Available online at www.sciencedirect.com
SCIENCE DIRECT®
Journal of Nuclear Materials 327 (2004) 140–147
journal of nuclear materials
www.elsevier.com/locate/jnucmat

Materials for high performance light water reactors

K. Ehrlich [a], J. Konys [a,*], L. Heikinheimo [b]

[a] Forschungszentrum Karlsruhe GMBH, Institut für Materialforschung II, P.O. Box 3640, 76021 Karlsruhe, Germany
[b] VTT Industrial Systems, P.O. Box 1704, 02044 VTT, Finland

Received 3 September 2003; accepted 29 January 2004

Abstract

A state-of-the-art study was performed to investigate the operational conditions for in-core and out-of-core materials in a high performance light water reactor (HPLWR) and to evaluate the potential of existing structural materials for application in fuel elements, core structures and out-of-core components. In the conventional parts of a HPLWR-plant the approved materials of supercritical fossil power plants (SCFPP) can be used for given temperatures ($\leqslant 600$ °C) and pressures (≈ 250 bar). These are either commercial ferritic/martensitic or austenitic stainless steels. Taking the conditions of existing light water reactors (LWR) into account an assessment of potential cladding materials was made, based on existing creep-rupture data, an extensive analysis of the corrosion in conventional steam power plants and available information on material behaviour under irradiation. As a major result it is shown that for an assumed maximum temperature of 650 °C not only Ni-alloys, but also austenitic stainless steels can be used as cladding materials.

1. Introduction

The major driving forces for an improvement of existing light water reactors (LWR) are a higher thermodynamic efficiency and a utilization of balance of plant (BoP) technologies of modern supercritical fossil power plants (SCFPP). A high performance light water reactor (HPLWR) study, performed recently within the 5th European Framework Program, reviewed and assessed the state-of-the-art of supercritical-water cooled reactors [1] and came to the conclusion that the once-through reactor concept developed by Dobashi et al. [2] could prove to be competitive with other advanced LWRs as well with modern fossil power plants.

For an evaluation of such a novel HPLWR-concept, the selection of appropriate materials to be used as cladding-, wrapper- and other core structural materials in the reactor core and their compatibility with out-of-core materials presently used in SCFPPs is an important issue. In Section 2 of this paper, the initial design requirements by Dobashi et al. [2] are reviewed and, where necessary, are modified, and relevant parameters for the operation of different in-core and out-of-core components are elaborated. In comparison to existing LWRs for the in-core materials a higher operational temperature, higher burn-up combined with higher radiation damage and a higher pressure are the major differences. In Section 3 a survey of materials used in SCFPP is given, combined with an analysis of their corrosion behaviour in steam environment and a review of published data on corrosion in water under sub- and supercritical conditions, including effect of stress corrosion cracking (SCC). In Section 4 possible material groups are identified, which in principle can fulfill the requirements as cladding materials in HPLWR fuel assemblies. Based on existing mechanical properties, corrosion behaviour and experience under irradiation, a first assessment of their potential is elaborated. In Section 5 recommendations for a proper selection of materials for out-of-core and in-core components are made and major areas of insufficient data base are identified.

* Corresponding author. Tel.: +49-7247 82 3720; fax: +49-7247 82 3956.
E-mail address: juergen.konys@imf.fzk.de (J. Konys).

0022-3115/$ - see front matter
doi:10.1016/j.jnucmat.2004.01.020

2. Design data and operational conditions

2.1. Reference design data

The work was started by adopting of Dobashi et al. design data [2] as reference for in-core fuel element assemblies and adding design recommendations by Bittermann et al. [3] for the reactor pressure vessel (RPV). The presently used parameters for mature supercritical conventional power plants were taken for the out-of-core components [4]. Table 1 gives a compilation of these initial 'reference design data'. In the course of this study the data for in-core application were modified by extending the maximum burnup of fuel elements from 45 to 70 GWd/t U, which corresponds to a lifetime of about 45 000 h and to allow MOX fuels for an enrichment of about 5% and more. Also, a higher pin pre-pressurization (80 bar) was adopted. Table 1 also contains coolant pressure and temperature data to be expected in the different components and requirements for the lifetime of the RPV and out-of-core components. Potential materials initially considered for application in different components are also listed, though a more detailed description of materials follows in Sections 3 and 4.

