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Building on knowledge base of sodium cooled fast spectrum reactors to develop materials technology for fusion reactors

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ABSTRACT

The alloys 316L(N) and Mod. 9Cr–1Mo steel are the major structural materials for fabrication of structural components in sodium cooled fast reactors (SFRs). Various factors influencing the mechanical behaviour of these alloys and different modes of deformation and failure in SFR systems, their analysis and the simulated tests performed on components for assessment of structural integrity and the applicability of RCC-MR code for the design and validation of components are highlighted. The procedures followed for optimal design of die and punch for the near net shape forming of petals of main vessel of 500 MWe prototype fast breeder reactor (PFBR); the safe temperature and strain rate domains established using dynamic materials model for forming of 316L(N) and 9Cr–1Mo steels components by various industrial processes are illustrated. Weldability problems associated with 316L(N) and Mo. 9Cr–1Mo are briefly discussed. The utilization of artificial neural network models for prediction of creep rupture life and delta-ferrite in austenitic stainless steel welds is described. The usage of non-destructive examination techniques in characterization of deformation, fracture and various microstructural features in SFR materials is briefly discussed. Most of the experience gained on SFR systems could be utilized in developing science and technology for fusion reactors. Summary of the current status of knowledge on various aspects of fission and fusion systems with emphasis on cross fertilization of research is presented.

1. Introduction

The structural components of Generation-IV reactor systems will undergo exposure to high temperatures (Table 1), high-neutron doses and corrosive environment. The service environments in Gen-IV systems pose significant challenges to materials selection, qualification and developmental efforts. The structural materials that are being considered for use in Gen-IV systems are listed in Table 2. In view of the commonality of operating temperature conditions in SFRs and fusion reactor systems, synergism exists for cross-cutting research on ferritic–martensitic steels, austenitic stainless steels and oxide dispersion strengthened (ODS) alloys.

For the range of service conditions in Gen-IV systems, including possible accident scenarios, the proposed materials must meet design objectives in the areas of (i) dimensional stability including void swelling, thermal creep, irradiation creep, stress relaxation and growth, (ii) strength, ductility and toughness, (iii) resistance to fatigue cracking and helium embrittlement, (iv) neutronic properties for core internals, (v) physical and chemical compatibility with the coolant, (vi) thermal properties during anticipated and off-normal operations, and (vii) interactions with other materials in the systems. Significant opportunities exist for sharing of information on the technology of irradiation testing, specimen miniaturization, advanced methods of property measurement and development of materials property databases to ensure consistency and to facilitate the exchange of data among fission and fusion programmes.

2. Mechanical properties of 316L(N) and Mod. 9Cr–1Mo steels

2.1. Creep rupture properties

Material properties can vary substantially with metallurgical conditions. Heat-to-heat variations in chemical composition translate into variability of component performance that adds an uncertainty to the ability of plant operators to predict materials performance. Significant heat-to-heat variations in the creep rupture life of 316 SS have been observed (Fig. 1) due to subtle differences in grain size and amounts of minor elements such as carbon, boron and nitrogen in different heats. In general, nitrogen alloyed low carbon austenitic stainless steel [3] 316L(N) displayed better rupture life (Fig. 2) and lower creep rate compared to 316 SS as a result of solid solution strengthening by nitrogen and precipitation strengthening by fine carbides. 316L(N) SS with stringent composition control displayed much higher creep strength than the minimum stress to rupture values specified in RCC-MR design code [4]. Heat-to-heat variations in stress rupture properties of 316

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Table 1
Different Gen-IV nuclear reactor systems [1].

Reactor system	Coolant	Neutron spectrum	Core outlet temp (°C)
Gas cooled fast reactor (GFR)	Gas (e.g. He)	Fast	~850
Lead-cooled reactor (LFR)	Liquid metal (e.g. Pb, Pb–Bi)	Fast	550–800
Molten salt reactor	Molten salt (fluoride salts)	Thermal	700–800
Liquid metal cooled fast breeder reactor (LMFBR)	Liquid metal (Na)	Fast	~550
Very high-temperature reactor (VHTR)	Gas (e.g. He)	Thermal	>900
Super critical water cooled reactor (SCWR)	Water	Thermal/fast	350–620

Table 2
Summary of various candidate structural materials for Gen-IV reactors [2].

