

7. Chapter 7. Cladding & Structural Material Effects

As discussed previously in this document and more widely throughout various training courses and technical documentation - LMFRs place additional challenges on in-core components due to the higher operating temperatures, neutron flux, fuel burn-up targets and liquid metal coolants. The pursuit of materials that support the enhanced operation of LMFRs has led to significant research and development to gather information on both established materials, as well as more advanced materials over the last 10-20 years.

7.1. Common Irradiation Effects, across LMFRs

7.1.1. Austenitic and Ferritic/Martensitic (F/M) steels

Internationally, significant irradiation experience has been gained with both austenitic and ferritic/martensitic (F/M) steels that results in relatively large and public domain databases regarding irradiation induced swelling and creep behaviour of optimised grades. The experience of austenitic steels owes to their good high-temperature strength and corrosion resistance; this has also led to them being established as structural materials. Ferritic steels are promising core materials because in general their swelling resistance is better than that of austenitic steels as well as having higher thermal conductivity, lower thermal expansion [J 7.5]. On the other hand, ferritic steels are inferior compared with austenitic steels in terms of high-temperature strength.

This experience allows predications as to likely behaviour to be made with confidence. However material and thus operational limits due to either dose (austenitic) or temperature (F/M) indicate that current grades may not be sufficient to achieve the high burn-ups targeted by many modern and especially future LMFRs. This has led to the development of many advanced austenitic or F/M steels as well as alternative “Advanced Claddings” to overcome these limitations and these advances as part of national programmes are discussed in the following sections.

7.1.2. Advanced Claddings

To overcome some of these irradiation and corrosion induced limitations, significant research has been conducted internationally into advanced claddings that entail:

- **Oxide Dispersion Strengthened (ODS) F/M steels**, development was started in the United States of America as advanced materials that had superior swelling resistance and high-temperature strength than legacy F/M steels [J 7.7, 7.8]. The important feature of ODS ferritic steels is they're strengthening by a number of very fine and stable oxide particles. The enhanced swelling resistance was confirmed in the PIEs of FFTF/MOTA irradiation as shown in Fig. 7.2 [7.9].

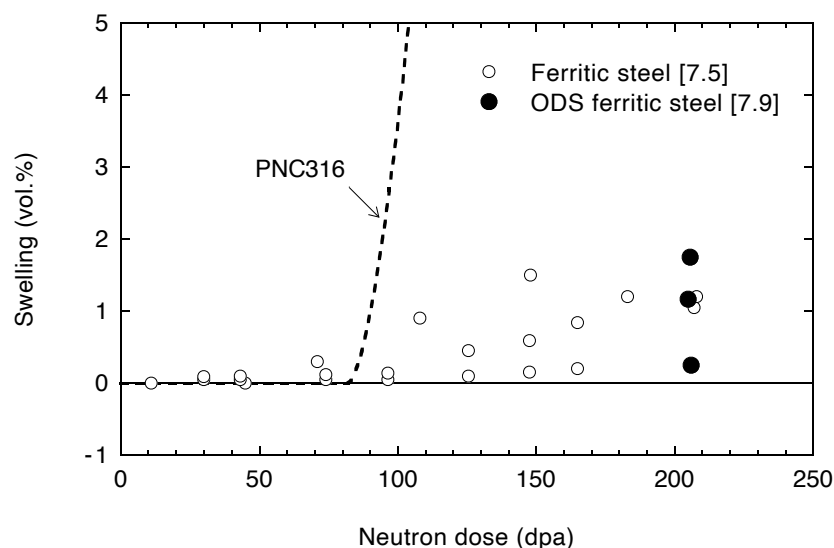


Fig. 7.2 Swelling of austenitic (PNC316) and ferritic (ODS and not ODS) steels as a function of neutron dose

- **High Entropy Alloys (HEAs)**, a fundamentally new material concept, rely on the high configurational entropies of mixing of typically four or more metallic elements in near equal quantities to stabilise solid solution phases and limit the formation of potentially embrittling intermetallic phases as part of the four 'core' effects described by Yeh *et al.* (REFERENCE Yeh, "Recent Progress in High-Entropy Alloys" *Annales De Chimie - Science des Matériaux*, vol 31, page 633-648, **2006** and Cantor *et al.* "Microstructure development in equiatomic multicomponent alloys" *Materials Science and Engineering A*, Vol 375-377, page 213-218, **2004**). Irradiation data is principally limited to ion rather than neutron irradiation data due to the relative maturity of HEAs; with significant gaps existing - especially at doses and temperatures applicable to LMFRs [REFERENCE - Pickering *et al.* in "High-Entropy Alloys for Advanced Nuclear Applications" *Entropy*, **2021**, 23(1), 98].
- **FeCrAl alloys**, were developed for environments that required oxidation resistance at high temperatures in the 1960s [REFERENCE - Field *et al.*, *JNM*, **2017**, vol 489, page 118-128]. FeCrAl alloys were irradiated in the Dounreay Fast Reactor and more recent low dose irradiations have indicated minimal swelling or creep [REFERENCE - Little, *JNM*, **1979**, vol 87, page 11-24 and Field *et al.* "Handbook of the Material Properties of FeCrAl Alloys for Nuclear Power Production Applications" ORNL/SPR-2018/905 Rev. 1 ORNL USA, August **2018**].
- **High-Silicon or Alumina Forming Austenitic (AFA) steels** and Coated Claddings are specific materials under consideration for Heavy Liquid Metal-cooled Fast Reactors (HLM-FRs) with either a significant silicon or alumina content in the base material, or deposited as a corrosion resistant coating enabling the use of more established austenitic or F/M steels.
- **Ceramics claddings** including silicon carbide fibres imbedded in a silicon carbide matrix (SiC/SiC) are interesting candidates for use in demanding nuclear environments with SiC/SiC showing similarities with monolithic SiC. Detailed summaries of SiC/SiC and SiC are contained within [REFERENCE Snead *et al.* *JNM*, **2007**, vol 371, page 239-377 and Koyanagi *et al.* "SiC/SiC Cladding Materials Properties Handbook" ORNL/TM-2017/385" ORNL, USA, August 2017]

A scarcity in available irradiation data limits the ability to predict irradiation induced creep and swelling for advanced materials such as ODS and SiC/SiC with any reliability; however what research has been conducted and disseminated indicates that such grades may well meet the needs of future LMFR designs. More innovative material grades such as FeCrAl and HEAs unsurprisingly have even smaller datasets, but again the information that does exist indicates that such materials may well be sufficient for use in future LMFRs.