2.2. Neutronics and radiation damage parameters

The reference neutronic data in Table 1 have been re-evaluated for a modified core design and fuel assemblies by Rimpault and Testa [5]. According to their results, the total average and maximum neutron flux data can vary in the central core within 3.2 and 7.0×10^{14} n/cm^2 s. This leads to fluence levels accumulated during 45 000 h of irradiation between 5.2 and 11.3×10^{22} n/cm^2, respectively. The fraction of neutrons with an energy equal or larger than 1 MeV is in the range of 23%. However, for an assessment of the radiation damage in structure materials, the number of displacements per lattice atom (dpa) has, replaced the usual figures of neutron flux and fluence. Tanskanen's and Wasastjerna's [6] results of displacement damage rates, based on

Table 1
Initial and revised HPLWR reference design data

In core data	
Coolant pressure (bar)	25
Coolant inlet/outlet temperature (°C)	0280/508
Fuel/Enrichment	UO_2/ $\leqslant$ 5%
Fuel/Enrichment, revised	MOX/ $\geqslant$ 5%, to be determined
Burnup [GWd/t U]/lifetime (h)	45/30 000
Burnup [GWd/t U]/lifetime (h), revised	70/45 000
Neutron flux [n_{total}/cm^2 s]/fluence (n_{total}/cm^2)	$5\times10^{14}/8\times10^{22}$ (average)
	$7\times10^{14}/11.3\times10^{22}$ (revised)
Cladding outer-diameter/thickness (mm)	8/0.4
Cladding maximal surface temperature (°C)	620 for Ni-alloys
Pin pre-pressurization (bar)	$\leqslant 40$; revised: $\leqslant 80$
Potential core structure and cladding materials	
Austenitic stainless steels	1.4550, 316L(N), 1.4970
Ferritic/martensitic steels	1.4914, FV 448, EM10
Ni-based alloys	PE 16, Inconel 625, Inconel 718
Reactor pressure vessel	
Coolant pressure (bar)	250–275
Temperature (°C)	350
Lifetime (years)	60
Materials	Ferritic steels (20MnMoNi 5 5)
Out-of-core data	
Life steam pressure (bar)	250–275
Life steam/reheat temperature (°C)	540/560
Lifetime [h]	200 000
Materials	
Ferritic/martensitic steels	X20 CrMoV12 1, P 91, E 911, P 92 (NF616), P122 (HCM12A)
Austenitic stainless steels	1.4910, TP347HFG, Super304, NF709, Incoloy 800

the above neutron data and calculated with MCNP using cross-sections from IRDF-90, give values between 1.7 and 3.8×10^{-7} dpa/s for Fe. If one adopts for a first estimate these radiation damage parameters for Fe also for Ni and other major alloying elements of steels and Ni-based alloys, an average number of 27.6 dpa and a maximum of 61.8 dpa has to be expected for core structure materials after an exposure of 45 000 h.

The variation of k_{eff} in an HPLWR core with material has also been investigated by Rimpault et al. [7] and Tanskanen and Wasastjerna [6]. They give for a typical stainless steel (AISI 316) a k_{eff} value of 1.148 and for Inconel 625, a Ni-based alloy, 1.0975. Thus, the contribution of a typical Ni-alloy to the total neutron absorption is in the range of 14 and that of austenitic stainless steels nearby 10%. This has an important consequence that by use of stainless steels the necessary average enrichment to achieve a burnup of 70 000 GWd/t U can be reduced by 0.9% when compared with a Ni-alloy as core structural and cladding material.

The formation of helium via inelastic (n,α)-reactions causes high-temperature embrittlement in structural alloys. Therefore, calculations on He-generation have also been made by Tanskanen and Wasastjerna [6] for the alloys SS 1.4970 and Inconel 718 which are typical representatives for austenitic steels and Ni-alloys. In the mostly thermal neutron spectrum of a HPLWR two major reactions contribute the essential part to the generation of helium, namely the $^{10}B(n,\alpha)Li^{7}$- and the $^{58}Ni(n,\gamma)^{59}Ni(n,\alpha)^{56}Fe$-successive reactions. The first reaction dominates at the initial irradiation phase because the (n,α)-cross-section for ^{10}B is much larger than that of the Ni-double-step reaction. Since both investigated alloys contain a similar natural boron content in the range of 300–400 appm, their helium production is in the same range of 60–85 appm, caused by a complete burn-out of the ^{10}B isotope after about 3 years of irradiation.

The much higher Ni content in Inconel 718 produces via the Ni-successive reaction in this fluence range on average about 2.5×10^{-7} appm He/s compared with about 7×10^{-8} appm He/s in the alloy 1.4970, i.e. about four times more helium, but is still much lower than the contribution via boron. At higher fluence levels or a harder neutron spectrum the Ni-content would play a greater role in the production of helium.

