier against corrosion by lead. Testing of both 9 Cr and 14 Cr ODS steels with the beneficial dispersion of Y2O3 is foreseen.

4.2.5 Foreseen irradiation tests in Russian BOR 60

Since it is expected that the corrosive effect, accompanied by material embrittlement, could act in a synergic way with the irradiation damage typical for the fast spectrum, a systematic experimentation considering both chemical exposure and the irradiation factors has been planned. An irradiation campaign in the BOR 60 reactor under fast spectrum in lead environment is under preparation. The experiments will associate environmental effects due to the lead at 550°C and fast spectrum irradiation effects. The changes of mechanical properties will be measured after exposure.

The materials to be tested are:

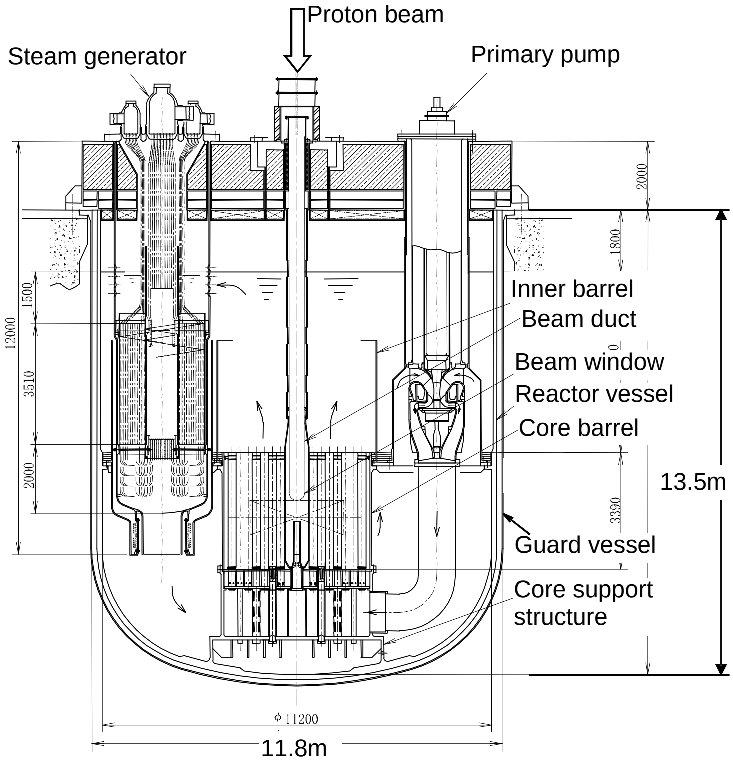
- Type 316 L stainless steel;
- coated Type 316L stainless steel;
- T91;
- coated T91;
- 15 -15 Ti steel;
- 9 Cr ODS;
- 14 Cr ODS.

4.3 Japan

Figure 10 shows the conceptual design of JAEA's reference design for an ADS. The JAEA reference ADS currently only has a conceptual design, so detailed requirements for the structural materials have not been indicated. The major reactor parameters are listed below.

- proton beam: 1.5 GeV;
- spallation target: lead-bismuth;
- coolant: lead-bismuth;
- maximum $k_{eff} = 0.97$;
- thermal output: 800 MWth;
- minor actinide initial inventory: 2.5 tonnes;
- fuel composition: (MA +Pu)N + ZrN, where MA indicates minor actinides;
- transmutation rate: 250 kg-MA/year;
- 600 effective full power days (EFPD), 1 batch.

Figure 10: Conceptual design of JAEA's reference ADS



Neutrons generated by the 1.5 GeV proton beam induced to the spallation target (lead-bismuth) burn the nitride fuels (MA+Pu)N+ZrN) in this reactor. Steam is generated by the SG in the RPV. Initial MA inventory is 2.5 tonnes and 250 kg of MA are transmuted yearly. MA-containing fuel is exchanged after 600 EFPD. Thermal output of the reactor is 800 MWth and predicted reactor lifetime is 40 years.

Major components of the reactor are the beam window, fuel cladding, in-reactor components (inner barrel, core barrel etc.) and RPV. The design parameters of JAEA's reference ADS are listed in Table 10. 9Cr martensitic steel is the first candidate material for these components because of compatibility with the LBE. Martensitic steels, such as T91 and F82H (Japanese reduced activation martensitic steel for fusion reactor structural material), are considered first candidate materials. Austenitic stainless steels, such as Type 316 stainless steel and JPCA (Japanese Ti-stabilised Type 316 stainless steel for fusion reactor), are the second candidate materials.

Table 10: Design parameters of JAEA's reference ADS

Components	Replaceability Y=yes N=no	Coolant	Maximum temperature	First and second candidate materials
Beam window	Y	Pb-Bi Inlet: 300°C	500°C (tentative)	First: Mod.9Cr-1Mo(T91) (or F82H) Second: Type 316 stainless steel (or JPCA)
Fuel cladding	Y	Pb-Bi Inlet: 300°C Outlet: 410°C	580°C; design modification to reduce this value is underway	First: Mod.9Cr-1Mo(T91) (or F82H) Second: Type 316 stainless steel (or JPCA)
In-reactor components (inner barrel, core barrel, etc.)	Y/N	Pb-Bi Inlet: 300°C Outlet: 410°C	410°C	First: Mod.9Cr-1Mo(T91) (or F82H) Second: Type 316 stainless steel (or JPCA)
Reactor vessel	N	Pb-Bi Inlet: 300°C Outlet: 410°C	410°C	First: Mod.9Cr-1Mo(T91) (or F82H) Second: Type 316 stainless steel (or JPCA)

The beam window is used in the most severe condition. Irradiation damage is estimated at 98 dpa, accompanied with 14 000 appm helium and 1 900 appm hydrogen in the case of JPCA after 300 full power days (FPDs) [1]. Major material issues are (1) compatibility of structural material with LBE and (2) irradiation behaviour of the beam window. Investigation of the compatibility of structural material with LBE is ongoing in the JAEA. An irradiated material property database for F82H steel has been prepared by the fusion programme and contains irradiation data for Type 316 stainless steels and JPCA. However, the beam window can be replaced after a certain period of use.

4.3.1 Reference

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4.4 Switzerland

4.4.1 Introduction

Developing spallation targets with either high power or high proton beam intensity for different applications, such as neutron scattering science and nuclear waste transmutation devices [including accelerator-driven systems (ADS)], is of high interest at the Paul Scherrer Institute (PSI). ADS-related materials research has been conducted by the Spallation Neutron Source Division since the mid 1990s. Research activities are mainly focused on two important topics: radiation damage and liquid metal embrittlement effects in materials. For radiation damage related studies, five irradiation experiments have been carried out in the SINQ Target Irradiation Programme (STIP) under high-energy proton and spallation neutron mixed spectrum, which is nearly the same as that foreseen in a spallation target of an ADS. F/M steels and austenitic steels are the main materials used in STIP. For the liquid metal embrittlement investigations, F/M steels in either un-irradiated or irradiated condition have been mechanically tested in liquid LBE, the coolant for XT-ADS (where XT refers to eXperimenTal) (e.g. MYRRHA).

4.4.2 Radiation damage study

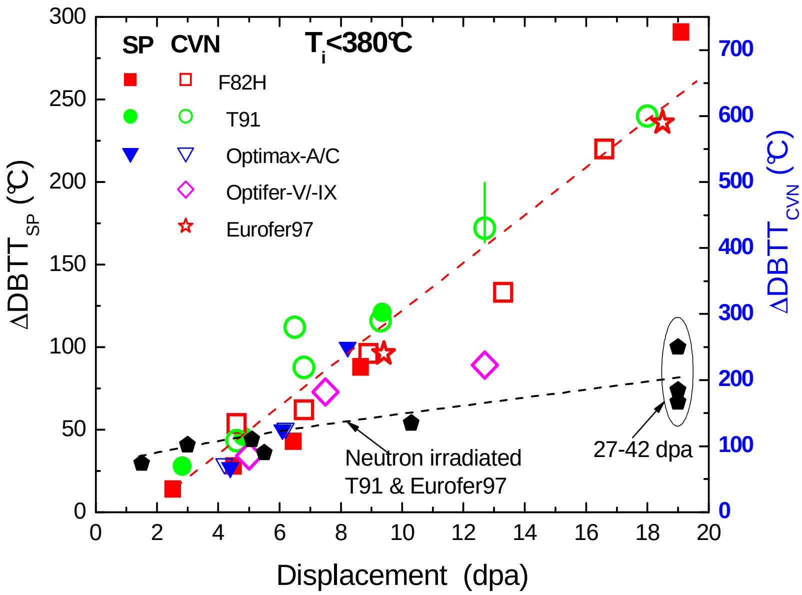
Specimens irradiated in the STIP experiments were placed directly inside the SINQ spallation targets [1]. Therefore, the specimens received intensive irradiation of both high-energy protons and spallation neutrons. As the specimens had to be enclosed in tubes with an inner diameter of about 9.5 mm, the specimens were all miniature types designed for assessing mechanical properties, such as tensile, fatigue, fracture toughness, DBTT and microstructural changes of materials. Because each SINQ solid target was in use for two years, every STIP irradiation experiment also lasted two years. During 1998 and 2008, five experiments were done. In between, in 2006, the MEGAPIE liquid LBE target was irradiated without any test specimens. More than 40 kinds of materials, including iron, nickel, aluminum, zirconium, molybdenum, and tungsten alloys and pure metals such as tantalum, were irradiated in the STIP to doses up to 20 dpa (in steels) at temperatures up to 800°C. Some specimens were irradiated in contact with stagnant mercury, LBE and lead. The components of the MEGAPIE target are the most valuable specimens for studying the synergetic effects of irradiation and LBE corrosion/ embrittlement.

So far, post-irradiation examinations (PIE) performed on STIP specimens are mainly focused on F/M steels and Type 316 stainless steel. The changes in microstructure and mechanical properties have been investigated and published in the Journal of Nuclear Materials [2]. In F/M steels, it was observed that at $T_i \leq 300°C$, the main features were small defect clusters of 1 to 2 nm in size and small dislocation loops of a few nm in size. With increasing irradiation dose, the densities of defect clusters and loops increased and the size of loops increased as well. At $T_i > 300°C$, the densities of defect clusters and loops were found to decrease with increasing irradiation temperature. High-density helium bubbles with an average size above 1 nm were observed in specimens with <~500 appm helium and irradiated at ≤ 180°C. With increasing irradiation dose and temperature, the size of bubbles increased. Tensile tests indicated that specimens irradiated at lower temperatures of <~350°C exhibited tensile properties similar to that obtained from neutron irradiation, while specimens irradiated at higher temperatures of >~400°C demonstrated significant hardening, which is normally not observed in neutron irradiation. This hardening should be induced by helium bubbles. The most striking feature of F/M steels irradiated in the STIP was helium embrittlement effect. Results of Charpy impact tests (Figure 11 [3]) showed that the DBTT increased continuously with irradiation dose or helium content. At high doses, the DBTT shift induced by helium became dominant. Meanwhile, bend tests indicated that the fracture toughness of F/M steels decreased with irradiation dose and could fall below 50 MPa($m^{1/2}$), even at high test temperatures.

The results of austenitic steels are similar to that of F/M steels, except for the DBTT shift. Microstructural investigations showed that the main features of irradiation damage were high-density, small black dot defects and large Frank loops. The density and size of the small dot defects were independent of irradiation dose, with a mean size of 1 to 2 nm and a density of about $2\text{-}5\times10^{23}$ m^{-3}. The density of Frank loops varied little with dose, while size increased with dose. Small bubbles were observed in specimens irradiated to 10 dpa or higher at temperatures above about 300°C. Mechanical tests also revealed more evident hardening and embrittlement of specimens irradiated in the STIP, compared to neutron irradiated specimens.

A conclusion is that the allowable radiation dose for F/M or austenitic steels applied in spallation irradiation environments can be much more limited, as compared to that in neutron irradiation cases.

Figure 11: DBTT shift as a function of the irradiation dose for different F/M steels irradiated in the STIP



Neutron irradiation data are quoted from the literature.

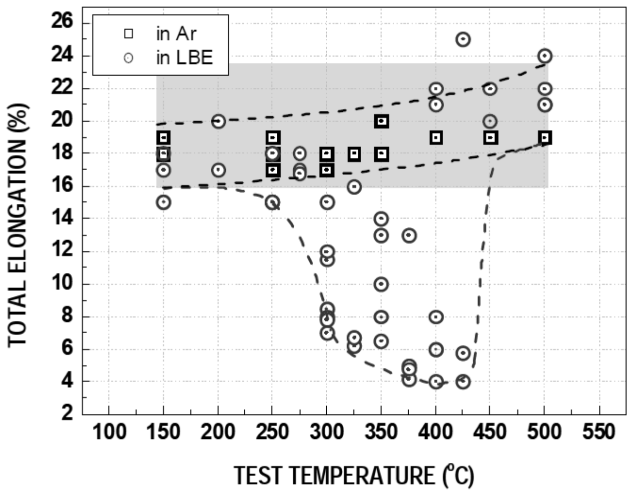
4.4.3 LBE embrittlement study

The aim of the work is: (a) to investigate the effects of LBE embrittlement on the mechanical properties of F/M steels (candidate materials for applications in spallation targets) in various states and (b) to study the mechanisms of LBE embrittlement effects on mechanical properties of F/M steels. Mechanical tests, such as slow-strain-rate tensile (SSRT) tests and three-point bending tests have been performed to characterise the mechanical properties of T91 steel (the representative of F/M steels in this study) and determine the effects of LBE embrittlement on these mechanical properties. Scanning electron microscopy (SEM) observations have been conducted to obtain microstructural information needed to understand the embrittlement mechanisms [4].