7.2. Liquid metal specific corrosion considerations

7.2.1. Liquid sodium considerations

It is important to re-iterate that as multiple SFRs have been built and have, or indeed are currently, operating then light metal induced corrosion challenges can be readily mitigated. The primary interaction of steels with liquid sodium is dissolution and mass transport or oxidation, depending on coolant oxygen levels. Thus careful oxygen and/or temperature control can avoid any material corrosion, or the rate is well within allowable design limits to enable efficient operation of multiyear fuel cycles (REFERENCE - Yoshida and Furukawa "Chapter 21: Corrosion issues in sodium-cooled fast reactor (SFR) systems" in *Nuclear Corrosion Science and Engineering*, Woodhead Publishing Ltd., **2012**, page 773-806).

7.2.2. Heavy Liquid metal considerations

Lead and lead-bismuth eutectic cooled fast reactors offer many advantages, but prior to their widespread deployment several challenges must first be overcome. This publication will only consider fuel materials, and section cladding in particular and thus is concerned with the corrosive properties of liquid heavy metals principally the solubility of traditional alloying elements which are significantly higher for heavy liquid metals than in light liquid metal coolants. See [REFERENCE "IAEA Nuclear Power Series NP-T-1.6 - Liquid Metal Coolants for Fast Reactors cooled by sodium, lead and lead-bismuth eutectic" **2012**, Vienna, Austria"] for a more detailed overview. This requires different approaches as mass transfer, surface modification and as already mentioned, element dissolution are much more significant. Two key factors imperative to control of corrosion in heavy liquid metal are oxygen content and temperature of the coolant [REFERENCE Schroer *et al.*, "Aspects of minimising steel corrosion in liquid lead-alloys by

addition of oxygen”, *Nuclear Engineering and Design*, **2011**, vol 241., page 4913-4923] and with careful control of these, corrosion can be controlled. Different alloys, with a different base metal composition will have slightly different oxygen contents required to form passivating effects on the coating. A key concern of cladding corrosion is that the main corrosion products accumulate in cold spots and on the surface of the reactor pool and thus will have to be regularly removed should it not be possible to minimise.

Question for CM, would a figure be useful to explain corrosion, transition and oxidation phases?

7.3. Conventional nuclear steel cladding

7.3.1. Austenitic steels

Multiple SFRs have been built and operated across the globe and have historically utilised optimised versions austenitic type 316 stainless steel such as PNC316 (REFERENCES Japan 7.1+7.2), 15-15Ti (REFERENCE Maillard et al., “Swelling an Irradiation Creep of Neutron-Irradiated 316Ti and 15-15Ti Steels” in *Effects of Irradiation on Materials: 16th International Symposium ASTM STP 175*, ASTM, **1994**, page 824-837) and Nimonic alloy PE16 (REFERENCE - https://inis.iaea.org/search/search.aspx?orig_q=RN:22004598). Indeed, PE16 with its high nickel content was even proposed as the fuel cladding for jointly designed, but never built, European Fast Reactor (EFR).

JAEA developed PNC316 (16Cr-14Ni-0.5Mo-P, B, Ti, Nb) as a modified 316 stainless steel for SFR cladding material with enhanced swelling resistance and high-temperature strength. Several techniques, including 20% cold work inducing dislocations and micronizing carbides due to addition of P, B, Ti and Nb, were applied to PNC316 to further enhance it [7.1, 7.2]. PNC316 has been applied to Joyo and Monju driver fuels, and several post-irradiation examinations (PIEs) have been performed. On the other hand, PNC316 has a limit in dimensional changes due to the specific swelling of austenitic matrix, as shown in Fig. 7.1 [7.3, 7.4]. In order to further improve the swelling property of austenitic steels, JAEA developed PNC1520 based on PNC316 with an increased Ni content of 20 at.% (15Cr-20Ni-2.5Mo-1.7Mn- P, B, Ti, Nb). However, no significant improvement in swelling property was observed during irradiation testing [7.4].

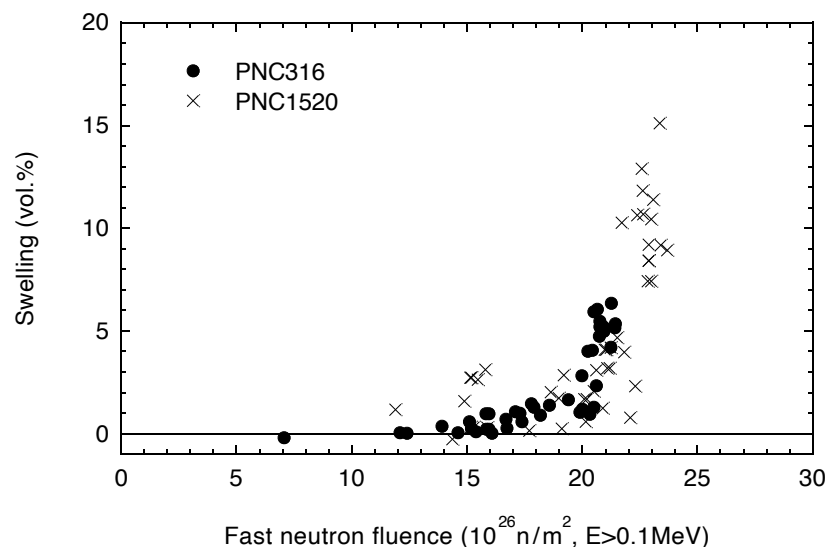


Fig. 7.1 Swelling of PNC316 and PNC1520 cladding as a function of fast neutron fluence