Boron is an important alloying element in austenitic steels as well as in Ni-alloys, because it improves the high-temperature creep strength. A reduction of the He production via the B(n,α)-reaction under irradiation is possible using isotopically clean ^{11}B, which has a very low (n,α)-cross-section, compared to natural boron. The separation of both boron isotopes is technically possible and widely used for the fabrication of absorber steels. If this technique would be applied, alloys with a lower Ni-content would have an advantage.

2.3. Water chemistry

Another point that has to be considered is the water chemistry to be applied in a HPLWR. The major restrictions on the specification of water chemistry in this once-through cycle come from the necessary limitation of impurities in the feedwater for the reactor. In the HPLWR core a transition from the subcritical to the supercritical state of water occurs, which is connected with a strong decrease of impurity solubility and hence is responsible for the formation of deposits. Due to the fact that the coolant water remains in a single-phase condition along the whole once-through cycle, the control of oxygen by adding of hydrogen, like in pressurized water reactors (PWRs) or in boiling water reactors (BWRs), seems to be very likely. Based on the current information it is concluded, that the water chemistry of conventional SCFPPs and of PWRs or BWRs could be combined to define specific conditions for a HPLWR [8,9]. However, it is necessary, to investigate whether the usual way in conventional power plants to increase the pH-values of the water by adding ammonia hydrazines is compatible with the necessary specifications of the in-core water chemistry. Regarding the radiolysis in supercritical cores, no reliable data are available at the moment. Nevertheless, it is expected that the radiolytic water decomposition in a HPLWR will not exceed the normal values observed in existing BWRs and that the higher system pressure and temperature will shift the chemical equilibrium to the waterside. Hence, the addition of hydrogen to the system will reduce the radiolytic oxygen production significantly.

3. Out-of-core materials

3.1. Materials

The selection of appropriate structural materials for out-of-core components can be based on the data and experience developed for fossil fuel combustion plants operating in supercritical (SC) regimes. In the open literature, data are available on selected materials for key components in SCFPPs, e.g. for superheater tubes, hot sections and turbine materials and are summarized in Table 2 to be discussed in more detail further below. In many research projects service properties like creep-rupture and oxidation behavior are investigated. However, published data about the behaviour of these steels under supercritical water conditions are still limited, mainly due to the small number of new SC-plants in Europe.

Due to the fact that the present steels have a rather good creep strength within the envisaged HPLWR temperature range ($T_{max} \approx 600$ °C) the crucial point for material selection in future HPLWRs will be the

Table 2
Materials for key components in USC power plants [11]

Component/material	Nominal composition (wt%)
Piping materials	
X20 CrMoV 12 1	0.2C–12Cr–1Mo–0.3V
T 91/P 91	0.1C–9Cr–1Mo–V–Nb
HCM12A	0.1C–11Cr–0.5Mo–1.8W–1Cu–V–Nb–N–B
1.4910	0.03C–17Cr–13Ni–3Mo–N–B
TP347HFG	18Cr–10Ni–1Nb
Super heaters	
TP347HFG	18Cr–10Ni–1Nb
Super 304H	18Cr–9Ni–0.4Nb–Cu–N
NF709	20Cr–25Ni–1.5Mo–0.25Nb–0.05Ti–N
Esshete 1250	0.016/0.15C–14/16Cr–9/11Ni–0.8/1.2Mo-0.72/1.25Nb–N–B
HR3C	25Cr–20Ni–0.4Nb–N
Thick section boiler comp.	
NF616	0.1C–9Cr–0.5Mo–1.8W–V–Nb–N–B
HCM12A	0.1C–11Cr–0.5Mo–1.8W–1Cu–V–Nb–N–B
Turbine rotors	
COST E/F	0.12C–10Cr–1Mo/1.5Mo–1W/0W–V–Nb–N
COST B	0.17C–9.5Cr–1.5Mo–0.01B–V–Nb–N
HR1200	0.09C–11Cr–0.2Mo–2.7W–2.5Co–V–Nb–N–B

corrosion behaviour under supercritical conditions. The oxidation may be the life-limiting factor in thin wall tubes.