SSRT tests on T91 steel in the standard metallurgical (SM) state, tempered at 760°C, revealed that it might encounter LBE embrittlement in the temperature range of 300 to 425°C. A well-defined "ductility trough" for a F/M steel-LBE system was evidenced from the reduction of fracture strain (or total elongation) of T91 F/M steel in this temperature range (Figure 12). SEM observations of the fracture surfaces showed that the fracture mode of specimens suffering embrittlement effects was brittle transgranular cleavage.

SSRT tests on hardened T91 steel, tempered at 500 or 600°C, showed more pronounced LBE embrittlement effects. In comparison to that of specimens tempered at 760°C, the "ductility troughs" of specimens tempered at 600 and 500°C covered a wider temperature range. The results demonstrated clearly that LBE embrittlement effects on the tensile properties of F/M steels were strongly enhanced by hardening. Both the influenced temperature range and degradation level increased with the hardening amount. SSRT tests on T91 and F82H F/M steels irradiated at temperatures in the range of 140 to 500°C and doses up to 20 dpa revealed significant irradiation-induced hardening and embrittlement effects (loss of ductility), as compared to un-irradiated specimens. The SSRT tests performed on irradiated specimens in liquid LBE illustrated that specimens undergo a further reduction in ductility.

Figure 12: Total elongation versus test temperature for HT760 specimens tested either in Ar or LBE [4]



Similar conclusions were obtained from three-point bending tests on the same materials. The results showed that the fracture toughness of un-irradiated T91 steels decreased by 20 to 30% in tests in LBE at temperatures between 200 and 400°C. The LBE-induced reduction in fracture toughness increased with hardening introduced either by tempering at 600/500°C or by irradiation.

Practically, the results indicate that great attention should be paid to application of F/M steels in liquid LBE, since the mechanical properties of F/M steels, such as ductility and fracture toughness, can be substantially degraded by LBE embrittlement effects, particularly since LBE embrittlement is enhanced by irradiation-induced hardening under irradiation conditions.

4.4.4 Conclusions and outlook

Results obtained from materials irradiated in SINQ targets under a mixed spectrum of high-energy protons and spallation neutrons indicate more serious degradation in mechanical properties compared to that observed after irradiation in fission reactors, which can be attributed to helium-induced embrittlement effects. The results show special significance for predicting the behaviours of component materials, particularly the beam windows, in spallation targets. On the other hand, studies on embrittlement effect of LBE suggest that mechanical properties of F/M steels can be strongly affected by the presence of LBE. Moreover, the susceptibly could be increased with irradiation-induced hardening.

However, there is still much room to enrich the database and improve understanding of the mechanisms of degradation of materials properties in a spallation irradiation environment. In the next few years, through PIE of specimens irradiated in the STIP and the MEGAPIE target (a LBE target), these needs will be greatly filled.

4.4.5 References

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4.5 Europe – European Facility for Industrial Transmutation (EFIT)

4.5.1 Introduction

A common objective of partitioning and transmutation (P/T) of high-level waste as discharged from currently operating reactors is to reduce the burden on a final geological repository by reducing the radiotoxicity, volume and heat load of the waste. Possible P/T strategies can range from dedicated transmuters in a separate fuel cycle stratum of a stable or expanding nuclear energy scenario up to the scenario of a nuclear phase-out.

The objective of the EURopean Research Programme for the TRANSmutation of high-level nuclear waste in an ADS (EUROTRANS) is the study of transmutation at industrial scale of high-level waste in ADS. Moreover, to provide a reasonably reliable assessment of technological feasibility and a cost estimate for ADS-based transmutation and to possibly decide on the detailed design of an experimental ADS and its construction in the future, R&D activities have been performed in key technological areas as e.g. accelerator components, fuel, structural materials, thermal-hydraulics, heavy liquid metal technology and coupling experiments.

The more specific objective of EUROTRANS is the design and feasibility assessment of an industrial ADS prototype dedicated to transmutation with the following major activities [1]:

- Carry out a first advanced design of an experimental facility (realisation in a short-term), demonstrating the technical feasibility of transmutation in an ADS (XT-ADS), as well as to accomplish a conceptual design of the European Facility for Industrial Transmutation EFIT (realisation in the long-term). This step-wise approach is called the European Transmutation Demonstration (ETD) approach.
- Provide, for the above devices, validated experimental input (such as experimental techniques, dynamics, feedback effects, shielding, safety and licensing issues) from relevant experiments on the coupling of an accelerator, an external neutron and a subcritical blanket.
- Develop and demonstrate the necessary associated technologies, especially reliable linear accelerator components, fuels, structural materials at medium-to high-temperature and high-radiation exposure conditions, thermal-hydraulics, heavy liquid metal technologies and measurement techniques and nuclear data.
- Prove its overall technical feasibility.
- Carry out an economic assessment of the whole system, in order to start a decision process towards a European demonstration facility.

EUROTRANS has integrated critical masses of resources and activities of 51 participants (industry, national research centers and universities) from 16 European countries and is a five-year project that started in April 2005. The scientific and technical results produced by EUROTRANS are, among others, used for the definition of the Central Design Team (CDT) for a fast-spectrum transmutation experimental facility project [2], which started in 2009 and has the aim to perform a detailed design of an experimental facility with the objective to demonstrate ADS and transmutation feasibility.

Hereafter, the EFIT design will be described. In particular, the conceptual design of an EFIT foresees a core able to transmute the minor actinides (curium, americium and neptunium) at sizable scale.

4.5.2 European facility for industrial transmutation design

EFIT [1] is a subcritical reactor of 400 MWth, cooled by pure lead and driven by a proton accelerator. EFIT is designed as a transmutation demonstrator fuelled with MA with burning capability and as a sub-product, electricity generation.

The EFIT core is made of 180 hexagonal fuel assemblies, with 169 pins (168 fuel pins) each, the active length is 90 cm, the equivalent inner and outer radius are 43.7 cm and 151.5 cm, respectively.

Because EFIT is a hybrid reactor controlled by the spallation neutrons, it must be ensured that the core remains always subcritical, without having to rely on shut-off or control rods. The margin to criticality is an important requirement for the core definition that leads, eventually, to defining core size and reactor power. The operating subcriticality level of $k_{eff} = 0.97$ has been chosen in accordance with the requirement that the reactor core should remain subcritical under all plant conditions, including transients.

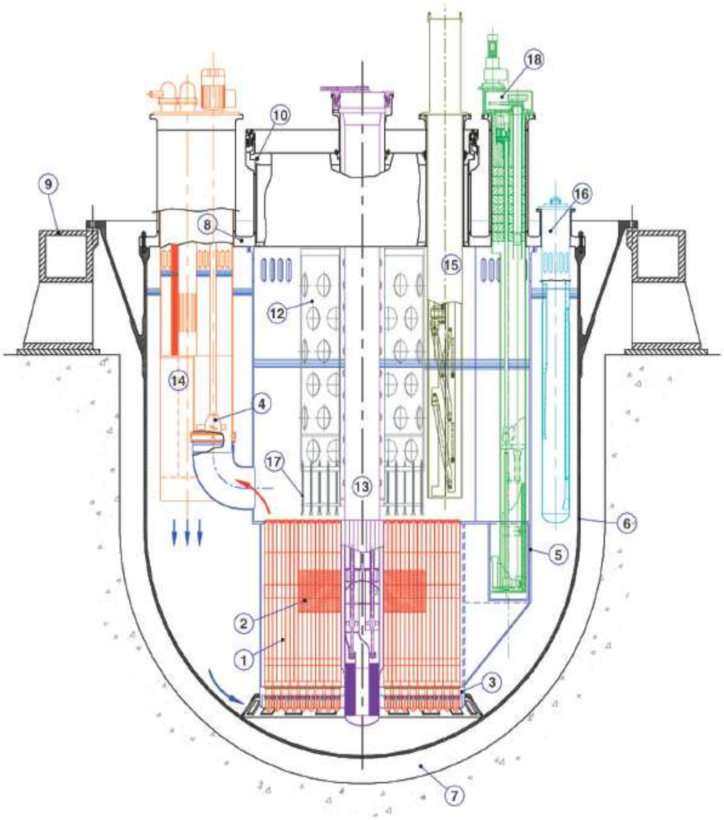
EFIT is fuelled with uranium-free plutonium and MA fuel. The configuration of the primary system is pool-type (Figure 1). The pool design has important beneficial features, including a simple low-temperature boundary containing all primary coolant, the large thermal capacity of the coolant in the primary vessel and a minimum of components and structures operating at the core outlet temperature. The primary coolant is molten lead, which is characterised by higher melting point than LBE or sodium.

The operating temperatures are: 400°C at core inlet (to have sufficient margin from the risk of lead freezing) and 480°C at core outlet. The proposed operating temperatures are, hence, a compromise between corrosion/erosion protection and performance. The speed of the primary coolant is kept low by design (less than 2 m/s), in order to limit erosion.

All primary coolant is contained within the RPV. Key components for reactor operation at power are the primary pumps (PP) and SGs. Four identical groups of these components are provided, each comprising two SGs, with one PP in-between. The lead coolant flow path is illustrated by arrows in Figure 13. The hot lead is pumped into the enclosed pool above the PP and SGs and driven shell-side downward across the SG helical-tube bundles into the cold pool.

The operational conditions of the different components of EFIT are reported in Table 11. For each component, the material validation needs are given with or without oxygen control, as well as with aluminisation coating.

Figure 13: EFIT design and scheme of the primary system



1 Reactor core	10 Rotating plug
2 Active zone	11 Above core structure (is not shown in the figure)
3 Diagrid	12 Target unit
4 Primary pump	13 Steam generator unit
5 Cylindrical inner vessel	14 Fuel handling machine
6 Reactor vessel	15 Filter unit
7 Reactor cavity	16 Core instrumentation
8 Reactor roof	17 Rotor lift machine
9 Reactor vessel support	18 DHR dip cooler

Table 11: Materials and operating conditions for EFIT

Component	Material	Minimum temperature for unlimited time (°C)[1]	Maximum temperature for: Unlimited time (°C)	Maximum temperature for: 1 week (°C)[2]	Maximum temperature for: 1 month (°C)	Maximum lead velocity (m/s)	Maximum neutron radiation damage (dpa)	Maximum mechanical loads (MPa)	Without oxygen control	With oxygen control	With aluminisation
Reactor vessel	Type 316L stainless steel	380	410	440	400	1	3×10^{-4}	109	X	X	
Internal structures	Type 316L stainless steel	380	480	470	420	0.4	2	na		X	
Above core structure	Type 316L stainless steel	380	480	470	420	0.4	na	na		X	
Steam generators	T91	380	480	470	420	0.6	$<10^{-4}$	na		X	
Primary loop circulation pumps	MAXTHAL (Ti_3SiC_2)[3]	380	480	470	420	10	$<10^{-4}$	na			
Fuel assemblies: Clad structures	T91	385 380	555 480	475 470	425 420	1.1 1.5	na na	13 14		X	X
Dummy assemblies	T91	380	480	470	420	0.01	na	na			
Refueling equipment	Type 316L stainless steel	380	480	470	420	0.3	0.4	na			
Decay heat removal heat exchanger	T91	380	400	440	400	0.2	$<10^{-4}$	na			
Purification system	Type 316L stainless steel	380	400	420	400	0.05	$<10^{-4}$	na			
Target structures pumps	T91 MAXTHAL (Ti_3SiC_2)[4]	380 380	570 520	420 420	380 380	1.3 4	180 100	na na			X

[1]All the reported minimum temperatures refer to the refueling condition that is assumed to be performed at 380°C except otherwise stated.

[2]Note that in emergency conditions the values are lower than the maximum temperature for unlimited time.

[3]The selected material for pumps and shafts needs to be confirmed.

[4]The selected material for pumps and shafts needs to be confirmed.

The following description is an excerpt from the DM4 DEMETRA deliverable 4.1 “Candidate Materials for XT-ADS and EFIT, Operating Conditions and Testing Requirements” [3]:

Concerning operating temperatures, three sets of values are provided in Table 11 (Operation at power; DHR via safety system N°1; DHR via safety system N°2). They refer to:

(a) Operation at power: the steady state values for operation at power.

(b) DHR via safety system N°1: decay heat removal in emergency conditions via the dedicated safety systems N°1, crediting three out of four DHR systems in operation (the entire installed heat removal capacity is 4x33 = 130%). For this event, the “stabilised” maximum value, that is the value attained following the short-term plant response to postulated accident, is given. It has to be noted that the “stabilised” value is equal or lower than the corresponding value for operation at power and subsequently it will decrease with time, according to the decreasing decay power. The short-term plant response may transiently lead, for the cladding, to temperature values higher than those in the table. This should be taken into account for the materials requirements and qualifications. The DHR N°2 is called in operation during emergency conditions and when the DHR N°2 is not available, hence these plant conditions are expected to last for no more than one week during all plant life.

(c) DHR via safety system N°2: decay heat removal in emergency conditions via the dedicated safety systems N°2, crediting three out of four systems in operation and the primary pumps not operating (the entire installed heat removal capacity is 4x33 = 133%). Also for this event, the “stabilised” maximum value that is the value attained following the short-term plant response to a postulated accident is given. Also during operation with DHR N°2 the temperatures remain lower than the operation at power. The DHR N°2 is the preferred system called into operation during emergency conditions; hence, these plant conditions are expected to last for no more than one month during all plant life.