In heavy liquid metals, corrosion of austenitic stainless steels is typically viewed to be controlled via two main oxidation process [REFERENCES Angiolini et al. “IAEA-CN245-227 - Towards a new approach for structural material of lead fast reactors” Vienna, Austria; Muller et al, “Investigation on oxygen controlled liquid lead corrosion of surface treated steels” *Journal of Nuclear Materials*, **2000**, vol 278, page85-95]. Most work internationally has shown that 15-15Ti typically has a tendency to develop localised (non-uniform) oxidation spots which maybe due to the chemical composition of the austenitic steel, or the cold-working - further work is required to clarify this

[REFERENCE Fazio et al “Compatibility Tests of steels in molten lead and lead-bismuth” *Journal of Nuclear Materials*, **2001**, vol 296, page 243-248].

Corrosion studies in the Czech Republic have confirmed formation of a thin protective oxide film with a tendency to develop deep oxidation spots with limited (maximum depth of 3µm) Pb penetration for 15-15Ti (see Figure xx.5) . Trials have been carried out on the MATLoo loop at the Rez Research Centre. This has enabled testing in liquid lead at 480°C in the test section, 400°C in the cold leg with a filter simulating temperature gradient between the core and steam generator, 1.6m/s flow velocity and oxygen content $\sim 1\text{E-}7$ wt.% O₂ for up to 16,000 hours and the claddings analysed for material loss and surface oxide composition. This research is in support of the ALFRED (Advanced Lead Fast Reactor European Demonstrator) project where an oxygen protection strategy is intended to control corrosion. Research has shown that 15-15Ti can protect itself with formation of a spinel film for at least 16,000 hours and limit material loss to the tens of micrometers, although this does lead to a depletion of oxide forming elements from the surface of the material and temperatures of circa 480-500°C. At higher temperatures, of 550°C or 600°C, Müller et al demonstrated clear LBE penetration into austenitic steels and after 4000 h at 600°C the heavy liquid metal had penetrated the cladding to depths of more than 100 µm and completely destroyed the 316L austenitic steel sample [REFERENCE Muller et al, “Behavior of steels in flowing liquid PbBi Eutectic Alloy at 420-600C after 4000-7200 h: treated steels” *Journal of Nuclear Materials*, **2004**, vol 335, page 163-168].

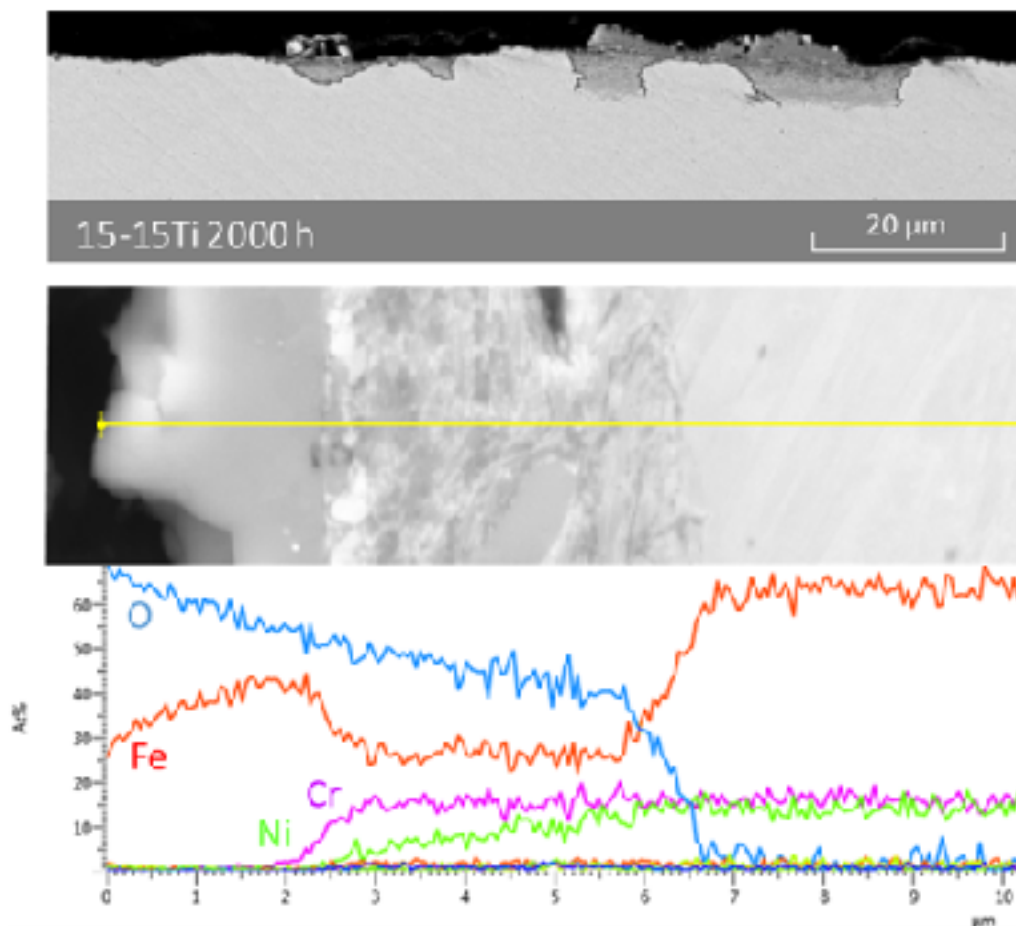


Fig. xx5 Example of localized oxidation of 15-15Ti after 2000h exposure in the oxygen-controlled liquid lead.