A literature review on potential steels suitable or used for out-of-core application in different components under supercritical and ultra-supercritical (USC) conditions was performed [10,11]. Based on this survey a listing of commercial modern SC fossil fuel power plant steels was made for out-of-core materials in Table 2. It includes the 9–12CrMoWV steels in non-stabilized or Nb-stabilized versions and the large family of austenitic stainless steels, starting with the classical 18Cr–9Ni versions and including the high Cr–Ni alloys of type Fe–20–25Cr 20–25Ni, used predominantly in superheaters. The mechanical data of these alloys are well known in the temperature range of 550–650 °C.

3.2. Oxidation and corrosion

It is assumed that in the conventional part of an HPLWR the well developed and tested materials of the commercial power plants can be adopted. For an application of materials in the HPLWR core a more detailed evaluation of their oxidation kinetics is necessary to estimate the corresponding metal loss and the oxide debris to be expected in the water cycle of the reactor core, where very thin-walled components and structures like fuel pin claddings and wrappers are exposed to supercritical water. This evaluation is based on published data from conventional steam power plants. A parabolic time dependence and an Arrhenius-type temperature dependence were assumed for the description of the oxidation behaviour in dry steam environment, and a linear time dependence was used for the material loss caused by spallation of produced oxide scales. Oxide growth and spallation constants were determined using fitting algorithms provided by a standard software for the temperature range between 550 and 650 °C. The derived formulas allow the calculation of the expected oxide and metal loss and have been applied to estimate the loss of wall thickness of thin cladding materials during the expected lifetime of 45 000 h. Fig. 1 gives the sum of oxidation and spallation effects at

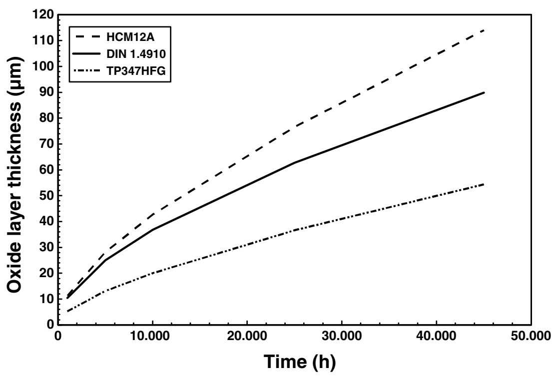

Fig. 1. A comparison of oxidation and spallation for ferritic and austenitic steels at 600 °C. Metal loss is half of the oxide thickness.

600 °C for selected ferritic/martensitic and austenitic stainless steels.

The corrosion of structural materials in supercritical water has been extensively investigated in high-oxygenated water in the frame of research for the supercritical water oxidation process (SCWO) [12]. Although the conditions in a future HPLWR are very different regarding the contents of oxygen and other impurities, some principal issues may be of general validity. For materials like steels and nickel-based alloys the principal corrosion rates as a function of pressure and temperature are shown in Fig. 2.

It is obvious, that the corrosion rate has a maximum around the critical temperature and decreases in the supercritical range. The maximum of the curve is explained as a result of (a) the increase of the corrosion rate with temperature (Arrhenius-type) and (b) the decrease of the dissociation constant and the density of water with temperature (c_{H+} becomes lower). Whether the maximum corrosion rate around the critical point of water (≈ 374 °C) is typical for very low-oxygen water or the increase of the corrosion rate far into the supercritical range as indicated by the dashed line in Fig. 2 is true, has to be validated by further investigations. For a conservative assessment of the corrosion rate, it is therefore agreed to rely on existing sub- and supercritical data.

Experience regarding the general corrosion behaviour in the running novel SCFPPs is still limited to low exposure times and data have not yet been published in the open literature. However, it is assumed that like in the subcritical steam regime, also in supercritical water, the formation of stable Cr_2O_3 layers is the dominant protective mechanism, and the thermodynamic stability of these protective layers is ensured at least up to 650 °C even at high pressure. This assumption is supported by older data on the corrosion behaviour of austenitic stainless steels in water where the system pressure range has been varied from 70 to 350 bar, and where no substantial differences in corrosion behaviour have been observed with increasing pressure [14]. A direct comparison of older data on the corrosion behaviour of ferritic and austenic steels as well as Ni-alloys in degassed supercritical water over a broad temperature range from 427 to 732 °C [15] with those mentioned above, did not lead to a general agreement, because different temperature and time dependences were found and exposure times in the older experiments were fairly short. Nevertheless, an excellent appearance of all materials was found at low temperature and when good adhering oxide films were present.