The data reported in Table 11 are extracted or extrapolated from the reports prepared in the frame of the EUROTRANS project [4-7].

No data are provided for decay heat removal via the non-safety grade systems (i.e. via the secondary system in a controlled way), since the associated temperatures will be less than the values corresponding to operation at power. This mode of decay heat removal will, however, be the prevailing one; this is reflected in the specified expected duration of conditions (b) and (c) during the entire plant life.

4.5.3 Identification of critical components and reference materials selection

The critical components of the EFIT are given below.

Reactor vessel

The selected material for the main vessel is Type 316L stainless steel. The minimum temperature is 380°C and the maximum operating temperature is 410°C. The maximum mechanical load is 109 MPa. The maximum lead velocity is 1 m/s. It should be noted that the maximum values here indicated are not spatially coincident. More detailed temperature and stress distribution can be found in reference [7].

Internal structure

The selected material for internal structures is the same as that of the RPV (Type 316L stainless steel). The maximum operating temperature is higher than the RPV (480°C). The maximum lead velocity is 0.4 m/s.

Above core structures

The selected material for the ACS is the same as that of the RPV (Type 316L stainless steel). The operating temperature is the same as that of the internal structures.

Refueling equipment and purification system

The selected material for these components is the same as that of the RPV (Type 316L stainless steel). The operating temperature range is the same as that of the internal structures.

Primary pumps

The PP suitable material has to be defined. Type 316L stainless steel could be a candidate, except for the component parts with high-level velocity, for which tentative candidates could be the MAXTHAL (Ti_3SiC_2) or Type 316L stainless steel with surface treatments (aluminisation, laser peening, etc.). The operating temperature range is the same as that of the internal structures, in some component parts; it is associated with a high lead velocity of 10 m/s.

Fuel assembly

The selected material for the fuel assembly is T91. The operating temperature under irradiation is 555°C for the cladding and 480°C for structures. Fuel cladding and fuel assembly structures operate under higher neutron flux and the value is not yet available. The maximum lead velocity is 1.1 m/s for the cladding and 1.5 m/s for the structures.

Dummy assembly

The selected material for these components is the same as that of the fuel assembly (T91). The operating conditions are bounded by the fuel assembly.

Heat exchanger (steam generator)

The selected material for these components is the same as that of the fuel assembly (T91). The operating conditions are bounded by the fuel assembly.

Target

The selected material for the target structures is the same as that of the fuel assembly (T91), yet a suitable material for the target pumps has to be defined (component parts with high-level velocity, tentatively candidates could be the MAXTHAL-Ti_3SiC_2 – or the Type 316L stainless steel, with surface treatments such as aluminisation, laser peening, etc.). The operating temperature under irradiation is 420 to 570°C for the target structures and 380 to 520°C for the target pumps. Target structures and pumps operate under higher neutron flux and can reach a neutron damage level of 180 dpa. The maximum lead velocity is 1.3 m/s for the target structures and 4 m/s for the target pump.

4.5.4 Rationale of materials selection for EFIT components

The critical parameters for each main component of the EFIT have been identified. Moreover, general trends on materials studies have already shown that within the low-temperature range an associated high dose embrittlement issue might be present. On the other hand, high temperature associated with high lead velocity is critical for the corrosion issue. Combined effects of lead corrosion and high neutron flux irradiation need to be considered as well. The components that will have, during operation, the highest level of irradiation damage are the structures of the spallation target and the core components (i.e. cladding tubes of the fuel element and other structures like wrapper). It is well known that irradiation induces specific effects, such as defects production (displacement of atoms, dislocation loops, and voids), phase instability (dissolution, precipitation, segregation) and changes in chemical composition (induced by spallation or transmutation reactions). These important modifications in the material microstructure induce degradation of mechanical properties, which will determine the lifetime of different components. Besides irradiation, other important contributions to the lifetime of components come from the presence of flowing liquid lead or LBE. Corrosion mechanisms and embrittlement phenomena can further limit the performance of structural materials. Moreover, the operating temperature is a very important factor that determines the materials behaviour (irradiation effects are strongly dependent on irradiation temperature). Past experimental programmes made evident that irradiation effects on structural materials are a critical issue in the low-temperature range. The irradiations, under neutron spectrum or proton/neutron mixed spectrum, showed that strong embrittlement and hardening occur with decreasing irradiation temperatures and a drastic degradation of tensile properties of 9Cr martensitic steels at low operating temperatures. The selection of a windowless target will relax the issue of damage related to proton irradiation. However, the effect of irradiation and high-temperature corrosion should not be neglected for target structures.

The preliminary selection of structural materials for EFIT has been made by taking into account existing materials and the database information for nuclear applications. The rationalisation of materials selection has been made as follows:

- Aluminum alloys (mostly used in material test reactors; MTRs) would not sustain this relatively high temperature and the irradiation conditions would yield to severe embrittlement of this class of materials.
- Ni-base alloys (mainly used in LWRs) have a high affinity to dissolve in lead-bismuth, in addition to their microstructural instability under irradiation.
- Zirconium alloys (used as fuel cladding in LWRs) have shown drastic loss of strength and ductility under high-temperature irradiation, especially in the presence of hydrogen.
- Austenitic steels (used mainly as internals in LWRs) are very susceptible to irradiation-induced swelling and creep. In addition, they showed a poor resistance to corrosion in liquid lead alloys at high temperatures (above 500°C the corrosion mechanism changes from oxidation to dissolution, regardless of the oxygen activity in the liquid metal). Nevertheless, due to the large database available, especially those of Type 316L stainless steel, austenitic steels are the best candidates for the components serving at relatively low temperatures and low irradiation damage.
- F/M steels (used as wrapper tubes in Phénix, candidate materials for fast reactors and fusion) appear to be promising candidate materials both for fuel cladding and spallation target structures. However, the major problem with the use of F/M steels in an irradiation field is the increase of DBTT and therefore toughness degradation, especially for temperatures lower than 400°C. Moreover, the corrosion mechanism of this class of steels seems to be dependent on oxygen activity in the liquid metal and temperature.

The above list led to the selection of potential candidate materials for the EFIT components as follows:

- Type 316L stainless steel for those components exposed to relatively low temperatures and dose as e.g. the RPV, internal structures, ACS, refueling equipment and purification system.
- T91 for the other components as e.g. the SGs, heat exchangers, core barrel, and fuel assemblies, including the cladding, dummy assemblies, target unit and pump

housing. However, due to the creep-to-rupture resistance of this steel, its use should be limited to ~550°C.

- For higher temperature application, ODS steels would be an alternative. However, materials data on ODS are very scarce and for the short-term realisation of the EFIT, a detailed design would not be practicable. Therefore, R&D activities on ODS are strongly suggested for the long-term development of ADS.

Regarding fuel cladding surface treatments, aluminisation is suggested to overcome high-temperature oxidation/corrosion issues. Regarding the pump impeller used in the primary loop, a suitable material for the LBE or pure lead environment with high operating temperature and high coolant velocity has to be identified (candidate materials are: MAXTHAL, Type 316L stainless steel or T91 with surface treatments, such as aluminisation, laser peening, etc.).

4.5.5 Materials data requirements

Experiments are performed on the proposed candidate structural materials, defining testing parameters while considering the operating conditions of temperature, coolant velocity, neutron irradiation and mechanical loads, which are established on the basis of the EFIT design requirements (see Table 11) to evaluate the:

- long-term oxide layer stability and thickness;
- long-term corrosion/erosion of reference materials exposed to molten lead or LBE;
- mechanical behaviour (fracture mechanics, relaxation-fatigue, stress corrosion cracking (SCC), creep, liquid metal embrittlement, fatigue, tensile) of reference materials exposed to molten lead or LBE;
- proton and neutron irradiation and PIE of reference materials;
- neutron irradiation and lead or LBE combined effect on the corrosion and mechanical behaviour of the reference material;
- high-temperature corrosion barrier development (mainly for cladding material and pump components).

Long-term oxide layer stability of candidate materials and heat transfer components (heat exchanger, SG, fuel cladding) are performed in dynamic liquid metal (lead and LBE). Structural material for heat transfer components, in fluent lead, is protected to avoid excessive corrosion by:

- controlled oxygen environment to form a possibly thin, compact and stable surface layer of oxide for operating temperatures lower than 500°C;
- protective surface treatment, like the German aluminisation treatment Gepulste Elektronenstrahlanlage (GESA), for operating temperatures higher than 500°C.

However, with a view to surface protection methods, structural material could strongly affect heat transfer capability. Therefore, from a design point of view it is necessary to have information such as:

- required initial oxide/coating thickness;
- kinetics for oxide layer build-up, as a function of time and operating conditions;
- conductivity of native material and total conductivity of combination with protective layer (oxide and/or aluminisation);
- behaviour under neutron irradiation;
- range of oxygen activity in the molten lead.

Structural material protection through control of the amount of oxygen in the molten lead or lead-bismuth pool needs tests to define the oxygen activity level/range versus operating temperature. Moreover, it is necessary to gather information on the oxygen and oxidation behaviour during accident conditions (control in this condition cannot be credited), as well as information on the type of oxygen sensors, their required minimum number and their preferred localisation.

Long-term corrosion/erosion tests of the candidate materials are performed in dynamic liquid metal (lead and LBE). Relevant parameters to be investigated are: Tmax and ΔT, oxygen activity of lead, flow rate and materials. Long-term corrosion/oxidation evaluation should be performed by relevant durations of tests, such that experimental results allow understanding materials behaviour over a wide time span. These tests are necessary to estimate time dependency of the corrosion/oxidation process (high temperature) and reaction product deposition (low temperature), at specific oxygen potentials in lead or LBE.

Short-to mid-term corrosion tests with high LBE flow rate, specific oxygen conditions, high temperatures, and temperature ramps are needed to evaluate material behaviour under accidental conditions (temperature increase, high turbulence, etc.). The activities of optimisation and testing of treated surface layers are aimed at validation of a lead or LBE corrosion protection system for fuel claddings and pump components. Compatibility tests are performed to prove the corrosion resistance of the treated surface in the temperature range of interest and after long-term exposure to lead or LBE. Moreover, tests of these coatings in a neutron irradiation field are also mandatory.

Finally, tests should be performed on welded steels and different surface states of the materials.

Mechanical behaviour tests are performed to validate structural material performance in HLM, in irradiation fields and possibly, the HLM/irradiation fields combined effects. Relevant tests are:

- Fracture mechanics tests: These tests are necessary to study engineering, elastic-plastic fracture mechanics and mechanisms for the candidate materials in LBE and lead. To this aim, resilience tests of (KCV) specimens as a function of ageing conditions in lead and LBE are done. Creep crack growth tests in stagnant lead or LBE with different oxygen activities should be performed on specimens with different standard geometries, special attention being paid to the effect of contact conditions (wetting/non-wetting) on the fracture process. Crack growth resistance curves and J resistance curves should be obtained. Constant load creep crack growth tests should be conducted using compact tensile (CT) and O-ring specimens for comparison with irradiated specimens. Fracture toughness data using small specimens (KLST and 1/4TCT) exposed to lead and LBE for short and long periods at different temperatures, should be obtained. The use of small specimens is recommended for comparison with the fracture toughness of irradiated material. Fracture toughness of the candidate materials after exposure to lead-bismuth should also be studied, using different specimen geometries [three-point bend, CT, single edged notched tensile (SENT)]. Specimens should be aged for relevant time periods in lead-bismuth at high temperatures with oxygen control before performing fracture toughness, tensile and Charpy tests. These tests should be performed in controlled LBE. Comparison with post mortem tests in air at room temperature might be questionable.
- Relaxation-fatigue (creep-fatigue) tests: Relaxation-fatigue experiments on candidate materials should be carried out in the identified temperature range. Special attention should be paid to fatigue crack initiation, short crack behaviour and ageing effects in lead and LBE. The effect of damage induced in the materials by prior exposure to lead or LBE (like cavitation-induced by contact with liquid metal) on fatigue resistance should be determined.

- Fatigue tests: Low cycle fatigue tests should be carried out on candidate materials exposed to lead and LBE in the identified temperature range. Low cycle fatigue tests should be performed on treated and non-treated surfaces exposed to lead and LBE under oxygen control at high temperatures.
- Liquid metal embrittlement tests: Fracture, relaxation-fatigue and fatigue tests should be analysed as a function of mechanical solicitation and contact conditions (intimate at atomic scale between steel and liquid metal) to assess the sensitivity to liquid metal embrittlement of the materials.
- Environmentally assisted cracking tests: The effect of long-term ageing in lead or LBE on the mechanical behaviour of the candidate materials should be studied by means of tensile tests carried out after materials have been aged in flowing lead or LBE. The importance of surface corrosion layers (de-alloyed steel and/or surface oxide) on crack propagation should be studied. On the other hand, slow strain rate tests on materials exposed to stagnant LBE should be conducted and the obtained results compared with those given by irradiated specimens. Pre-stressed C-ring and SENT specimens from the candidate materials should be used to study crack growth under lead and LBE conditions. Corrosion behaviour of surfaces treated under stress, exposed either to flowing lead or LBE at high temperatures, should be investigated. The objective is to check the protectiveness of the oxide on the materials used for fuel cladding (pressurised tubes).
- Core and spallation simulation tests: Proton and neutron irradiation tests, aimed to generate data relative to the evolution of mechanical properties and microstructure of structural materials under irradiation conditions, simulating the core and spallation environment of EFIT, are needed. The approach should consist of testing materials (including welds, corrosion protection coatings) with different available experimental tools, like experimentation in fission reactors and with combined effects induced by irradiation in the presence of liquid lead and LBE under neutron spectrum. Experimental results from these tests should allow definition of the temperature window of selected materials for both the core and the target. Peak temperatures during abnormal or accident conditions and maximum allowable irradiation damage are relevant parameters.