In summary, austenitic steels have been proven and validated for us in several sodium-cooled fast reactors with research ongoing to enable them to operate to burn-ups targeted by modern fast reactor designs. For heavy liquid metal-cooled systems, austenitic steels have shown they can protect themselves from corrosion under optimised oxygen partial pressure and temperature conditions through formation of a protective film but additional studies are required to understand the effects of surface finish as well as austenitic claddings under conditions more representative of those intended for heavy liquid metal-cooled reactor designs.

7.3.2. F/M

F/M steels based on a 9Cr-1Mo2 and 12Cr-1Mo3 base composition have been investigated as FR materials across the world [REFERENCE IAEA, “Liquid Metal Coolants for Fast Reactors cooled by Sodium, Lead and Lead-Bismuth Eutectic” IAEA Nuclear Power Series NP-T-1.6, Vienna, Austria, IAEA, **2012**; and Klueh and Harries “High-chromium ferritic and martensitic steels for nuclear applications: ASTM Monograph 3” West Conshohocken, PA, USA: ASTM, **2001**]. Finding use in wrappers given the lower operating temperatures, apart from HT-9 which has been used as fuel cladding. Indeed, F/M steels can have lower corrosion rates than austenitic steels under similar oxygen and temperature conditions yet are limited by temperature.

KAERI has developed advanced cladding material (FC92) which has superior thermal creep property to conventional cladding material, HT9. Optimization of alloying element has been carried out based on the Gr.92 material by decreasing C content, increasing Nb content to initiate the number of MX precipitates, adding Ta, and balancing B / N content to increase the stability of precipitate. Out-of-pile thermal creep property of FC92 material revealed that creep strength of FC92 was higher than HT9 material by 30%, which indicates the increase of thermal margin for metal fuel rods. Further thermo-physical comparisons have shown that the properties of FC92 behave similar to HT9.

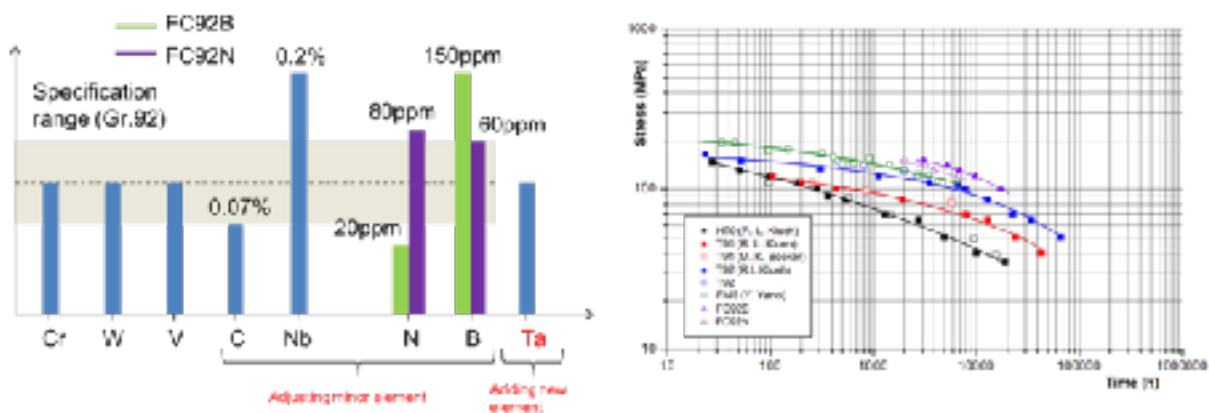


Fig. 1 Design of FC92 alloy and out-of pile creep-rupture property

To verify performance, KAERI undertook an irradiation test programme for fuel cladding under fast neutron flux, in the BOR-60 reactor, with irradiations taking place between 2014 and 2019. This consists of 2 test rigs Material Test Rig (MTR)-1 and 2), which are characterized by irradiation temperature ($600 \pm 30^\circ\text{C}$ in MTR-1, $650 \pm 32^\circ\text{C}$ in MTR-2) and maximum attainable doses at the end of the irradiation test of 45 and 75dpa, respectively. Full Post Irradiation Examination (PIE) of FC92 cladding materials are planned, but interim inspection indicated that in-reactor creep strength of FC92 was 30% higher than HT9.

Heavy liquid metals with very low oxygen contents have been reported to rapidly dissolve certain F/M steels and at temperatures around 550°C can even be of the order of mm/year [REFERENCE - Cabet et al., “Ferritic-martensitic steels for fission and fusion applications” *Journal of Nuclear Materials*, **2019**, vol 523, page 510-537]. Variation of the constituent elements reduces this, as seen during the testing of commercial specimen 9Cr-1Mo (T91 equivalent) in the MATLoo loop at the Rez Research Centre, Czech Republic under the same conditions as 15-15Ti (i.e. liquid lead at 480°C in the test section, 400°C in the cold leg with a filter simulating temperature gradient between the core and steam generator, 1.6m/s flow velocity and oxygen content $\sim 1\text{E-}7$ wt.% O_2) for up to 16,000 hours. The MATLoo testing noted that the 9Cr-1Mo (T91 equivalent) F-M steel suffered oxidation and the exfoliated magnetite scale entered into the circulating media and at the tested conditions had material loss in the tens of micrometers range. As for the 15-15Ti, the work noted that the effect of surface finish on the corrosion performance is an important factor that needs to be better understood. Increasing the Cr-content of the alloy enhanced the corrosion resistance, but greater Al or Si contents led to greater resistance - though less than for the corresponding High-Si or AFA (discussed later).