Comparing the corrosion rates in low-oxygen/chloride water from Boyd and Pray [15] with the corrosion in high oxygen/chloride water under SCWO conditions [12], a different behaviour is obvious, as shown by the maximum of the rate (full line) in Fig. 2. Further experiments have to prove, if the general or uniform corrosion of potential alloys like ferritic and austenitic steels is increasing or decreasing when changing from subcritical to supercritical water under HPLWR conditions.

3.3. Stress corrosion cracking (SCC)

A general concern is that different forms of stress corrosion cracking (SCC) could be a problem for the use of steels and especially Ni-alloys under high pressure supercritical water conditions. However, there exist some proven measures which can reduce this risk. For austenitic stainless steels which are prone to transgranular stress corrosion cracking (TGSCC) in high-oxygen/chloride containing water, it is necessary to obey the strict limitations of these species through appropriate water chemistry control in the HPLWR, which is possible as indicated in Fig. 3.

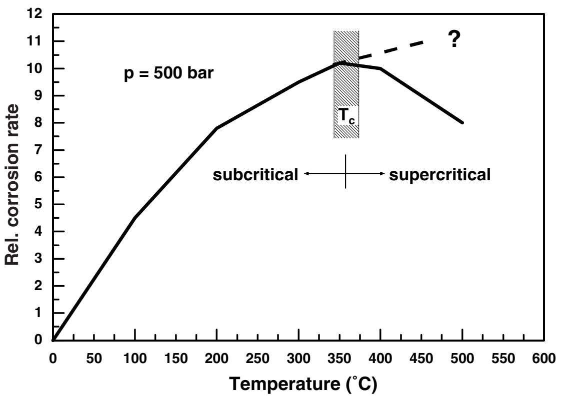

Fig. 2. Relative corrosion rate of alloys in high-oxygen/chloride water at 500 bar (oxygen is in large excess) after [13].

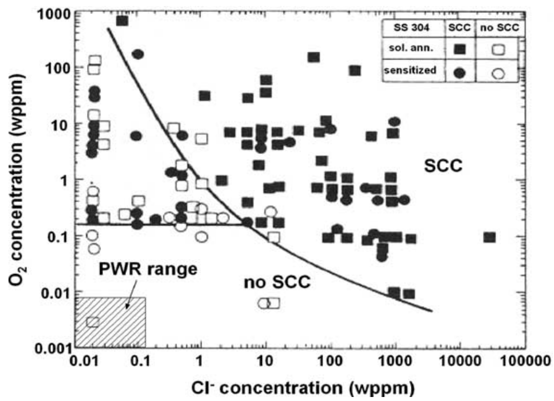

Fig. 3. The effect of O_2 and Cl^--content on the stress corrosion cracking of SS 304 in subcritical water.

Austenitic stainless steels and especially high Ni-containing alloys suffer from intergranular stress corrosion cracking (IGSCC). However, by an appropriate material composition such as an intermediate Ni content, a low carbon concentration or the use of carbon-binding or 'stabilizing' elements like Nb or Ti, the sensitivity of grain boundaries to IGSCC can be reduced. Nevertheless, one of the most uncertain areas remain the corrosion behaviour of all materials under supercritical water conditions and a possible influence on stress corrosion cracking phenomena.

4. In-core materials

4.1. Materials

A selection of available and promising materials for in-core application is based on tensile and creep-rupture data, corrosion behaviour in conventional steam power plants and on successful operation in nuclear reactors. Table 3 lists three material groups which in principle have the potential to fulfill the requirements as in-core cladding and structural materials in a future HPLWR. They have been investigated in more detail in this study.

The first group belongs to the class of well-known 9–12% CrMoV(Nb) ferritic/martensitic steels, which are presently applied in modern steam power plants. These materials are used in a tempered martensite condition and provide high creep strength and sufficient fracture toughness in the envisaged temperature window. Alloy P 91 is the variant of a modern tube and piping material, MANET II, developed for future fusion technology, is an optimized alloy with balanced creep/fracture toughness properties. Specific alloys of this group, like ferritic steel types 1.4914, EM 10 and FV 448, have shown an excellent irradiation behaviour as wrapper material of fuel elements in FBRs up to neutron fluence levels far exceeding 60 dpa [16]. A possible issue could be an irradiation-induced shift of the ductile-to-brittle transition temperature (DBTT) into the temperature range of 250–300 °C.

The second group of materials comprises of austenitic stainless steels with higher creep strength and/or improved corrosion resistance. Alloy 1.4970 is a Ti-stabilized 15Cr 15Ni steel which has been extensively tested in several chemical modifications as cladding material of fuel elements in fast breeder reactors (FBR) up to fluence levels of 150 dpa [17] and Incoloy 800 is a high Cr and Ni containing Ti-stabilized alloy with excellent corrosion behaviour, used very successfully in nuclear steam generators [18].