4.5.6 Analysis by component

Reactor vessel

On its internal surface, the material is in contact with lead or lead-bismuth liquid metal, which may result in long-term oxidation/corrosion/erosion. These phenomena could be more severe on welding joints. Degradation of this component can be due to:

- reduction of crack resistance (especially for welded joints);
- oxidation/corrosion/erosion (the RPV has to guarantee a long lifetime, irreplaceable);
- fatigue: crack initiation and propagation.

Mechanical pump

The structural material erosion, in fluent lead, is considered acceptable if the relative velocity between the lead and the structural surface is kept below 2 m/s. This limit cannot be respected for the mechanical pump, where the relative velocity is up to 10 m/s. Material capable of operating with acceptable performance in fluent lead, with relative velocities up to 10 m/s and environment temperatures up to 500°C, should be individualised and qualified.

Steam generator

The SG shells and tubes (outer side) are in contact with a flowing (1 m/s) lead or lead-bismuth liquid metal. The SG inner tubes are in contact with water/steam with a temperature gradient that may result in oxidation/corrosion/erosion on the outer surface or in an increase of DBTT (note: no tube welds are immersed in lead). Degradation of this component can be due to:

- reduction of crack resistance (for example as a result of flow localisation);
- oxidation/corrosion/erosion affecting the material in terms of loss of thermal conductivity or loss of resistance;
- fatigue: crack initiation and propagation;
- liquid metal embrittlement.

Moreover, in the case of a SG tube rupture (SGTR) accident, it is necessary to assess immediately the affected component to be isolated and emptied. This action would allow unaffected SGs to maintain operation and evacuate decay heat. Suitable means for earlier SGTR accident detection have to be developed.

Fuel cladding tubes

On the inner surface of the cladding, fuel pellet clad interaction may occur and can modify locally the material microstructure. It is important to verify that no severe chemical interaction occurs between the fuel pellet and the clad material that can affect integrity of the cladding. On its external surface, the material is subjected to about 1 m/s flow of pure lead, which may result in oxidation/corrosion/erosion. Moreover, under the above conditions, high neutron flux irradiation worsens the material characteristics. The mechanisms by which cladding integrity or functionality can be lost are:

- irradiation-induced loss of strength;
- irradiation creep: dimensional/geometry instability (for example by swelling, stress, relaxation, etc.);
- fatigue: crack initiation and propagation;
- reduction of crack resistance (for example as a result of flow localisation);
- oxidation/corrosion/erosion affecting the material in terms of loss of thermal conductivity or loss of resistance and polluting the primary coolant;
- liquid metal embrittlement;
- local loss of resistance as a result of cladding-fuel pellet interaction.

Spallation target

The spallation target is in contact with flowing LBE or lead with a temperature gradient. In addition to the flowing LBE or lead, high neutron flux irradiation worsens material characteristics. Degradation of this component can be due to:

- irradiation-induced loss of mechanical strength and fracture toughness;
- reduction of crack resistance (for example as a result of flow localisation);
- oxidation/corrosion/erosion affecting the material in terms of loss of thermal conductivity or loss of resistance and polluting the target lead or LBE;
- fatigue: crack initiation and propagation;
- liquid metal embrittlement.

Purification system

Eroded structural material and oxide amount should be limited to an acceptable value in the lead or lead-bismuth molten pool to avoid formation of flow blockages. Means for lead purification during plant operation should be developed and qualified.

4.5.7 Final remarks

It is obvious that the various material properties are interdependent. It is desirable to provide experimental data not only corresponding to end-of-life conditions, but also for intermediate conditions, which are equally important for mitigation purposes. Finally, both base metal and welds should be considered.

4.5.8 References

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5. Other systems: MSR and SCWR

5.1 Generation IV nuclear systems

Six systems have been selected for the international Generation IV programme: GFRs, LFRs, MSRs, SFRs, SCWRs, and VTHRs. These systems are described in conceptual detail in reference [1]. Each Generation IV concept has its own operating conditions (see Table 12 for approximate operating conditions). Potential material candidates for Generation IV concepts were initially described at a workshop jointly sponsored by the Department of Energy, Office of Nuclear Energy, Science, and the Technology and Office of Basic Energy Sciences [2]. Tables 13 through 15 list potential structural materials, as well as expected performance issues for various material classes; each table describes various potential operating temperature ranges. Table 16 lists typical alloys within each alloy class. The majority of the international work has focused on the VHTR and SFR, but some countries continue to research the other four concepts, which are briefly described below.

5.1.1 Molten salt reactors (MSR)

The MSR is the only Generation IV reactor concept that utilises liquid fuel [10]. MSRs originated in the United States in the 1950s and 1960s with the implementation of two major experimental programmes; the Aircraft Reactor Experiment (ARE) and the Molten Salt Reactor Experiment (MSRE) [9-13]. A MSR utilises the concept of dissolving uranium and thorium fuel in circulating molten salt. Molten fluoride salts emerge as primary candidate salts because of their wide range of solubility for uranium and thorium. These salts also exhibit many other attractive properties, such as low melting point, high boiling point, low vapour pressure, high thermal, chemical and radiation stability and optical transparency and provide the ability to operate the reactor without pressurisation. In a MSR, the fuel salt flows through the reactor core where fission occurs within the flowing salt and is then circulated through an IHX and back into the reactor core. The use of homogeneous liquid fuel obviates the need for solid fuel fabrication and the liquid fuel itself can be easily purified from fission products, at least compared to solid fuels. The ARE used a UF_4-ZrF_4-NaF fuel moderated by BeO and the MSRE used Li^7F-based fuel with UF_4, ThF_4, and BeF_2. Unfortunately, MSR development was discontinued [9], but the ARE and MSRE established a firm technological foundation for future work on MSRs.

Recently, a joint effort by Oak Ridge National Laboratory, the University of California, Berkeley and Sandia National Laboratories has led to the design of an advanced high-temperature reactor (AHTR) that uses molten fluoride salt as a coolant and similarly to the VHTR, utilises a graphite moderator and coated fuel particles and operates at coolant exit temperatures of 700 to 950°C [11-15].

5.1.2 Supercritical-water-cooled reactor (SCWR)

The SCWR, essentially an advanced LWR, is designed for baseload electricity production with a high thermal efficiency (about 45% versus 33%) and potential for considerable plant simplification compared to the current fleet of commercial LWRs. The base design for the SCWR is a thermal spectrum reactor using conventional LWR-type oxide fuel and employing water rods as a moderator. The SCWR will operate above the critical point of water (374°C, 22.1 MPa); therefore the coolant does not undergo a phase change while passing through the core. The inlet temperature of the reference SCWR is ~280°C, while the outlet temperature is 620°C. The operating pressure of the reactor is 25 MPa. The combination of radiation, high temperatures, pressures and a rather aggressive chemical environment makes the SCWR one of the more challenging reactors from the standpoint of materials selection.

Table 12: Approximate operating environments for Generation IV systems

Reactor type	Coolant inlet temp (°C)	Coolant outlet temp (°C)	Maximum dose (dpa*)	Pressure (Mpa)	Coolant
SCWR	290	500	15-67	25	water
VHTR	600	1 000	1-10	7	helium
SFR	370	550	200	0.1	sodium
LFR	600	800	200	0.1	lead
GFR	450	850	200	7	helium/SC CO_2
MSR	700	1 000	200	0.1	molten salt
PWR	290	320	100	16	water

*dpa is a displacement per atom and refers to a unit radiation material scientists used to normalise radiation damage across different reactor types. For one dpa, on average each atom has been knocked out of its lattice site once.

Table 13: Candidate materials and performance issues for low-temperature (<~350°C) applications

Structural material	Performance issues
Ferritic pressure vessel steels	radiation embrittlement (toughness, DBTT)
Fe-base austenitic stainless steels	SCC, IASCC, high-dose embrittlement
Ni-base austenitic alloys and superalloys	IGSCC, IG corrosion, weld metal SCC, IASCC
Zirconium alloys	corrosion, hydriding
F/M alloys	radiation embrittlement (toughness, DBTT), IGSCC, IASCC, hydrogen embrittlement

Table 14: Candidate alloys and performance issues for intermediate temperature (~350-650°C) applications

Structural material	Performance issues
Ferritic pressure vessel steels	radiation embrittlement, (toughness, DBTT)
Fe-base austenitic stainless steels	creep strength, swelling and embrittlement, corrosion, IASCC
Ni-base austenitic alloys and superalloys	helium embrittlement, creep strength, swelling and embrittlement, corrosion, IGSCC, IASCC
F/M alloys	radiation embrittlement (toughness, DBTT), corrosion, IASCC, hydrogen cracking, corrosion in lead-based coolants and molten salts

Table 15: Candidate materials and performance issues for high-temperature applications (>650°C)

Structural material	Performance issues
Iron- and nickel-base superalloys	creep behaviour, toughness, helium embrittlement
F/M alloys	creep behaviour, toughness, radiation-induced embrittlement, corrosion in lead-based coolants and molten salts, dispersion stability in ODS alloys
Refractory metal alloys	creep behaviour, toughness, radiation-induced embrittlement, corrosion, oxidation, impurity pickup
Ceramic composites and coatings	creep behaviour, radiation and environmental effects on interfaces, toughness, corrosion in lead-based coolants or molten salts
Graphite	creep strength, swelling, toughness, thermal conductivity

Table 16: Common alloys within each alloy class

Structural material	Performance issues
Ferritic pressure vessel steels	A508 Grade 3, T22, 3Cr-3WV, 9Cr-1MoVNb, 7-9Cr2WV, 12Cr-1MoWV
Fe-base austenitic stainless steels	304, 316, 347, D9, NF709, Alloy 800H
F/M alloys	HT9, T91, NF616, HCM12A, 9-12Cr ODS, 9CR-2WVTa
Ni-base austenitic alloys and superalloys	617, 625, 690, Haynes 230, Hastelloy X, Inconel MA 754
Refractory metal alloys	TZM, T-111, Nb-1Zr
Ceramic composites and coatings	SiC, ZrC, TiC, TiN, ZrN, ZrC, SiC-SiC composite, C-C composite

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5.2 France – thorium nuclear fuel cycle and the MSRs

The Centre National de la Recherche Scientifique (CNRS) research programme has been driven by the assessment of a non-moderated thorium molten salt reactor (TMSR-NM) concept. This concept presents a number of advantages:

- The physics of the thorium-uranium cycle is more flexible than that of the uranium-plutonium cycle: it allows breeding in the thermal regime, but also in the fast regime. This opens the possibility to investigate non-moderated MSR reactors, which can accomplish both the breeding and incineration of all actinide isotopes.
- TMSR-NM can be started with plutonium and minor actinides from the waste of Generation III reactors. It will incinerate them while effecting a gradual transition to the thorium-uranium cycle, opening usage of the vast natural thorium resources. In all modes of operation, it allows breeding at a rate compatible with the growth of energy needs foreseen for many developed and developing economies.
- The modest on-line recycling fluxes required by the TMSR-NM nuclear fuel, as well as the chemistry needed for this operation, appear to be within reach of present technology.
- Fuel reprocessing is integrated into the very operation of MSRs.

These aspects have been described in a full report, which presents the status as of June 2008 of the research conducted on a) the thorium cycle and b) the molten salt reactors by the CNRS teams. It outlines the main lines of work identified by the academic community for the coming years, within the overall exploratory mission assigned to this research. The text below is an updated version from the relevant section of that report.

5.2.1 Structural materials

Considerable documented past experience on material choices exists due to two experimental molten salt reactors built in the US and elsewhere [1] and references quoted therein. The essential problem identified for alloys used for structural components was that of corrosion by molten fluorides. Accordingly, the first reactor built in the 1950s (the ARE) used one of the best corrosion resistant alloys then available, i.e. a Ni-Cr alloy (an Inconel alloy of composition Ni-15%Cr-7%Fe, in wt %). It was found that although this alloy has excellent oxidation resistance in air, chromium was oxidised by the fluoride, producing

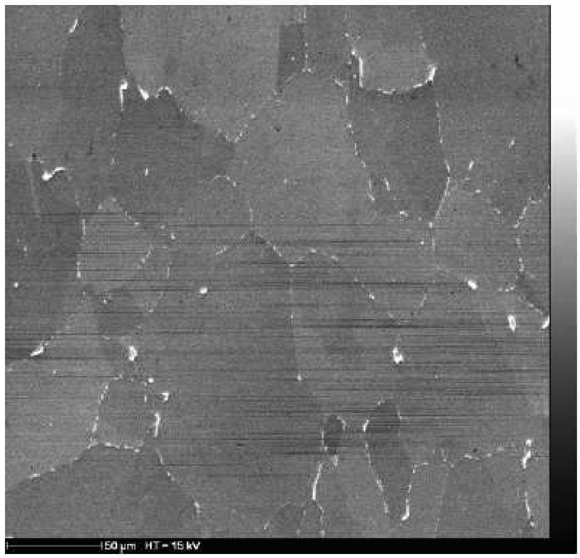
CrF_2 and leading to severe corrosion. Hence the next experimental reactor in the 1960s, the MSRE, used a chromium-free Ni-base alloy (Hastelloy B with composition Ni-29%Mo-5%Fe in wt%). This showed excellent corrosion resistance to molten fluoride salts, but unfortunately suffered from oxidation spalling on the air side. Further alloy development led to Hastelloy N (Ni-16%Mo-7%Cr-5%Fe-0.05%C, in wt%). Here, just sufficient chromium was added to achieve a compromise performance between molten fluoride salt corrosion and oxidation in air. The added chromium led to improved creep resistance through grain growth control. Subsequently, minor additions of niobium and aluminum to improve strength and oxidation resistance led to a new alloy that was tested in molten fluorides with a measured corrosion rate of 50 µm after about 11 years at 700°C. At that time, these alloys could be produced as tubes, plates, bars, forgings and castings and readily welded and brazed. It should also be noted that over 40 years, substantial improvements in industrial production have been made and alloys with substantially reduced levels of unwanted (impurities) or unnecessary (introduced via ferroalloys) elements can be achieved. It can be concluded that very reasonable solutions exist for structural materials for a molten fluoride salt reactor operating below 750°C.