Reactor system	F-M steel	ASS	ODS steel	Ni-base alloys	Graphite	Refractory alloys	Ceramics
GFR	P	P	P	P	–	P	P
Pb-LFR	P	P	S	–	–	S	S
MSR	–	–	–	P	P	S	S
LMFBR	P	P	P	–	–	–	–
SCWR	P	P	S	S	–	–	–
VHTR	S	–	–	P	P	S	P

P – Primary choice; S – secondary choice.

and 316L(N) steels have been rationalized by using a pair of constants to index each heat [5]. This heat indexing method made it possible to convert a vast pool of stress rupture data on these steels (590 data sets on 36 heats comprising tubes, plates and bars) into a knowledge base in the form of isothermal reference fits at eight temperature levels in the range 773–1123 K. A procedure has been developed for stress rupture life extrapolation of 316L(N) weld metal, using the data on base metal [5].

Higher creep–rupture strength could be obtained in Mod. 9Cr–1Mo steel in rolled, forged and tube product forms than the average strength values reported in RCC-MR design code by employing electroslag refining and exercising strict control of composition over a very narrow specified range [6]. A strict control of the tempering temperature in the heat treatment furnace is essential to avoid the local soft zones that produce lower creep strength in the longitudinal direction of seamless tubes. In order to design the steam generators of 500 MWe prototype fast breeder reactor (PFBR) for 40 years of lifetime with a load factor of 75%, detailed

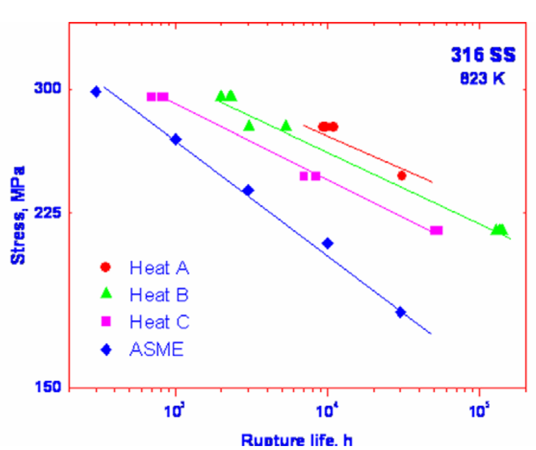

Fig. 1. Creep rupture strengths of various heats of type 316 SS.

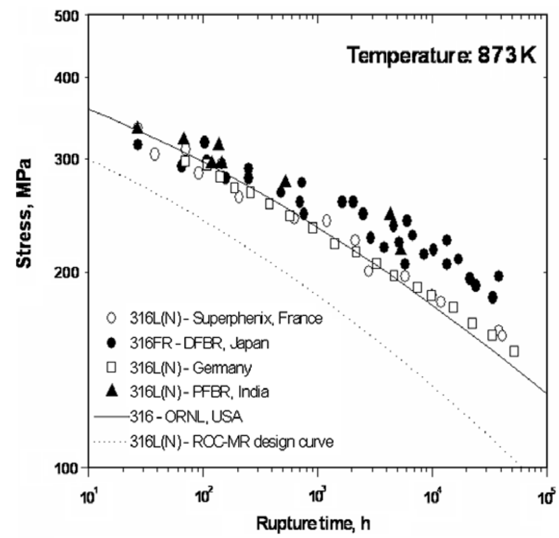

Fig. 2. Effect of nitrogen on the creep rupture strength.

investigations have been performed to obtain creep allowable stresses for Mod. 9Cr–1Mo steel at a life of 2,63,000 h. Allowable stress values predicted by Larson–Miller parametric approach (LMP) and multilayer perception artificial neural network (ANN) lie very close to those obtained by extrapolating RCC-MR (French Code) data at a creep rupture life value of 2,63,000 h [6]. Similar approaches could be adopted for predicting rupture life of reduced activation ferritic–martensitic steels, where data is currently limited.

Creep rupture lives of post weld heat treated Mod. 9Cr–1Mo weld joints were found to be significantly inferior compared to the base metal (Fig. 3) due to preferential accumulation of creep deformation coupled with extensive creep cavitation in the intercritical region of heat-affected zone (HAZ). This type of failure known as Type IV cracking [7] and has also been reported in HAZ of RAFM steel F82H after prolonged post weld heat treatment at 993 K [8]. Type IV cracking is a critical issue for RAFM steels as the localization of deformation in the soft region of HAZ would likely be enhanced after irradiation in fusion reactor systems.