PE 16, Inconel 625 and Inconel 718 are typical precipitation-hardened high-Ni containing alloys. PE 16 has successfully been tested as cladding material in FBR fuel elements [17]. Ni-alloys have in general a very high creep strength and show a low general corrosion in steam environment. Dependent on the chemical composition and the metallurgical state, they can, however, be prone to stress corrosion and irradiation-induced high-temperature helium embrittlement.

Table 3
Typical candidate in-core HPLWR materials

Material	Composition (wt%)
P 91	Fe–0.1C–9Cr–1Mo–V–Nb
MANET II	Fe–0.1C–11Cr-0.5Mo–V–Nb
1.4970	Fe–0.1C–15Cr–15Ni–Mo–Ti–B
Incoloy 800	Fe–0.05C–35Ni–20Cr–Ti
PE 16	Fe–0.1C–45Ni–15Cr–Ti/Al
Inconel 625	Ni–0.1C–20Cr–10Mo–5Fe–Nb–Ti/Al
Inconel 718	Ni–0.08C–20Cr–18Fe–Nb–Mo–Ti/Al

4.2. Properties and assessment

For the design of pin claddings and wrapper tubes of fuel elements and other in-core components, tensile and creep rupture properties are of great importance. Fig. 4 gives, as an example, the ultimate tensile strength of four selected alloys in different thermomechanical treatments as a function of temperature, and in Fig. 5, the creep rupture strength for 45 000 h endurance is plotted. The advantage of Ni-alloys in comparison with ferritic/martensitic and austenitic stainless steels, based only on these conventional properties is apparent in Table 4, where a first estimate is shown of which the upper temperature can be achieved by fuel pin claddings made of different materials for two given tangential stress levels of 100 and 200 MPa, respectively. For 100 MPa an upper temperature limit of 590 °C is realistic for the strongest 9–12% Cr steel MANET II, whereas 560 °C are appropriate for the low activation alloy EURALLOY. For the two selected austenitic stainless steels 1.4970 and Incoloy 800, dependent on their chemical

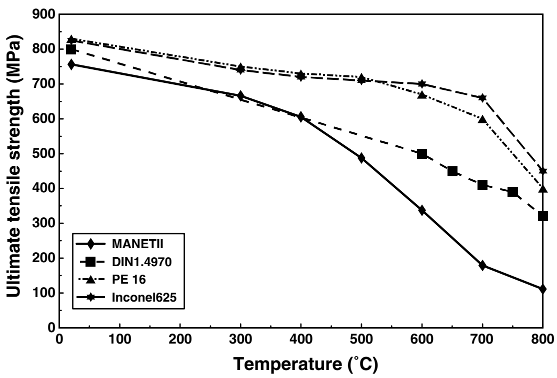

Fig. 4. Ultimate tensile strength R_M for selected alloys as a function of temperature.

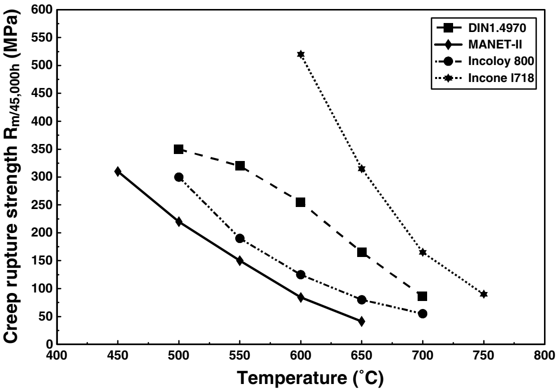

Fig. 5. Creep-rupture strength $R_{M/45\,000\,h}$ for selected alloys.

Table 4
Estimated maximum temperatures for different materials for the condition of $R_{M/45\,000\,h}$ at 100 and 200 MPa

Material	Stress (MPa)	T_{max} (°C)
MANET II	100	587
MANET II	200	512
EURALLOY	100	553
EURALLOY	200	494
1.497015	100	690
Cr–15Ni–Ti	200	629
Incoloy 800	100	625
	200	544
Inconel 718	100	712
	200	672
PE 16	100	690
	200	650

composition, an upper temperature limit ranging from 630 to 690 °C can be achieved. This upper temperature is further increased to 720 °C using a typical precipitation hardened Ni-alloy like Inconel 718. PE 16, another high Ni-containing alloy, is comparable with the high-strength austenitic stainless steels. The corresponding temperature limits at a stress level of 200 MPa are also given in Table 4 for comparison.