Nevertheless, even if past experience has provided a wealth of information and well-developed alloy solutions, a number of problems still require more work. As mentioned above, the alloy composition has to be a compromise between low Cr content for tolerable fluoride corrosion, and yet enough to obtain acceptably low oxidation rates in air. To obtain acceptable fluoride corrosion it is, in general, sufficient to control the redox potential of the salt. This is achieved by means of the UF_4/UF_3 system used as a redox buffer. The natural trend of the molten salt potential is to evolve towards high and thus oxidising values. It then becomes necessary to reduce a fraction of UF_4 into UF_3. This is done by an injection of small amounts of metallic beryllium, which induces the reaction: $Be + 2\ UF_4 \rightarrow 2\ UF_3 + BeF_2$. To stay within acceptable redox conditions, the typical value for the ratio UF_4/UF_3 should be 100. Under these conditions, overall corrosion, as measured in forced convection loops (~2 m/s) is approximately 3 µm/year for the core molten salt breeder reactor (MSBR) salt (with UF_4), but can reach 50 µm/year for the molten salt considered for the secondary loop ($NaBF_4$-NaF). Finally, some fission products, such as tellurium, may lead to the grain boundary embrittlement observed in Hastelloy N. This would occur via grain boundary diffusion, even though the diffusion coefficients are small. Precise mechanisms for the embrittlement (brittle intermetallic grain boundary phase, grain boundary segregation) remain unknown. Recently, Russian teams have suggested that an addition of 0.1% manganese in modified Hastelloy N would reduce this effect.

Irradiation resistance of Ni-base alloys is also a problem that needs to be addressed; helium production could be a limiting factor. In a fast neutron flux, transmutation of nickel leads to the production of helium atoms, which, when they diffuse and are allowed to combine, can produce severe intergranular embrittlement. For austenitic steels (i.e. Ni-containing stainless steels) this effect can be mitigated by a fine dispersion of precipitated carbide particles: the helium atoms are then trapped at the matrix/carbide interfaces. Similar effects are expected in Ni-base alloys, provided an adequate microstructure is obtained. Little is known about irradiation-induced swelling, creep and solute segregation. These effects should be investigated in candidate materials and eventually reduced to acceptable values, given the irradiation conditions by a fine tuning of the alloy composition and microstructure.

The CNRS teams have so far focused their research on less well-known nickel-tungsten-chromium alloys, in which molybdenum is substituted by tungsten. They have a number of potential advantages. Indeed, the substitution should lead to an increase in creep resistance by at least an order of magnitude. Moreover, long-term activation problems should be reduced. These alloys are strengthened by solid solution effects and by the presence of strong short-range atomic order. It is expected that the corrosion resistance in molten salts will be similar to alloys containing molybdenum, while the presence of tungsten improves the oxidation resistance in air. It might thus be possible to further reduce the chromium content. Another interesting feature is that large tungsten content leads to grain boundary precipitation of tungsten (see Figure 14), which can also improve creep resistance, allowing a higher operating temperature of the reactor.

Figure 14: Re-crystallised microstructure of a Ni-W-Cr alloy tailored for usage in contact with fluoride salts (SEM-BSE)



Note: The light contrast at the grain boundaries correspond to tungsten precipitates.

Following initial laboratory studies, several compositions have been produced on a semi-industrial scale by the French firm "Aubert et Duval". Work in progress concerns both molten fluoride corrosion and oxidation resistance in air. Since corrosion is strongly chemistry specific, further work is necessary to study corrosion of these alloys in the secondary loop ($NaBF_4$-NaF).

Finally, at the present early stage of selection of reactor specifications, other materials can be envisaged. For instance, instead of nickel-tungsten-chromium alloys, ZrC could be used for axial reflectors. The fabrication of large reactor ZrC components is challenging, in terms of synthesis, manufacturing, mechanical properties and behaviour under irradiation. Substantial research effort is required here to address the specific needs of molten salt reactors. Graphite might also be envisaged, since low porosity graphites have been previously developed for the MSRE. Unfortunately much of the know-how has been lost. Nevertheless, graphite development (with lower density) has continued for the high-temperature reactor, leading to substantial improvement of life expectancy in irradiation conditions. The key factor has been improvement of isotropy and homogeneity. Still, the difficulties involved in producing the appropriate graphite grade should not be underestimated and CNRS continues to support research in this area.

As of today, it does not seem that existing materials problems are so intractable as to exclude the realistic design of a TMSR-NM. For the metallic structural components, the proven Hastelloy N is already a good candidate for temperatures up to 750°C. Presently ongoing research holds the promise for substantially improved alloys (in terms of corrosion and oxidation resistance, high-temperature creep, reduced long-term activation). Given recent advances, graphite does not appear to raise such major concerns as was the case at the time of the MSBR. Nevertheless, more work is required on the effects of irradiation on Ni-base alloys. In addition, an investigation of material compatibility with various batch processing fluids (liquid bismuth-thorium alloys, chloride melts) is also needed.

The CNRS activities on the thorium cycle and molten salt reactors have been financed by the "Programme Concerté de Recherches sur les Réacteurs à Sels Fondus" (PCR-RSF) of PACEN. An additional contribution has come from the PACEN research group GEDEPEON, which is jointly operated by CNRS, CEA, EDF and AREVA NP. The FP6 Euratom programme, ALISIA, has provided support for international contacts and collaborations.

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5.3 Supercritical-water-cooled reactor

The following information has been gained from the HPLWR-phase 2 European Project [1].

The general layout of the high-performance light-water reactor (HPLWR) is sketched in Figure 15. Like in a PWR, the core is arranged in the lower half of the RPV. Control rods are inserted from the top of the reactor. Feed water entering the RPV is supplied through the down-comer to the lower plenum and through the upper plenum of the RPV, from where it can be used as moderator water flowing downward into the core. This counter current flow of moderator water and rising coolant requires a closed steam plenum on top of the core, which collects the steam to be supplied to the steam turbines and on the other hand, allows a moderator water supply to the top of the core. Moderator water and feed water through the down-comer need to be mixed homogeneously at the core bottom, which requires a mixing plenum underneath the core.

Figure 16 presents a segment of the RPV with outlets, the core barrel, steel reflector, control rod guide assembly (CRGA) and the positioned steam plenum. For simplification, only a single fuel assembly cluster is displayed here. The core barrel is suspended at the lower vessel top and centered in radial direction using four centering elements. The alignment in the vertical direction is realised using protruding supports at the bottom of the vessel. The core barrel sits, together with the CRGA, on a ledge machined from the RPV flange and is preloaded with a spring element. The lower vessel is braced with the closure head flange using reduced shank bolts and nuts. Two O-ring seals ensure leak tightness between the closure head and the lower RPV.

Figure 15: Scheme of the HPLWR [2]

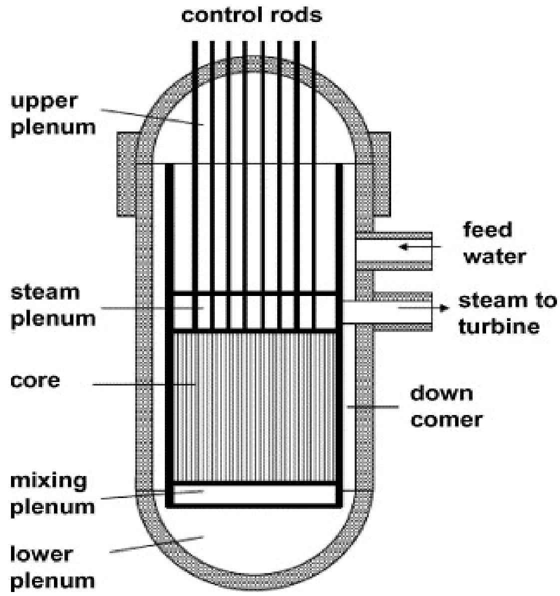


Figure 16: Design of the assembled RPV with internals [3]

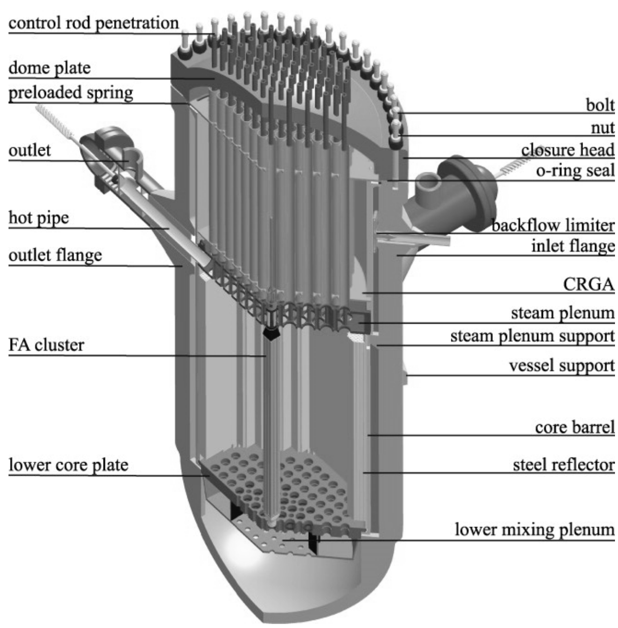


Table 17 reports some design characteristics of the RPV and internals of the HPLWR [3]. These operational conditions are based on the design data of Oka for in-core components and the design recommendations of Bittermann for the RPV [4]. The HPLWR has a modified core assembly in order to use fuel with lower enrichment, has its hot steam completely separated from the RPV and includes several safety features added to the original Oka design [5].

Table 17: Characteristics of the HPLWR RPV and internals [3]

Parameters RPV	
Operating/design pressure	25.0/28.75 Mpa
Operating/design temperature	280/350°C
Number of cold/hot nozzles	4/4
Dimensions RPV (m)	
Height (including closure head)	14.26
Height (excluding closure head)	11.42
Inner diameter	4.47
Wall thickness (cylindrical shell)	0.45
Wall thickness (bottom head)	0.20
Wall thickness (flange)	0.56
Wall thickness (closure head)	0.40
Material RPV/internals	
Vessel, closure head	20 MnMoNi 55 (SA 508)
Outlet pipe, hot pipe	X10CrMoVNb9-1
Steam plenum	Type 316L (N) stainless steel
Weight RPV/internals (t)	
Lower vessel	508
Closure head with nuts and bolts	164
Internals	594
Steam plenum	8.6

5.3.1 *Reactor pressure vessel*

Since the operational temperature of the RPV is similar to the conventional LWR, it has been decided to use similar structural materials for this component. The reference material is 20 MnMoNi 55 (SA 508 Cl 3), which has a tensile strength of 590 MPa at 350°C.

A huge database exists on the effect of radiation on low alloys steels, as 20MnMoNi55 and similar alloys, coming from the RPV surveillance programmes of existing operating LWRs. Radiation embrittlement mechanisms understanding is one of the main subjects of ongoing research programmes related to the LWR RPV, to assess long-term operation up to 60 (and possibly even 80) years. The outcomes of these projects would help the selection of new materials and the application of new RPV integrity assessment methods.

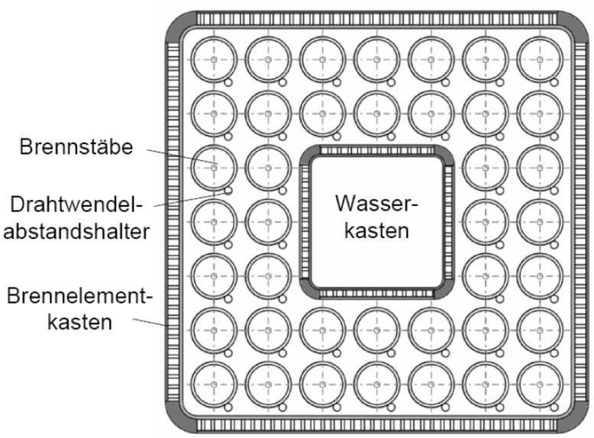
The size of the SCWR RPV is around double that of existing operating LWR RPVs, assuming a classical RPV material 450 mm thickness compared to the ~200 mm of a boiling water reactor [(BWR) RPV] in order to deal with the 25 MPa operating pressure. Maintaining mechanical and microstructural properties through thickness for such a thick pressure component is now a manufacturing challenge. Other important issues are welding and inspection.