2.2. Time dependent damage modes in high-temperature fatigue

Service conditions experienced by components in SFRs would also involve cyclic loading during start-up and shut down or during power transients. The combination of cyclic loading and steady state operation of SFRs at elevated temperatures introduces several time and temperature dependent damage processes that can have a detrimental effect on the performance of components. The low cycle fatigue (LCF) lives of 316L(N) SS and Mod. 9Cr–1Mo steel were shown to be significantly affected by dynamic strain ageing (DSA), inelastic deformation, deformation ratcheting, slip character, mean stress, oxidation, creep damage and phase instabilities in the temperature range of their operation [9,10]. These time dependent mechanisms have been found to act either independently or synergistically depending on the test conditions and led to premature fatigue failures. DSA has been noticed to be particularly harmful in 316L(N) at moderately elevated temperatures where creep and oxidation damage effects were not significant. The negative temperature dependence and strain rate sensitivity of half-life cyclic stress are established as potent indicators of DSA effects in fatigue [10]. The out-of-phase TMF tests on 316L(N) SS were established to be more deleterious in the sub-creep range (DSA) while in-phase tests were found to be more

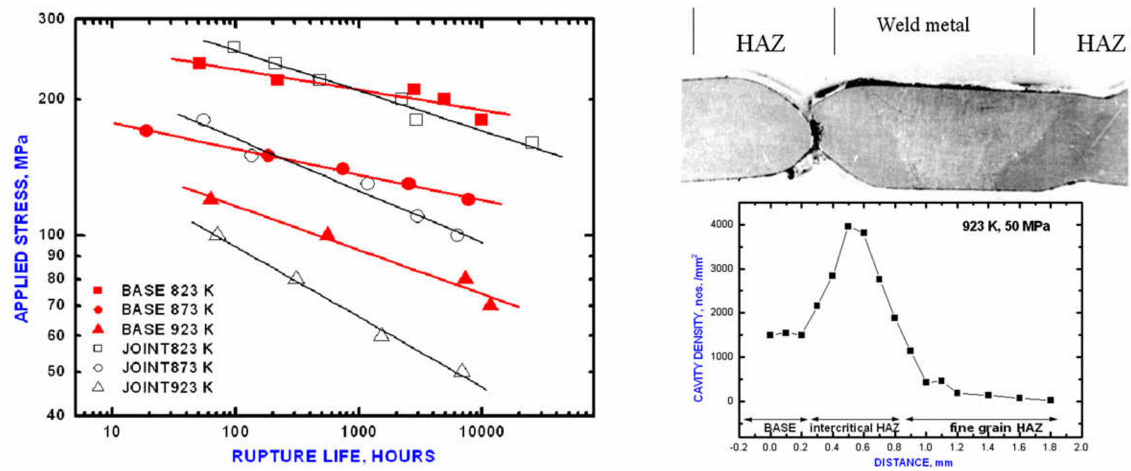

Fig. 3. (a) Variation of rupture life of modified 9Cr–1Mo-base metal and weld joint with applied stress and (b) Creep–failure location of the weld joint and creep cavity density with distance of modified 9Cr–1Mo similar weld joint failure location of the weld joint at 823 K, 120 MPa.

detrimental in the creep range [11] (Fig. 4). It may be noted that the operating temperature range of 316(L)N in fusion reactors lie where DSA is the predominant damage process. Furthermore, under relevant operating temperature of 316L(N) in ITER, the alloy exhibits pronounced cyclic softening and lower response stresses that are undesirable.

Strain controlled fatigue of Mod. 9Cr–1Mo promoted initial hardening followed by cyclic softening in the range 300–873 K. Similar type of cyclic stress response has been observed in RAFM steels [12]. For both the steels, in creep–fatigue interaction tests, compression holds were found more damaging than tension holds. The compressive dwell sensitivity in Mod. 9Cr–1Mo resulted from the combined effects of increased amount of inelastic strain during hold in a given cycle and the detrimental effects of oxidation on enhancing crack propagation. It has been shown that creep and fatigue damage summation value in a damage summation rule is different for 316L(N), 2.25Cr–1Mo, 9Cr–1Mo and Mod. 9Cr–1Mo. The creep–fatigue interaction data is of relevance to low activation ferritic/martensitic steels being considered for fusion systems. Blanket modules of fusion reactors are subjected during service to alternating thermal and mechanical stresses as a consequence of the pulsed reactor operation. Therefore thermo-mechanical fatigue (TMF) testing is more appropriate. TMF experiments at a maximum temperature of 823 K on EUROFER 97 showed a drastic reduction in life compared to isothermal LCF tests [13].

2.3. Generation of fatigue design curves for 9Cr steels

Fatigue properties depend upon the material processing route. In general, thick section forged components show lower life as compared to hot rolled thin section products. Using the fatigue data generated on small specimens and employing the philosophy embodied in the ASME code case N-47, fatigue design curves have been developed for thick section 9Cr–1Mo steel [14]. The extrapolated fatigue curves were used to construct preliminary fatigue design curves by applying a reduction factor of 20 on the number of reversals or a factor of two on the strain amplitude. These factors cover effects such as environment, size, surface finish and scatter of the data. Similar approach is needed for generating fatigue design curves for RAFM steels till the design codes get established.