However, it has to be mentioned that an upper stress limit exists which is caused by an immediate buckling of thin-walled clad tubes under the high cooling pressure. This limit has been estimated under somewhat conservative assumptions, to lie for all materials in the range of 150 MPa and is mainly dependent upon the initial pressure difference and the expected hot-channel temperature under extreme operational conditions. This limit also determines the dimensional design of fuel claddings, especially the minimum wall thickness in dependence of the pin diameter. For example, an 8 mm fuel pin diameter needs a minimum cladding wall thickness of 0.45 mm for given pressure and temperature conditions.

Whereas the first comparison of achievable maximum temperatures for the different alloys in Table 4 is based on a constant maximum differential compressive stress caused by the cooling medium, a more detailed analysis of a time-dependent development of stress has shown, how conservative this estimate is. A time dependence of the exerted stress level in thin claddings is due to an increase of the inner pressure through fission gas release, which reduces the differential pressure. This can be overlapped by a possible increase of the hoop stress, caused by the reduction of the wall thickness through outer corrosion and incompatibility with the fuel material. First calculations based on reasonable assumptions on wall thickness reduction by corrosion effects and increase of pin inner pressure by fission gas release show a steady decrease of differential pressure with increasing burn-up. Therefore, the estimates of maximum achievable temperatures, where a constant initial differential pressure in Table 4 was assumed, are conservative.

This assessment on potential in-core also has taken into account irradiation effects like swelling and irradiation creep and comes to the conclusion that at a maximum fluence of 60 dpa swelling is in the range of about 1–1.5% for all three materials (1.4970, MANET II and PE 16) and can be tolerated. Less clear is to what extent the relatively high stress level of and above 100 MPa will reduce the time to rupture properties measured for the non-irradiated materials due to irradiation. Very few results, available for austenitic stainless steels and Ni-alloys, indicate that under irradiation a reduction of time to rupture has to be expected [19]. Like in the case of stress corrosion an experimental investigation of this problem is of highest priority in the next phase of exploration.

5. Summary and conclusions

The *design* data for in-core components compiled in Table 1 are very ambitious in comparison with conventional LWRs, especially with regard to the high coolant pressure ($\leqslant 250$ bar) and the increase of the water temperature from 290 °C inlet to 510 °C outlet, which causes a transition from the subcritical to the supercritical state in the core. The temperature of the claddings of fuel elements can reach more than 600 °C and the calculated neutron exposure accumulates up to 1.13×10^{23} n/cm^2 or about 60 dpa for an envisaged target of 70 GWd/t U burnup. The high neutron- and γ-irradiation associated with this burnup target of the fuel elements leads also to the formation of undesirable elements like helium via inelastic nuclear reactions in the alloys. This can lead, in combination with the displacement of atoms, to changes in the mechanical and micro-structural properties and generate dimensional distortions in the cladding and core structures. In this respect a HPLWR core resembles more the operational conditions of a fast breeder reactor than a light water reactor. Classical Zr-alloys extensively used in conventional LWRs cannot withstand such conditions, so that steels and Ni-alloys have to be considered as cladding and core structural materials.

For the *conventional or out-of-core components* like pipings, pre- and reheaters and turbines of this novel plant the operational design data summarized in Table 1 are moderate with regard to the expected working temperature ($\leqslant 600$ °C) and pressure levels (250–275 bar) of the cooling medium. They lie at the lower range of operational parameters for presently operating subcritical and supercritical fossil power plants. For these conditions, ferritic/martensitic 9–12% Cr steels like 1.4922 and P 91 or HCM12 are used today. For elevated temperatures up to a maximum of 650 °C austenitic stainless steels such as 1.4910, TP 347 HFG and others are available. For most of these alloys a broad technical data base and material properties including creep rupture properties and corrosion behaviour are available for subcritical steam conditions, whereas experience under supercritical water conditions is still restricted and the data have not yet been published.

A further increase of the operational temperature beyond 650 °C would necessitate the use of highly alloyed Ni-based superalloys with a strongly improved high temperature creep rupture strength and high oxidation resistance. The development of such high-temperature supercritical fossil power plants is, however, in a premature stage with unknown results.