7.4. Advanced nuclear claddings

7.4.1. Advanced Claddings

1. ODS

To address the temperature limitations of F/M steels, international research has been conducted into ODS versions. The retention of higher temperature strength through introduction of fine oxide particles has been reported and described by multiple authors [REFERENCE Cheon et al., “SFR evaluation: core materials”, *JNM*, **2009**, vol 392, 324-330; Ukai et al, “Oxide dispersion-strengthened/ ferrite-martensite steels as core materials for Gen. IV Nuclear Reactors. Chapter 10” in *Structural Materials for Generation IV Nuclear Reactors*, Elsevier, **2017**, page 357-414; Hoelzer, “History and outlook of ODS/NFA ferritic alloys for nuclear applications” *Trans. American Nuclear Soc.*, vol 118, page 1587-1590, **2018**”.

In Japan, JAEA has been developing ODS steels for three decades and regard the 9Cr tempered martensitic ODS steel as the main material for cladding tubes of FRs in Japan with a base chemical composition of Fe-9Cr-2W-0.2Ti-0.35Y₂O₃ [7.10-7.13]. The 9Cr concept aims to avoid the risk of aging and/or irradiation-induced embrittlement. Furthermore, recent research has revealed that 9Cr-ODS steel has three times the tensile strength of conventional steel at an ultrahigh temperature of 1200°C. Thus, the ODS steel cladding tubes have a great advantage in improving the economic efficiency by increasing the burn-up and safety in severe accidents [7.18].

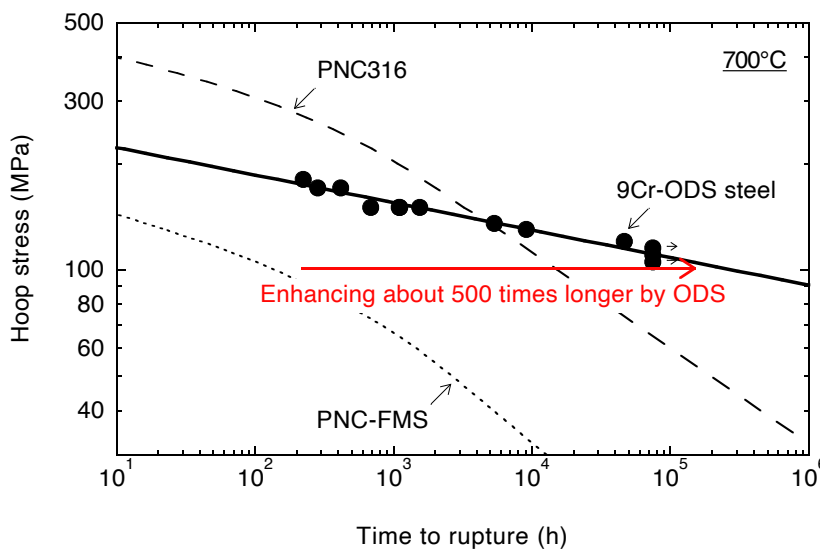


Fig. 7.4 Comparison of creep strength of cladding tubes of 9Cr-ODS steel and other materials

JAEA is continuing to evaluate the mechanical properties of ODS to establish the material strength standards, and develop mass production technology. In the future, JAEA will look to evaluate the irradiation effects of ODS cladding tubes and ultimately demonstrate performance via fuel pin irradiation tests.

In Korea, KAERI have been developing “advanced radiation resistant oxide dispersion strengthened (ARROS) steel. Two candidate materials with base chemical compositions of (in wt%) 10Cr-1Mo-0.5Mn-0.1V-0.25Ti-0.35Y₂O₃ and 10Cr-1Mo-0.2Ni-0.05Zr-0.2Ti-0.35Y₂O₃ were selected as representative materials. Both shows excellent tensile and creep strengths at 700°C as shown in Fig. 7.5. While more studies are required for the irradiation performance of ODS steel.

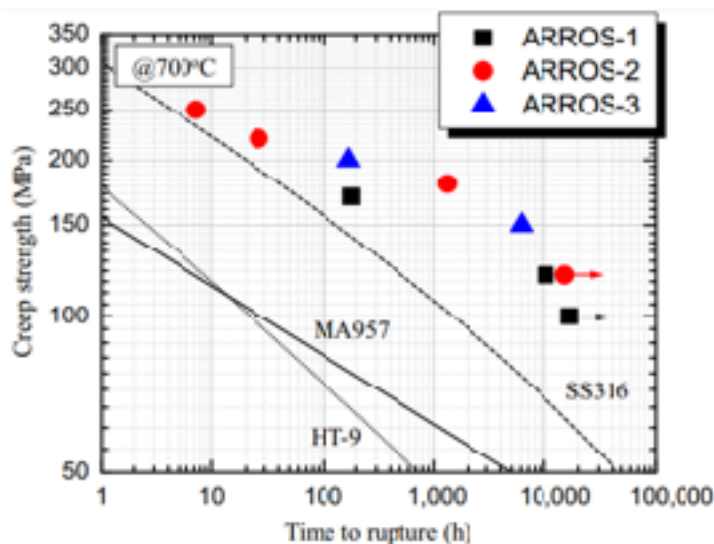


Fig. 7.5 Creep rupture strengths of advanced radiation resistant oxide dispersion strengthened (ARROS) and conventional steels.

Opportunities exist for the use of ODS in heavy liquid metal coolants, but significant research is required to validate its compatibility with Pb/LBE through longterm corrosion testing.

Awaiting paragraph review and contribution by France (& others) on ODS as well as wider contribution to section.