The design life of 60 years could impose a potential thermal ageing effect. Recent studies have shown that thermal ageing RPV steels could be a concern when the temperature rises from 300 to 350°C for 40 years of design life, due to phosphorous segregation at grain boundaries [6].

5.3.2 In-core components

The challenges of structural materials of the HPLWR are mainly related to the cladding materials and the fuel assembly box. Since the highest peak temperature of the clad is more than 630°C, the typical clad materials used in LWRs (zirconium alloys) cannot be used. As far as the fuel assembly box is concerned, this should be thermally isolated to minimise heating of the moderator (water). Thermal isolation of the boxes, on the other hand, implies that those boxes can reach temperatures up to 600°C and the moderator (water) has an excess pressure of 500 kPa. For these temperatures and pressure conditions, zirconium alloys cannot be used and the boxes require a higher stiffness when compared to those of a BWR. Some design solutions to overcome the issues of the fuel assembly box have been presented [1]. Figure 17 shows a schematic of the fuel assembly box.

Figure 17: Fuel assembly box



Brennstäbe = fuel rods; Drahtwendelabstandshalter = wired fuel rod spacer;
Brennkasten = fuel assembly box; Wasserkasten = water (moderator) box.

Selection of cladding materials is difficult, as cladding is exposed to neutron irradiation (up to 60 dpa), high temperature (up to 630°C), under high stress (> 100 MPa) and wetted by supercritical water [4]. The main issues to be addressed for core internals are oxidation, SCC and irradiation-induced degradation [7].

Initially three groups of materials were considered for the in-core materials: F/M steels including ODS versions, high-temperature strength austenitic stainless steels and Ni-base alloys. Within Work Package 4 of the HPLWR Phase 2 project, the tests materials foreseen are [8]:

- ferritic steels 9-12 Cr: Eurofer97, Eurofer 97 ODS and HCM12;
- austenitic Cr-Ni steels: PM2000 ODS 19%Cr, Type 316NG (nuclear grade) stainless steel, TP347H, Sanicro 28, 15Cr15NiTi (1.4970);
- high alloys and Ni-base alloys: Alloy 800H, Inconel 690, Inconel 625.

Water chemistry in SCWRs is still an open point. A combination of the boiler cycle of conventional supercritical power plants and LWRs could be adapted to the HPLWR [4]. The use of hydrogen water chemistry to suppress radiolysis is not clear, the hydrogen concentration needed seems to be very high and the effect of water density is not known [9]. Weight gain tests of HPLWR candidate in-core materials were made under BWR normal water chemistry (NWC 250 bar, 100 to 150 ppb oxygen) and hydrogen water chemistry (HWC; 250 bar 33 cc/kg hydrogen). The weight gain of the 15Cr-15Ni alloy (1.4970) and ODS steels are comparable at 650°C, being lower that the weight gain of 361NG and F/M steels. At the same temperature, Type 316NG stainless steel is not better a performer that conventional F/M steels. Another conclusion of this work is that HWC did not reduce oxidation rates at 650°C. However, the results must be further analysed [10].

An experimental SCWR loop operating in a research reactor has been designed at UVJ Rez, Czech Republic. This loop will be used for testing and optimisation of water chemistry and studies of radiolysis of water at supercritical conditions within the HPLWR Phase 2 project [11].

Beside the need for a definition of water chemistry and radiolysis for the HPLWR, the following issues should be taking into account [7]:

- the evolution of oxide layers and stability with time;
- SCC data is needed for sensitised and cold-worked materials under SCW conditions;
- the influence of cyclic and constant loads should be also studied.

Regarding irradiation damage, one of the concerns on the use of austenitic stainless steels is the reduction of creep rupture time due to the formation and unstable growth of helium bubbles at grain boundaries [12]. A main limitation of F/M steels is irradiation hardening at or less than 400°C [13]. Phase stability under irradiation and irradiation-induced segregation needs to be investigated for F/M steels. Recent results of F82H (a 9Cr F/M steel) irradiated at 500°C show the evidence of neutron embrittlement without hardening. This non-hardening embrittlement could be caused by precipitation of laves phases during irradiation [14]. The creep strength of F/M steel at temperatures above 600°C is limited.

Commercial Ni-base alloys under irradiation at 400 to 600°C at a neutron dose of 10 to 15 dpa suffered a significant reduction in tensile ductility. For higher neutron doses, substantial levels of swelling occur [15].

The number of materials candidates are quite huge, but from available data on creep strength, corrosion properties and experience in commercial nuclear power plants, it was concluded that Type 316NG stainless steel is the most promising steam plenum material for the HPLWR project, see Table 1. Higher chromium–nickel containing alloys, which promises improved corrosion properties, would need improvement of strength properties [16] for other core internals.

Optimisation of the chemical composition of candidate austenitic stainless steels is needed to achieve an optimum in creep-rupture strength and corrosion resistance [4].

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6. Conclusion

6.1 Main issues of the materials for advanced nuclear systems

The Generation IV initiative defined six advanced reactor concepts that had the ability to greatly improve nuclear reactor performance in terms of safety, proliferation resistance, economic performance and minimisation of waste. For all six concepts, improvements in material performance will be critical to the ultimate success of the reactor concept. Tables 18, 19 and 20 provide lists the key materials being studied in each country, broken down by reactor concept. Careful analysis of the tables indicates that many materials, or material classes, are common across concepts allowing for many opportunities for cross-cutting research programmes that would ultimately benefit multiple concepts.

Both for austenitic steels and F/M steels, a number of improved alloy compositions have been developed and continue to be developed. Nevertheless, the physical metallurgy reasoning which guides alloy development is far from obvious, and would merit some attention at a fundamental level. Initiatives to improve nickel based alloys, where even modest improvements in high-temperature strength would be clearly beneficial, have been limited. It is very unlikely that there is a single universal improved alloy suitable for all types of applications within any of these alloy classes,. Specific alloy development may well be required for the optimised design of specific components. Furthermore, although world resources for most alloying elements exist, the quality of the minerals used may well be different, leading to different levels of different impurities. Extra care should therefore be taken in the definition and analysis of alloy compositions.

A number of reactor projects rely to a considerable extent on the promise of better performance (stress, creep, temperature, even corrosion resistance) of ODS steels. However, substantial research at all levels will be required before any of these can be used in reactor components, from applied (fabrication, shaping or welding) to the more fundamental (physical metallurgy, microstructural stability or irradiation effects). The amount of effort required, from a fundamental understanding to the mastering of robust manufacturing processes, including forming and assembling, should not be underestimated. Concerted international research with clear go, no-go steps would be helpful here.

The C/C and SiC/SiC composites have been studied extensively and are used for other (aerospace) applications, but it is unclear whether they can be readily transposed to nuclear reactors without substantial further research, notably for fuel rod cladding. Much less work has been undertaken on the Ti_3SiC_2 ceramics and the knowledge base remains somewhat limited.

The properties of nuclear graphite have been studied for many years as they were extensively used in the UK AGR programme. Unfortunately, the graphite grades used in the past no longer exist and so studies are ongoing to qualify modern graphite grades, as the properties change with input materials and processing conditions.

6.2 Summary of material requirements for various advanced nuclear systems

Table 18: Material requirements for VHTR

System	Candidate materials	Issues to be solved
VHTR	Republic of Korea • F/M (Mod.9Cr1Mo) or low alloy steel for the reactor vessel; • high-temperature nickel alloys (Hastelloy X, IN 617 or Haynes 230) for gas ducts and intermediate heat exchanger; • C/C, SiC/SiC composites or austenitic stainless steels (16Cr-16Ni-2Mo-1Nb or 800H) for control rods; • Nuclear grade graphite for the reflector.	• creep fatigue, embrittlement, thermal expansion, thermal fatigue, crack resistance, oxidation resistance, carburisation resistance and thermal conductivity; • high-temperature creep (950°C) for nickel alloys; • corrosion resistance in impure He for nickel alloys; • flexural strength for composites; • oxidation resistance in impure He for composites.
	United States • low alloy steel (SA-508) or Grade 91 steel for the pressure vessel; • high alloy stainless steels (alloy 800H), graphite and SiC/SiC or C/C composites for reactor internals; • nickel alloys (Haynes 230, Alloy 800H, Inconel 617, Hastelloy X) or ceramics for heat transfer system and heat exchangers.	
	Japan • graphite or SiC/SiC composites for the core and reflector; • 2.25Cr-Mo steel for pressure vessel and high-temperature coaxial tubes, with possible use of 9Cr F/M steel for extended lifetimes; • Hastelloy XR for high-temperature coaxial tube liner and IHX heat-pipe; • Alloy 800H for control rod sleeve in the HTTR and also in VHTR (either, but below 900°C for 5 years). C/C and SiC/SiC composites are also considered.	
	Switzerland • Materials of interest are conventional RPV steels, grade 91, IN617, IN800H and others; • ODS steels, TiAl or fibre reinforced ceramics for high-temperature options.	
	France • 9%Cr F/M steels (and improved versions) for the pressure vessel; • C/C or SiC/SiC composites for control rods; • nickel-based alloys (Inconel 617, Haynes 230) or ODS alloys for intermediate heat exchangers; • nickel-based alloys (Haynes 230, Inconel 617) for the primary cooling circuit; • graphite for the core ; • SiC, SiC/SiC composites (as well as other carbides and nitrides) for cladding materials.	

Table 19: Material requirements for SFR

System	Candidate materials	Issues to be solved
SFR	United States • stainless steel 316 for the reactor vessel; • advanced stainless steels (Ti-stabilised 316, D9 (15Cr-15Ni-Ti), HT-UPS, and NF709) for internal structures; • F/M (Mod.9Cr1Mo) steel for intermediate heat exchanger, piping and steam generator; • F/M (Mod.9Cr1Mo) or improved F/M steel for cladding material; • ODS steels (MA-957, 12YWT, and 14YWT) for cladding.	• design requirements for cladding material (Tmax 650°C, 250 dpa, swelling <5%); • swelling at T< 400°C for cladding; • assembly and welding of ODS steels; • assembly and welding of advanced F/M steels; • complete database required for HT-9, advanced F/M steels and ODS.
	France • ODS steels for cladding with 15-15Ti or advanced austenitic steels as back-up; • 316LN austenitic steel for below core structures and primary circuit (cold structures – 400 °C, low irradiation); • improved 316 type steels or nickel-based alloys (Inconel 617) for above core structures (hot structures – 550 °C, low irradiation); • F/M steels (e.g. T91) for secondary circuit and GV are strongly considered due to the potential gain of a much more compact circuit.	
	Japan • ODS and precipitation hardenened (PH) ferritic steels for core components (cladding and duct tubes respectively); • 316FR and Modified 9Cr-1Mo steel for structures; - Modified 9Cr-1Mo steel for primary and secondary coolant systems, intermediate heat exchangers and steam generators.	
	Republic of Korea • Type 316 stainless steel for the reactor vessel; • Mod. 9Cr-1Mo for IHX, IHTS, piping and SG; • HT9 for cladding.	

Table 20: Material requirements for other systems

System	Candidate materials	Issues to be solved
ADS	• essentially F/M (T91 or F82H) or austenitic (316L or JPCA) steels for most components; • additionally MAXTHAL (Ti3SiC2) for LM circulation pumps, possible use of protective surface coatings (aluminisation); • longer term : ODS for higher temperature applications.	• irradiation embrittlement; • lm embrittlement; • combined irradiation and lm embrittlement; • high-temperature corrosion (flowing lead); • stress corrosion cracking; • creep resistance.
LFR	ELSY • improved austenitic steels, such as AIM1 (15-15 Ti mod Si); • improved F/M steels (T91, EM-10, HT9, F82H, etc.); • ODS steels (9Cr ODS or 14Cr ODS).	• LME and LM corrosion; • use of coatings.
SCWR	Republic of Korea (under evaluation) • F/M steels (T91-I, T91-II, T92, T122) ; • high Ni alloys (Alloy 625, 690 and 800H); • ODS alloys (MA 956). Europe • low alloy steels for the pressure vessel; • austenitic stainless steels, F/M steels, ODS steels, nickel alloys are all considered as candidate materials for cladding; • 316NG (LN) stainless steel for the steam plenum.	• corrosion; • stress corrosion cracking; • cladding materials; • high-temperature strength.
TMSR*	Nickel-based high-temperature alloys (Ni-W-Cr) for most components in contact with molten salts.	• high-temperature strength and creep resistance; • corrosion and embrittlement in molten salts.

*Thorium molten salt reactor (TMSR).

Appendix I: Research activities on materials for Generation IV nuclear systems in the United Kingdom

Introduction

This short report has been prepared for the OECD/NEA Expert Group on Innovative Structure Materials, for the purpose of summarising the current United Kingdom (UK) position on nuclear materials research for Gen-IV systems. The information has been collected from publicly available documents and discussions with researchers in the UK. It is as complete as possible, but may be missing some elements, as there is currently no UK national programme on nuclear materials or nuclear fission research that co-ordinates or records this activity.

Background: UK participation in GIF

The UK is a charter member of the Generation IV International Forum (GIF). The 2003 Energy White Paper aimed to support the UK skills needed to "Keep the Nuclear Option Open" over the longer term and capability to keep abreast of international developments and inform UK policy development. In the 2005 Framework Agreement, the UK agreed to undertake research and development in the six selected Generation IV technologies, along with the United States, Canada, France and Japan. However, in 2006 the UK withdrew from active status in GIF and there is currently no UK government funding for UK researchers to participate in GIF activities. UK researchers can participate in European projects through Euratom, but there are no current national programmes on Gen-IV to contribute funding or co-ordinate the interactions. Hence there is an ad-hoc participation in European Union programmes and international collaborations.