Under the assumption that the conventional part of an HPLWR is operated at or below a maximum temperature/pressure of 650 °C/250 bar one can conclude that, from a standpoint of creep-rupture strength, corrosion resistance and irradiation behaviour, not only Ni-alloys, but also high-strength austenitic stainless steels can fulfill the requirements for *in-core cladding and structural materials*. Taking into account specific items like the neutron absorption, the sensitivity to irradiation-induced helium embrittlement and stress corrosion cracking the austenitic stainless steels might even be the better choice. In a long-term view dispersion-strengthened ferritic alloys should also be taken into consideration.

The assessment has also shown that the largest uncertainties in the analysis of appropriate in-core materials lie in the still unknown effect of supercritical water conditions, including water chemistry/radiolysis, on the corrosion behaviour. The combined influence of a high stress state and irradiation on stress corrosion and on the deformation mechanisms, which govern the creep-rupture and creep buckling properties is another open issue. Future R&D work should therefore concentrate on such research topics.

Acknowledgements

This project was supported by the European Commission, 5th Framework Programme of the European Atomic Energy Community ('the Community'), under contract no. FIKI-CT-2000-00033. The authors acknowledge the valuable contributions of S. Leistikow, H. Steiner, P. Arnoux and M. Schirra to this paper.

References

[1] D. Squarer, D. Bittermann, T. Schulenberg, Y. Oka, P. Dumaz, R. Kyrki-Rajamaki, N. Aksan, C. Marcazy, A. Souyri, in: ICAPP 03, Cordoba, Spain, 4–7 May 2003.
[2] K. Dobashi, Y. Oka, S. Koshizuka, in: Proceedings ICONE6, ASME, NY, 1998.
[3] D. Bittermann, D. Squarer, T. Schulenberg, Y. Oka, P. Dumaz, R. Kyrki-Rajamaki, N. Aksan, C. Marcazy, C. Souyri, in: ICAPP 03, Cordoba, Spain, 4–7 May 2003.
[4] E. Metcalf, B. Scarlin, Proc. Mater. Adv. Power Eng., Parts I–III 1 (1998) 35.
[5] G. Rimpault, E. Testa, private communication.
[6] Tanskanen, F. Wasastjerna, VTT Processes, Project Report, YE-PR-13/02, 2002.
[7] G. Rimpault, C. Maraczy, R. Kyrki-Rajamäki, Y. Oka, T. Schulenberg, in: ICAPP 03, Cordoba, Spain, 4–7 May 2003.
[8] F. Garzarolli, R. Holzer, Nucl. Energy 31 (1) (1992) 65.
[9] VGB Guidelines for boiler feedwater, boiler water and steam of steam generators with a permissible operating pressure >68 bar, VGB-R 450 Le (1995), Verlag technisch-wissenschaftlicher Schriften, Essen, Germany.
[10] J. Lecomte-Beckers, F. Schubert, P.J. Ennis (Eds.), Proceedings of the 6th Liege Conference, Materials for Advanced Power Engineering 1998, Forschungszentrum Jülich, Germany, 1998.
[11] R.U. Husemann, Werkstoffe und ihre Gebrauchseigenschaften für Überhitzer- und Zwischenüberhitzerrohre in Kraftwerken mit erhöhten Dampfparametern, Teil 1: Neue Überhitzerwerkstoffe, Betriebsversuche, Hochtemperaturkorrosionen, VGB Kraftwerkstechnik 9 (1999) 84.
[12] J. Konys, S. Fodi, J. Hausselt, H. Schmidt, V. Casal, Corrosion 55 (1999) 45.
[13] L.B. Kriksunov, D.D. MacDonald, in: Proceedings 1st International Workshop on Supercritical Water Oxidation, Jacksonville, Florida, USA, 1995.
[14] P.B. Longton, The oxidation of iron- and nickel-based alloys in supercritical steam, TRG-Report 1144C, 1966.
[15] W.K. Boyd, H.A. Pray, Corros. nace 13 (1957) 375.
[16] R.L. Klueh, D.R. Harries, ASTM 2001, 2001, p. 56 (Chapter 5).
[17] C. Brown, V. Levy, J.L. Seran, K. Ehrlich, R.J.C. Roger, H. Bergmann, in: Proceedings Int. Conf. on Fast Reactors and Related Fuel Cycles, Kyoto, Japan, vol. 1, 1991, p. 7.5.
[18] E. Tenckhoff, M. Erve, Atomwirtschaft 4 (1992) 185.
[19] C. Wassilew, K. Ehrlich, H.-J. Bergmann, ASTM-STP 956 (1987) 30.