2. High Entropy Alloys

Arguably, the most extensive public domain study on the corrosion resistant properties of HEAs in liquid metals is the work by Voyevodin (REFERENCE Voyevodin et al. "Evaluation of the High-Entropy Cr-Fe-Mn-Ni Alloys compatibility with a liquid lead coolant" in *Technical Meeting on Structural Materials of Heavy Liquid Metal cooled Fast Reactors*, Vienna, 2019) who focused on lead melts and used CrFeMnNi based HEAs. The main mechanisms of corrosion damage were proposed for temperatures of 480°C and 580°C, with a recommendation that these alloys could even be used with a lead melt of up to 500°C. For higher temperatures, Voyevodin recommended ODS HEAs along with further studies to gain a comprehensive understanding of the relevant mechanisms. Similar observations were seen by Huang et al in liquid LBE [REFERENCE 71 Huang et al. "Liquid Metal Embrittlement Susceptibility of a High-Entropy Alloy exposed to oxygen-depleted liquid lead bismuth eutectic at 250 and 350 °C" *JNM*, **2020**, vol 528, 151859]. This led Huang to suggest that cladding for LBE, though similarly it is likely to apply to pure Pb-cooled systems as well, should use less Ni and Mn to minimise the process of dissolution corrosion - even considering the relative sluggish diffusivity of Mn and Ni in HEAs.

Reich to address this includes the recent European Energy Research Alliance - Joint Programme on Nuclear Materials (EERA-JPM) and the High Entropy Alloys for Nuclear Applications (HEAFNA) pilot programme launched by KIT (Karlsruhe Institute of Technology) [https://snetp.eu/wp-content/uploads/2021/02/Presentation_A.-Weisenburger.pdf] as well as several academic endeavours in the UK [REFERENCE - Pickering et al. in "High-Entropy Alloys for Advanced Nuclear Applications" *Entropy*, **2021**, 23(1), 98]

3. FeCrAl Alloys

The relatively low thermal and electrical conductivity of these materials meant they have only recently been considered as cladding candidates - but the corrosion resistance of these materials has made them candidates for heavy liquid metal cooled reactors [Field et al. "Handbook of the Material Properties of FeCrAl Alloys for Nuclear Power Production Applications" ORNL/SPR-2018/905 Rev. 1 ORNL USA, August **2018**]. Comparison of commercially available FeCrAl alloys as well as modified variants has been carried out with promising corrosion resistant results occurring [REFERENCES Weisenburger et al., "Oxide scales formed on Fe-Cr-Al-based model alloys exposed to oxygen containing molten lead" *Journal of Nuclear Materials*, **2013**, vol 437, page 282-292; Domstedt et al., "Corrosion studies of a low alloyed Fe-10Cr-4Al steel exposed in

liquid Pb at very high temperatures” *Journal of Nuclear Materials*, **2020**, vol 531, pg 152022]. It is important to note that difficulties may exist given the aforementioned properties as well as challenges associated with processing and irradiation.

Awaiting paragraph review and contribution by US (& others) on FeCrAl, as well as wider contribution to chapter.

4. High-Si and Alumina-Forming Austenitic Steels

Above 550°C, higher oxygen contents of the coolant do not lead to formation of a protective oxide onto the surface of conventional austenitic steels prompting alloying of steels with elements that form strongly passivating oxides even at low oxygen concentrations [REFERENCE Shi et al. “Corrosion resistance and microstructure stability of austenitic Fe-Cr-Al-Ni Model Alloys exposed to oxygen-containing molten lead”, *Journal of Nuclear Materials*, **2019**, vol 524, page 177-190]. The aim of the addition of Al has been to form a strong external layer of oxide rather than oxidise the bulk material and Shi et al have identified a base (wt.%) for AFAs of (20-29)Ni-(15.2-16.5)Cr-(2.3-4.3)Al-Fe and also recommended other alloying elements to further enhance properties. However, the high Ni-contents make the materials expensive and research into alternative elements has been conducted; as has studies into “reactive elements” such as Y and Zr, that could be used to assist in formation of the alumina [REFERENCE Weisenburger et al., “ID-21 Development of alumina forming materials for corrosion mitigation in heavy liquid metal cooled nuclear reactors” in *IAEA Technical Meeting on Structural Materials for Heavy Liquid Metal cooled Fast Reactors*, October 2019]. It is important to highlight that studies have shown “high” wt% O₂ content of the coolant is required to form a protective alumina layer on the AFA. Awaiting review and further contribution from Sweden (& others?) on AFA.

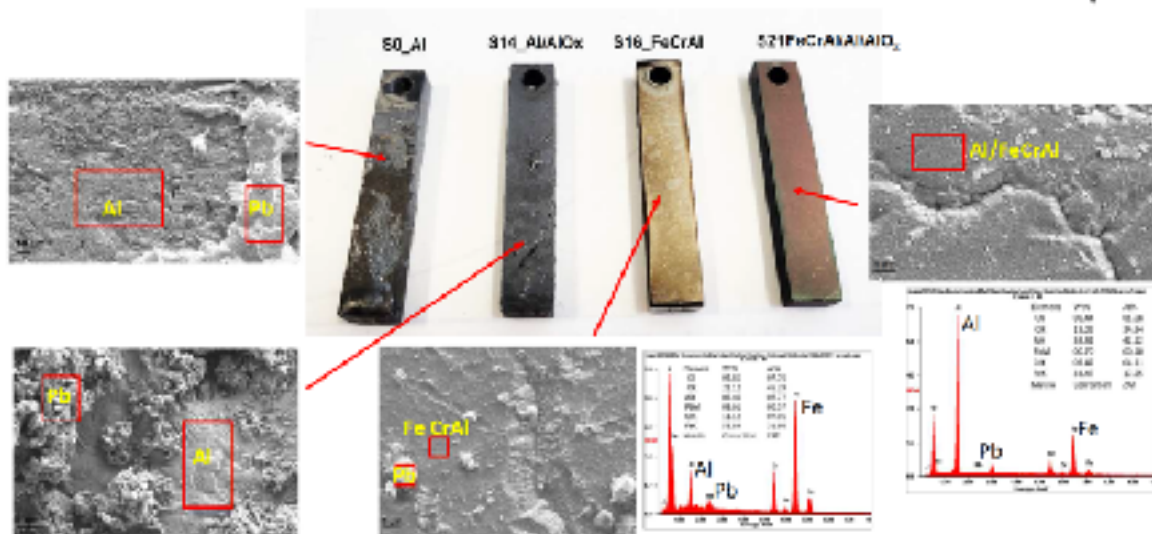
In Russia, fast reactor claddings designated as EP302/EP302M that have higher than normal levels of Si appear to be well suited for heavy liquid metal exposure. Long term testing (50,000 hours) at 600°C in LBE (unknown O₂ content) didn’t adversely impact mechanical properties and the material even retained its shine. [REFERENCES - Dedul et al “ID-16 Some new R&D focus in structural materials licensing for the SVBR-100 reactor facilities” in *IAEA Technical Meeting on Structural Materials for Heavy Liquid Metal cooled Fast Reactors*, October 2019; Shutko “Research of corrosion behaviour of steam generator tubes for lead-cooled power unit” in *IAEA-TECDOC-1978 Structural Materials for Heavy Liquid Metal cooled Fast Reactors - proceedings of a technical meeting*, October 2019]. Awaiting paragraph review by Russia on EP302(M) and wide contribution to the SFR and LBE/LFR sections.