UK participation in the EERA nuclear materials joint programme

The objective of the EERA Nuclear Materials Joint Programme is to identify key priority topics and funding opportunities with the purpose of supporting, in an efficient way, the development and optimisation of a sustainable nuclear energy. The JP agreement includes joint planning and implementation of research activities, sharing of research infrastructures and facilities and optimisation of efforts and resources. The challenge identified is the availability of appropriate structural and clad materials that are able to withstand the severe conditions of high temperature and thermal gradients, high irradiation doses, long lifetime and corrosive environments.

The Joint Programme has four main research themes, which are the following:

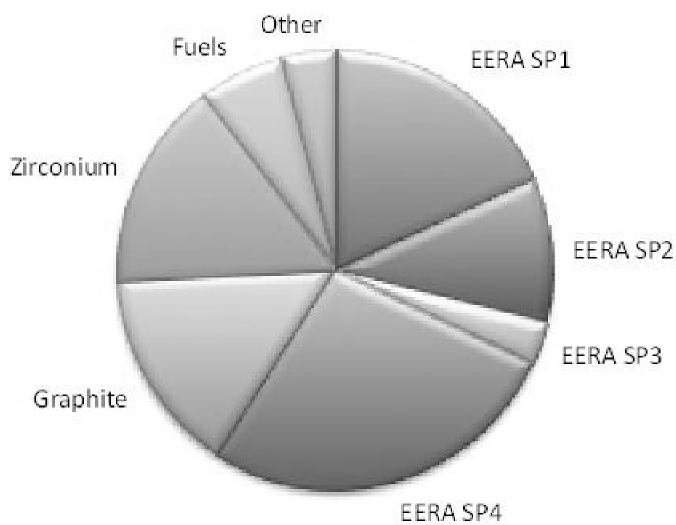
- material development and screening;
- experimental characterisation and qualification/validation;
- fabrication issues;
- pre-normative research.

These have been structured into four sub-programmes:

- SP1: Support to the European Sustainable Nuclear Industrial Initiative;
- SP2: Oxide Dispersed Strengthened (ODS) Steels;
- SP3: Refractory Materials: Ceramic Composites and Metal-based alloys;
- SP4: Modelling: Correlation, Simulation and Experimental Validation.

The author conducted a survey of nuclear materials research in UK universities in mid-2010. The intention was to identify research that was potentially relevant to Generation IV materials, although no proscription was placed on this. The research groups that were contacted were identified primarily from public data on EPSRC (Engineering and Physical Research Council) funded grants and from research papers published in the literature. Those contacted were also requested to nominate other research groups in the UK who would be relevant. The focus of the survey was on university research, although the National Nuclear Laboratory and the National Physical Laboratory also contributed to the survey. In total, data were obtained on active projects from research groups at 15 Universities, the National Physical Laboratory (NPL) and the Culham Centre for Fusion Energy (CCFE). Approximately 70 projects were identified, on which 126 researchers had been or were employed. Projects at Universities made up 118 of these researchers, of whom 99 were working on projects that were currently active or about to start. The average proportion of researchers is approximately 40% postdocs to 60% PhDs. The descriptions of the EERA Joint Programme were used to allocate the UK projects to one of the four sub-programmes. This is a provisional description to show the approximate balance of UK activity and is shown in Figure A.1.

Figure A.1: UK Generation IV nuclear materials activity (by number of researchers, including completed projects), grouped by theme



UK research council support

The UK Research Councils support publicly funded research at UK universities, of which the Engineering and Physical Sciences Research Council (EPSRC) is the supporter of nuclear materials research. It has stated a need for an "important foundation in broadening out the nuclear academic research base by funding specific programmes".

At present, there is no funding directed at Generation IV international programmes. However, the UK Research Councils (i.e. EPSRC) have recently expressed a wish to encourage UK researchers to participate in the European Energy Research Alliance (EERA) collaborative programmes, where this will add value to current UK energy research efforts and support the UK low-carbon strategy. The EERA joint programmes include innovative nuclear fission technologies. The research councils do not wish desirable participation to be impeded by lack of funding.

Potential participants will be expected firstly to seek to contribute all or part of existing Research Council funded projects into the collaborative programmes. The Councils anticipate that this might require some modifications of existing project plans and are ready to consider this on a case-by-case basis. In doing so they will expect the objectives of each existing project to remain broadly unchanged. Where participation requires additional funding, a proposal will be required, which will be expected to justify the added value to the UK and demonstrate that the proposer's existing support cannot reasonably be used for the purpose. Such proposals will be competing against other research proposals in the EPSRC remit.

UK researchers who wish to engage with international Generation IV programmes, such as EERA, in this way are encouraged to discuss potential participation with the Research Council, which will offer advice on the likely fit of proposed activities to existing projects, to overall UK activity and objectives and identify appropriate contacts in the Research Councils. At the present time the EPSRC does not offer funding from its own resources, nor does it on behalf of the other research councils.

In recent years EPSRC support for nuclear fission research has included programmes on "Keeping the Nuclear Option Open (KNOO)", "Sustainable Nuclear Energy", "Waste management (DIAMOND)" and the Nuclear Engineering Doctorate Programme. Some details on current and recent programmes are given in the following sections.

EPSRC KNOO research consortium

This significant consortium ran for four years with a budget of £6M, from October 2005 to March 2010. The research focus was maintaining current generating capacity, fission within a sustainable energy economy and future fission power. It funded research at the Universities of Manchester, Imperial College (London), Newcastle, Leeds, Bristol, Open University and Cardiff. It was a wide-ranging activity, with four work packages. Its aim was training of Ph.D. researchers and invigoration of the university research community in nuclear fission. The KNOO project leaders have represented the UK by attending Euratom co-ordination meetings on GIF issues and the consortium's research has had links to Euratom FP7 programmes RAPHAEL and GCFR STREP. The breadth of the research is shown by the research projects in each work package.

- KNOO WP1: fuel, thermal-hydraulics and reactor systems:
  - experimental and computational modelling of heat transfer through crud;
  - multi-pin coupled re-flood model;
  - study of rewetting processes during bottom re-flooding of PWR core after loss of coolant accident;

- computational modelling of re-flood heat transfer and fluid dynamic processes inside a pressurised water reactor during a loss of coolant accident;
- interactions of water droplets with hot solid surfaces: modelling and experimentation;
- thermal-hydraulic studies of re-flooding inside a fuel bundle in a PWR during a LOCA;
- innovative approaches to nuclear industry inspection and monitoring.

- KNOO WP2: materials performance issues within nuclear applications:
  - residual stress measurement in non-metallic materials;
  - mechanical performance of nuclear cladding and structural material;
  - nuclear graphite;
  - microstructural characterisation of nuclear graphite;
  - simulation of irradiated graphite;
  - micromechanical behaviour Of zirconium alloys;
  - mechanical performance of nuclear structural materials;
  - modelling SCC in corrosion-resistant materials;
  - mechanistic understanding and predictive modelling of chloride-induced atmospheric stress corrosion cracking (AISCC) in austenitic stainless steels;
  - mechanistic understanding of irradiation effects on stress corrosion cracking of stainless steels;
  - wireless data acquisition and power management.
- KNOO WP3: nuclear waste management:
  - structure and defect stability of calcium fluorapatite radionuclide colloidal ternary systems;
  - performance and evolution of Pu-contaminated CMS cementitious wasteforms;
  - investigation of particle-laden flow using large eddy simulation;
  - the influence of mixed salts on the zeta potential and shear yield stress of titanium dioxide (TiO_2);
  - transport and rheology of nuclear sludge;
  - ZrC_{1-x} for VHTR coated particle fuel: non-stoichiometry and effects on thermal and mechanical properties;
  - physical modelling of solid-liquid systems in pipe-flow;
  - cements for immobilising sludges.
- KNOO WP4: safety and performance for innovative reactor systems:
  - coupled neutronic fluid dynamic accident transient analysis of the Gen-IV VHTR using the finite element transient criticality (FETCH) code;
  - development of the FETCH 3D unstructured mesh coupled transient fault/severe accident model for innovative reactors and support for KNOO Ph.D. students;
  - nuclear CFD for fault studies – uncertainty modelling and benchmarking of LES calculations;

– atomistic modelling of fission gas release in uranium dioxide;

– investigation and demonstration of the FETCH technology applied to asymmetric within vessel faults/severe accidents for GFR;

– turbulence modelling for post-trip reactor core flows;

– microstructural characterisation of nuclear graphite.

EPSRC nuclear fission consortia

In October 2009, bids were invited from the EPSRC, from consortia against a series of identified research areas related to nuclear fission power generation. The themes and the expected topics were identified following a UK workshop, attended by UK stakeholders in universities and industry. Following the outline proposal stage, ten consortia were invited to submit full proposals.

The themes and possible areas of focus identified by the UK workshop were as follows:

- novel separations science and processes for spent nuclear fuel (covers existing and future fuels); the aim here is to support the U/Pu fuel cycle:
  - chemistry and radiation science of separated nuclear fuel;
  - novel separations to partition actinides (uranium, plutonium, neptunium, curium, etc.) and long-lived radionuclides;
  - process engineering covering separation of high burn-up/novel fuel;
  - process/chemical models for novel separations of spent fuel;
  - nuclear systems analysis to understand the impact of novel separation technologies on the wider sustainability of nuclear energy;
  - nuclear data covering fission product yields and decay data from spent fuel.
- long-term materials behaviour:
  - development of models capable of predicting materials behaviour from processing and fabrication stages through to realistic operating conditions and long-term operation (60 to 100 years);
  - address simulation of nuclear related materials issues (metals and ceramics), such as radiation damage, interfacial effects such as IASCC, and long-term ageing/creep;
  - development of advanced materials examination techniques capable of assessing material damage/performance under long-term operational conditions and techniques should look at the state of the material, rather than traditional NDT techniques focused on looking for defects;
  - methods for prediction of long-term component response, with and without defects, based on advanced materials models;
  - impact of CRUD on material behaviour.
- fuels:
  - underpinning research aimed at elucidating the fundamental processes related to nuclear fuel performance, including aspects of fuel manufacture, characterisation and in core behaviour;
  - radiation damage effects;

  - fission product behaviour in chemical thermal gradients;
  - cladding behaviour and microstructure evolution;
  - ultimately the aim is to support high fuel burn-up.
- reactor systems:
  - thermal-hydraulics and heat transfer in reactor systems;
  - modelling the whole core and complex multiphase flows to simplify design and improve safety through passive systems and natural circulation;
  - improved understanding of system response to severe accidents and faults;
  - materials and chemistry underpinning the behaviour of reactor systems;
  - nuclear data, including neutron reaction cross-sections and reactor kinetics;
  - formation of CRUD, its role in activity build-up and mitigation.

Of the 10 full proposals, 6 were funded to a total value of approximately £6.3M. The successful consortia launched in 2010, were as follows. Further details (i.e. grant holders, partners and abstract) on each of these projects are available via www.epsrc.ac.uk, by consulting "Grants on the web" with reference to the project number (e.g. EP/I003320/1).

- The Development of Advanced Technologies and Modelling Capabilities to Improve the Safety and Performance of Nuclear Fuel (EP/I003320/1) (£1 164 276).
- Performance and Reliability of Metallic Materials for Nuclear Fission Power Generation (EP/I003282/1) (£343 629).
- MBase: The Molecular Basis of Advanced Nuclear Fuel Separations (EP/I002855/1, EP/I002928/1, EP/I002952/1, EP/I003002/1) (£1 303 701).
- Nuclear Data: Fission Yields, Decay Heat and Neutron Reaction Cross-Sections (EP/I003126/1, EP/I00324X/1, EP/I003258/1) (£564 949).
- Computational Modelling for Advanced Nuclear Power Plants (EP/I003010/1) (£1 569 231).
- Fundamentals of current and future uses of nuclear graphite (EP/I002588/1, EP/I002707/1, EP/I003169/1, EP/I003223/1, EP/I003312/1) (£1 330 092).

Other UK research council funded research

A snapshot of other recent and current research projects funded by the EPSRC is listed below. They indicate the breadth and size of UK research and capability on nuclear materials and were mainly funded via "responsive mode" and not necessarily in response to focused calls for proposals. This list does not include directly industrial funded projects, nor Ph.D. projects that are supported using Research Council funds provided to universities directly.

Some of these projects have been completed in recent years and others are currently in progress. The value of the award is also given. Further details (i.e. grant holders, partners and abstract) on each of these projects are available via www.epsrc.ac.uk, by consulting "Grants on the web" with reference to the project number (e.g. GR/R02481/01).

- Materials for fusion & fission power (EP/H018921/1) (£5 810 166).
- Atomistic Model of Structural Evolution of Nuclear Graphite (GR/R02481/01) (£191 545).
- In-Situ TEM Studies of Ion-Irradiated Materials (EP/E017266/1) (£642 772).