5. Coatings

The addition of corrosion resistant elements to austenitic or F/M steels has already been discussed, but varying the elements can have unintended consequences by increasing thermal or radiation embrittlement as well as making manufacturing more challenging. An alternative to this strategy is achieving corrosion control through depositing a protection coating layer and the use of either established or advanced cladding materials. Various coatings have been investigated including metal alloys, (Si- or Al-based) oxides, carbides and nitrides deposited by either plasma spray methods, Chemical Vapour Deposition, Pulsed Laser Deposition, and Physical Vapour Deposition - specific examples include [REFERENCES Engelko et al “Surface Modification/ alloying using intense pulsed electron beam as a tool for improving the corrosion resistance of steels exposed to heavy liquid metals” *Journal of Nuclear Materials*, **2011**, vol 415, page 270-275; Miorin et al., “Al rich PVD protective coatings: A promising approach to prevent T91 steel corrosion in stagnant liquid lead” *Surface Coating Technology*, **2019**, vol 377, page 124890].

In the UK, research has focused on deploying coatings onto surrogate materials using magnetron sputtering (a type of PVD) to ensure a homogeneous corrosion resistant layer [REFERENCE - Kulczyk-Malecka, “A1-1-MoM-3 Surface Coatings for Improved Corrosion Resistance of Steels in Heavy Liquid Metal Coolants” in *ICMCTF 2022*, San Diego, California USA]. Samples with alumina and FeCrAl coatings with and without interlayers have been subjected to molten Pb-testing as well as thermal cycling studies. No excessive Pb inward diffusion was observed and studies are continuing, including at the Bagnor University Liquid Lead Erosion/corrosion Test Loop.

Molten Lead Test Results



Example of UK manufactured coated cladding following molten lead exposure.

6. Ceramics

It is possible for several different kinds of reaction to occur between SiC-based ceramics, both SiC/SiC and monolithic SiC and sodium and these are not discussed in detail here, instead see [REFERENCES: J. Braun, "Consequences des interactions physico-chimiques resultants de l'utilisation d'un material composite SiC/SiC dans un environnement caracteristique des reacteurs nucleaires du futur" PhD Thesis, **2014**, 01236626; Munro et al, "Corrosion characteristics of silicon carbide and silicon nitride" *Journal of Research of the National Institute of Standards and Technology*, **1993**, vol 98, no 5, page 607-631]. Given these reactions and the need for significant work to determine the values of oxygen and coolant temperature necessary to bound active and passive regions for SiC/SiC in liquid sodium. In general, SiC/SiC can be viewed as having good compatibility with a modern SFR environment, as long as material purity can be guaranteed and validation tests across a range of conditions, especially under transients and design base accident scenarios, are conducted on SiC/SiC composites.

Limited studies have been reported on the testing of SiC/SiC composites in lead, with more having been conducted in LBE, but similar traits have been found that indicate high material quality is key to retaining material integrity and limiting corrosion [REFERENCES (Tunison - "compatibility of SiC and SiC Composites with molten lead" - <https://www.osti.gov/servlets/purl/881300>); Takahashi and Kondo "Corrosion resistance of ceramics SiC and Si3N4 in flowing lead-bismuth eutectic" *Progress in Nuclear Energy*, **2011**, vol 53, page 1061-1065].

7.5. Recent advances in manufacture technologies for liquid metal cooled fast reactor claddings

7.5.1. Austenitic

7.5.2. F/M

KAERI have also considered manufacturing technology for FC92 F/M steel and have undertaken parametric studies including considering the effect of cold work and heat treatment (intermediate / final heat treatment) to find out the optimum range. In addition, the fabrication of FC92 cladding tube manufacturing has been initiated in collaboration with domestic steelmaking company in South Korea. This includes manufacturing on 2 tonnes of ingot through vacuum

induction melting (VIM), followed by secondary refining process (ESR). Ingot was made into mother tube having 46mm outer diameter and 3.5mm thickness by hot extrusion and subsequent tempering process. Mother tubes were made into seamless cladding tubes followed by successive cold work and intermediate heat treatment. After the intermediate tube has been finally cold drawn, it has 7.4mm outer diameter and 0.5mm thickness and is followed by a final heat treatment then characterized by normalizing and tempering process under controlled environment.

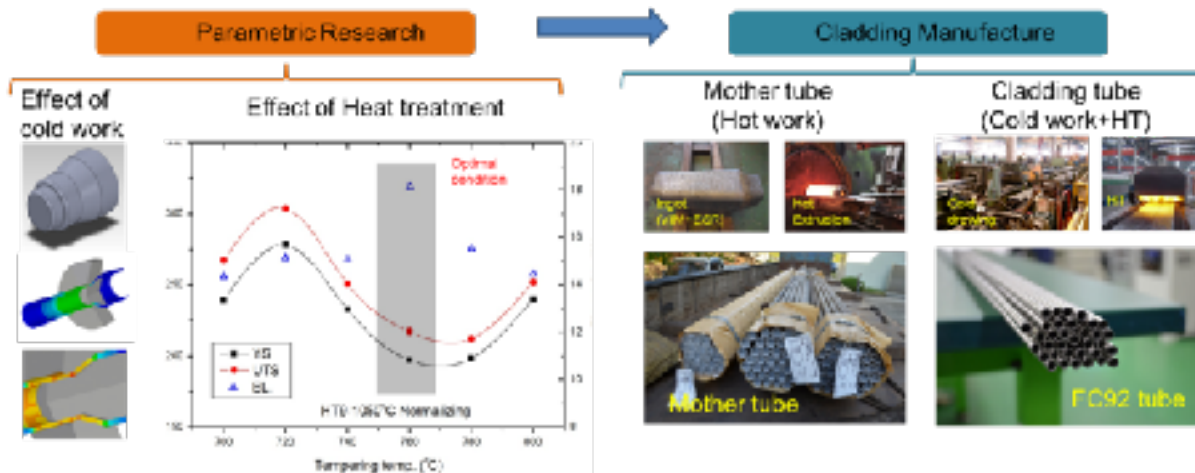


Fig. 2 Manufacture of FC92 cladding tube

Dimensional measurement (Outer/inner diameter, thickness, ovality, straightness and roughness), mechanical as well as thermo-physical properties have been evaluated over the manufactured FC92 cladding tube.

7.5.3. Advanced Claddings

1. ODS

To disperse very fine nanoscale oxide particles without them being able to dissolve into the matrix certain technologies are necessary as it is imperative the dispersion should be uniform and the bulk density sufficiently high for strengthening.

In Japan, the particularly high homogeneity necessitated the upgrading of the powder metallurgical process based on the failure caused by heterogeneity in previous irradiation tests [7.15, 7.16]. In general, the raw powders are pre-alloyed in the melting process, and a mechanical alloying process is adopted to add Y_2O_3 using an high energy attritor mill. The mechanical alloying powder was hot-extruded, and the added Y_2O_3 was re-precipitated. These improvements successfully reduced inclusions to a minimum, resulting in similar ductility (Fig 7.3) to the originally cast ferritic steel and enabling stable rolling of cladding tubes to length of approximately 2 m [7.17]. Recently, pressurized resistance welding technology has been established in Japan as the base technology of joining ODS steel cladding tube and end plugs [7.19].

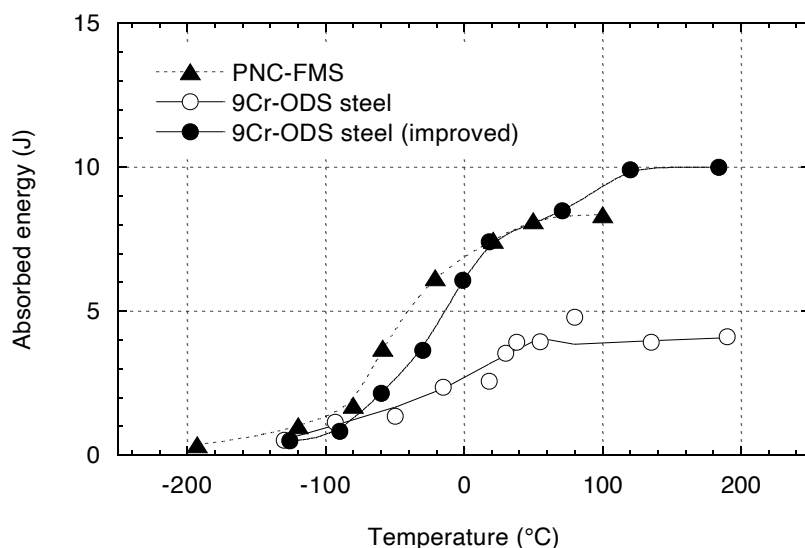


Fig. 7.3 Charpy impact properties of 9Cr-ODS steels manufactured by old and updated processes

In Korea, considerable progress on the fabrication and joining technologies of ODS steels has also been made.

2. FeCrAl, AFA and High Entropy Alloys

Manufacturing activities for HEAs for nuclear environments are currently typically limited to small batch sizes at research institutes, and in the UK this is predominantly led by academia with involvement of the wider fission and fusion environment. A similarly limited manufacturing capability exists for AFAs and FeCrAl alloys internationally. In general, challenges associated with advanced metallic cladding manufacturing are partly associated with maintaining product quality across batches and during scale-up which in someways is impacted by the varying and currently small scale demand for these “exotic” material grades.

3. Coatings

Various manufacturing methods exist for deposition of coatings onto metals, especially outside of traditional nuclear manufacturing. In the UK, research has transitioned from lab scale PVD systems to pilot scale capabilities able to coat multiple, 4 m long fuel rods.

4. Ceramics

Manufacture of high purity SiC to enable fibre and matrix manufacture is a key challenge, especially when economics is factored into fuel cladding manufacture given the difficulty and cost. As a means to reduce the cost associated with SiC/SiC manufacture, the UK has been considering both hybrid matrix production techniques as well as microwave assisted chemical vapour infiltration matrix production. Joining of SiC is also a key topic area for further consideration. In addition, to this, joining of SiC endocarps to SiC/SiC tubes is challenging and the UK has been considering laser based joining techniques as a means to ensure a hermetical seal.