- Innovative Accelerator Technology for Accelerator-Driven Sub-critical Reactors (EP/F028121/1) (£142 341).
- Intergranular Segregation In Nuclear Pressure Vessel Ferritic Steels (GR/R05901/01) (£84 842).
- Ion irradiations of fusion reactor materials (EP/F004451/1) (£48 792).
- Predictive Modelling of Mechanical Properties of Materials for Fusion Power Plants (GR/S81193/01) (£103 747).
- Putting next generation fusion materials on the fast track (EP/E035671/1, EP/E035868/1) (£117 781 and £743 778).
- Stress and Creep Damage Evolution in Materials for Ultra-Supercritical Power Plant (EP/G068305/1) (£140 585).
- Thorium Fueled Accelerator-Driven Sub-critical Reactors for Power Generation (EP/G009864/1) (£251 248).
- UK Fusion Programme 2008-2010 (EP/G003955/1) (£50 988 172).
- Worldwide network of in-situ TEM/ion accelerator facilities (EP/F012853/1) (£38 887).
- Zirconium alloys for high burn-up fuel in current and advanced light-water-cooled reactors (EP/E036384/1, EP/E036481/1) (£626 742 and £267 815).

UK Generation IV experience

The UK has prior experience relevant to VHTRs and GFRs, on the basis of its gas-cooled advanced gas reactor experience and HTR research history, which includes the SFR. There is no prior UK experience in LFRs, MSRs and SCWR designs.

UK universities are involved in several current EURATOM projects. These include GETMAT (Generation IV) with the Universities of Liverpool and Edinburgh, CARBOWASTE (Generation II/IV) with the University of Manchester and various UK industrial partners, including Nexia, Amec and the Nuclear Decommissioning Authority (NDA), PERFORM 60 (Generation II/III) with the University of Manchester and UK industrial partners, including R-R, Serco and British Energy and STYLE, (Generation II/III) with the University of Manchester and UK industrial partners of R-R, Serco, British Energy and RAPHAEL and (Generation IV) with the University of Manchester, Serco, Amec and Nexia Solutions. Nexia Solutions is now part of the new UK National Nuclear Laboratory (NNL) and has expressed an interest in participating in Euratom programmes in the future.

UK National Nuclear Laboratory and Dalton Cumbrian Facility

The UK NNL is managed by a consortium of Battelle, Serco and the University of Manchester. Its central laboratory (Sellafield) has active laboratories (i.e. hot cells and glove boxes). An access agreement for UK university researchers has been established. Researchers need to obtain additional funding for this and this is now becoming available via the NDA, for example. The NNL's current focus is on Generation II/III issues, but Generation IV is on the NNL agenda and there is an intention to engage in FP7 programmes when the opportunity arises.

The Dalton Cumbrian Facility is a joint activity of the University of Manchester and the NDA. This research center will focus on irradiation damage and irradiation chemistry and will be located adjacent to the NNL Central Laboratory. It will have an ion source and gamma source for irradiation studies, with a suite of characterisation and analysis tools. These will be available in 2011-12. The facility will be accessible to UK universities.

Background documents

There are several documents that provide a useful introduction to UK experience and research capability in nuclear materials. These are listed below and the documents are available for consultation.

Materials UK energy review

Report 3 on Nuclear Energy Materials (2007) (www.matuk.co.uk/energy.htm) identified the UK core competencies as follows:

- mechanisms of in-service and in-repository corrosion and degradation of materials;
- predicting the behaviour of welded structures subjected to high temperatures and complex loadings;
- predicting irradiation damage effects in fission and fusion materials;
- development of new and improved methods for non-destructive monitoring and evaluation of materials in service.

EPSRC scoping workshop for future activities in nuclear power research and training

This workshop was held in June 2009. A copy of the report is available from the EPSRC website (www.epsrc.ac.uk). The aims of the workshop were the generation of a map of UK capabilities and identification of possible activities for the Research Councils to focus on. Input from the research community was sought regarding how to take the outputs forward. The workshop did not attempt to define a fully accurate map of capabilities and sought to generate a list of possible activities. It did not look to gain a consensus view on which was a priority. It was regarded as a starting point for wider community input. The workshop was attended by representatives from the university sector, national labs and other stakeholders. This workshop was followed by the October 2009 Nuclear Fission Call for Research Consortia described above.

EPSRC/STFC (Science and Technology Facilities Council) review of nuclear physics and nuclear engineering

This review was published in September 2009 and is available on the EPSRC website (www.epsrc.ac.uk). Its terms of reference were:

- to review the scope of RCUK funded activity (Research Councils UK) and training in nuclear physics and nuclear engineering in the UK;
- to identify the skills and expertise with relevance to future economic impact in related application areas;
- to comment on the ability of the scope and volume of the current nuclear physics and nuclear engineering activity to deliver these skills and expertise;
- to identify any changes required in the scope or priorities of the nuclear physics and nuclear engineering activity in the UK;
- to comment on UK competitiveness in skill and expertise provision.

It raised concerns about the need for a coherent roadmap describing the nuclear landscape to better position the UK's research and training needs within a resurgent nuclear sector and the need to ensure an appropriate position for the UK in international research and development initiatives in advanced reactor systems and fuel cycle development.

The report made 19 recommendations, of which two are very relevant to Gen-IV materials research. These were for the:

- "…UK to reinvigorate its involvement in the Generation IV International Forum and other related international initiatives";
- "…Research Councils to encourage research into Generation IV technologies and related fuel cycle topics".

Other UK research activity relevant to Generation IV fission materials

A search of published journal articles on nuclear materials relevant to Generation IV reactor designs has been done for the purpose of identifying UK centers with research programmes that may be relevant to the EERA activity. It is possible that some relevant UK research has not been included.

The search was restricted to publications since 2004 that included UK universities among the author affiliations, in order to identify research groups that were currently active. Distinctions have not been made whether the UK University was leading or participating in international collaborations. Information on the source of funding for these activities may be available in the publications, but has not been collated.

The search focused on materials research conducted on topics such as ODS alloys, F/M alloys, SiC-SiC composites and irradiation damage, as these were relevant to the EERA activity on nuclear materials. Corrosion research on Ni-base alloys or stainless steels is largely undertaken at various universities, including Manchester, Birmingham, Oxford and Imperial College, London. Graphite research is concentrated at Manchester University, with research also at Hull, Birmingham, Cardiff and Surrey. The University of Oxford has a significant activity on ODS alloys, with work on ferritic/bainitic steels at Cambridge and some work on bonding of ODS steels at Brunel, for example. Loughborough, Liverpool and Surrey have research on irradiation damage effects in ODS and F/M steels, including segregation effects at grain boundaries. There has been work on SiC-SiC composites at Imperial College, London, where modelling of irradiation damage effects (mostly on fuels) is also a current focus. Other work has been done at St Andrew's University on high-temperature vapourisation of ODS alloys.

A selection of article titles is provided below as an indication of the type of work being done:

- Design of nano-composites for ultra-high strengths and radiation damage tolerance.
- Effects of ion irradiation on the microstructure of an ODS/Fel2Cr alloy.
- Radiation damage theory: Past, present and future.
- Raphael: Achievements within the VHTR materials and components programmes.
- Fabrication of CNT-SiC/SiC composites by electrophoretic deposition.
- Infiltration of a 3-D fabric for the production of SiC/SiC composites by means of electrophoretic deposition.
- Grain boundary chemistry before and after ion implantation in ODS steels.
- Dynamic observations of heavy-ion damage in Fe and Fe-Cr alloys.
- Irradiation-induced precipitation modelling of ferritic steels.

- Effect of heat treatment on microstructure and hardness of Eurofer 97, Eurofer ODS and T92 steels.
- Microstructural characterisation of Y2O3 ODS-Fe-Cr model alloys.
- TEM characterisation of heavy-ion irradiation damage in FeCr alloys.
- Void formation in ODS EUROFER produced by hot isostatic pressing.
- *In situ* transmission electron microscopy and ion irradiation of ferritic materials.
- Microstructural characterisation of Y2O3 ODS-Fe-Cr model alloys.
- Void formation in ODS EUROFER produced by hot isostatic pressing.
- Dislocation-obstacle interactions at atomic level in irradiated metals.
- Generation IV nuclear power: A review of the state of the science.
- Modelling of radiation damage in Fe-Cr alloys.
- Core/shell structures of oxygen-rich nanofeatures in ODS Fe-Cr alloys.
- Development of a Young's modulus model for Gilsocarbon graphites irradiated in inert environments.
- Electrophoretic deposition in the production of SiC/SiC composites for fusion reactor applications.
- RAPHAEL: Materials and components highlights from the HTR FP5 and FP6 programmes.
- Effects of annealing and ion implantation on the nano-structure of the ODS Eurofer 97 steel.
- Oxide nanoparticle dispersion in an ODS/Fe12Cr model alloy.
- Heavy-ion irradiations of Fe and Fe-Cr model alloys Part 1: Damage evolution in thin-foils at lower doses.
- Heavy-ion irradiations of Fe and Fe-Cr model alloys Part 2: Damage evolution in thin-foils at higher doses.
- Irradiation-induced grain boundary chromium microchemistry in high alloy ferritic steels.
- Designing optimised experiments for the international fusion materials irradiation facility.
- Chromium vapourisation of the ferritic steel Crofer22APU and ODS Cr5Fe1Y2O3 alloy.
- Modelling of radiation damage in Fe-Cr alloys.
- Effects of neutron irradiation on precipitation in reactor pressure vessel steels.
- Irradiation assisted grain boundary segregation in steels.
- The role of irradiation-induced phosphorus segregation in the ductile-to-brittle transition temperature in ferritic steels.
- A future for nuclear power.
- Effects of neutron irradiation on precipitation in reactor pressure vessel steels.
- A comparison of microstructural developments in TLP diffusion bonds made using ODS Ni alloy.

- Radiation-and thermally-induced phosphorus inter-granular segregation in pressure vessel steels.
- Modelling dislocation-obstacle interactions in metals exposed to an irradiation environment.
- *In situ* transmission electron microscopy investigation of radiation effects.
- Modelling of equilibrium grain boundary solute segregation under irradiation.
- Electron microscopy of defects in oxide-dispersion strengthened ferritic alloys.
- Confinement of interstitial cluster diffusion by oversized solute atoms.
- A multi-scale approach to irradiation-induced segregation at various grain boundaries.
- Suppression of interstitial cluster diffusion by oversized solute atoms.
- The use of electrophoretic deposition for the fabrication of ceramic matrix composites.
- Multi-scale modelling of radiation damage in metals: From defect generation to material properties.

Appendix II: List of materials database (2012)

AFCI Materials Handbook

- Content description:
  - This handbook provides data on radiation effects in ferritic/martensitic steels, austenitic steels and other metals after irradiation in high energy proton and neutron fluxes. The handbook was originally developed for the Accelerator Production of Tritium Programme and has been modified (Rev. 5) to support the Advanced Fuel Cycle Initiative.
- Country: US
- Storage media: CD
- Accessibility:
  - (controlled distribution) available to US citizens and non-US citizens with DOE approval

Nuclear Systems Materials Handbook

- Content description:
  - Materials properties and design data from the US fast reactor development programme.
- Country: US
- Storage media: hardcopy
- Accessibility:
  - (controlled distribution) available to US citizens and non-US citizens with DOE approval

Multimegawatt Materials Handbook

- Content description:
  - Datasheets containing properties of materials considered for space nuclear systems. This is an incomplete database that has not been kept current.
- Country: US
- Storage media: pdf file retrievable from the American Nuclear Society ANST Division web site (http://www-rsicc.ornl.gov/ANST_site/multimega.pdf)
- Accessibility: open

NMIS

- Content description:
  - Broad coverage of materials properties in many systems that may be used as background for understanding materials performance in nuclear systems.
- Country: Japan
- Storage media: web, http://mits.nims.go.jp/db_top_eng.htm
- Accessibility: open with registration.

Generation IV Materials Handbook

- Content description:
  - This handbook provides data on structural materials for Generation IV reactor systems, particularly the very-high temperature gas-cooled reactor (VHTR). The handbook was developed for the US. DOE to enable storage and secure transfer of materials data among signatories to the Generation IV International Forum's (GIF's) materials exchange agreements. It is currently approved for the GIF VHTR materials project arrangement.
- Country: US
- Storage media: server and web-based
- Accessibility:
  - (controlled distribution) available to US citizens and non-US citizens with DOE approval, in accordance with GIF data sharing requirements

Fusion Network

- Content description:
  - This site provides a link to multiple handbooks developed by the US fusion programmes including SiC, Pb-Li, ITER, Structural Materials, Breeder Materials, Multiplier Materials, and Other Materials Handbooks.
- Country: US
- Storage media: web (http://fusionnet.seas.ucla.edu/fusionnetwork/)
- Accessibility: open

BDEM

- Content description:
  - This handbook provides data on mechanical properties, irradiation effects on ferritic/martensitic steels, austenitic steels and other metals. This database is used for the RCC-MR.
- Country: France
- Storage media:
- Accessibility:
  - (controlled distribution) extractions of the database can be sold upon approval

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Status Report on Structural Materials for Advanced Nuclear Systems

Materials performance is critical to the safe and economic operation of any nuclear system. As the international community pursues the development of Generation IV reactor concepts and accelerator-driven transmutation systems, it will be increasingly necessary to develop advanced materials capable of tolerating the more challenging environments of these new systems. The international community supports numerous materials research programmes, with each country determining its individual focus on a case-by-case basis. In many instances, similar alloys of materials systems are being studied in several countries, providing the opportunity for collaborative and cross-cutting research that benefits different systems.

This report is a snapshot of the current materials programmes supporting the development of advanced concepts. The descriptions of the research are grouped by concept, and national programmes are described within each concept. The report provides an overall sense of the importance of materials research worldwide and the opportunities for synergy among the countries represented in this overview.

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