3. Optimization of die profile and assembly sequence of petals for PFBR main vessel

The fabrication of large size pressure vessels of SFRs necessitates pressing, assembling and welding of petals to form the final shape within the stringent dimensional tolerances. In order to predict springback in plate bending, a three-dimensional FEM based process models have been developed for fabrication of petals [15]. Based on simulations from the two-dimensional FEM modeling of the plate forming process, the optimum FEM formulation

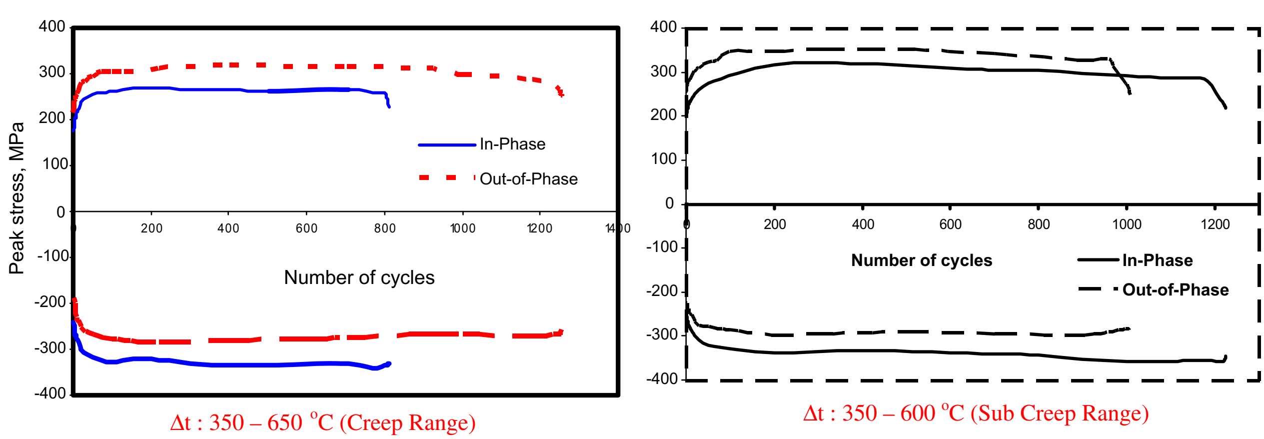

Fig. 4. Comparison of cyclic stress responses during in-phase and out-of-phase TMF cycling.

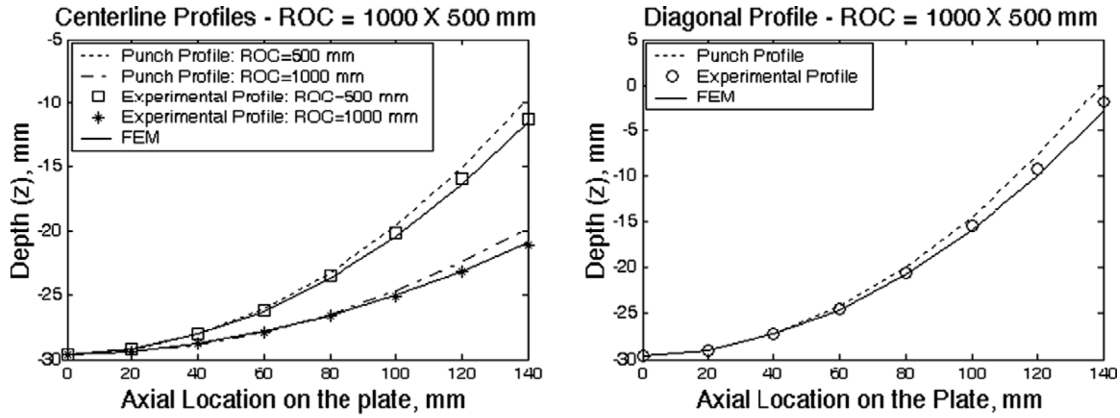

Fig. 5. FEM simulation of centerline and diagonal profiles of the plate before and after springback (experimental data points are also included).

was identified as sequentially using a dynamic explicit formulation with contact modeling for the pressing phase and a static implicit approach for springback calculation in the Lagrangian framework. The details pertaining to double curvature bending by 3-D FEM modeling and experiments on $300 \times 300 \times 15$ mm plates are shown in Fig. 5. The differential distribution of strains that occurred during plate forming led to differential springback in the plate with a petal shape deviating from die and punch shapes. The FEM based approach is indispensable in fabrication of vessel components of ITER and DEMO fusion reactors.

4. Metal forming maps

In order to achieve the required service properties in PFBR components the microstructural development during hot working has been carefully controlled to avoid defects and flow instabilities. This necessitated determination of flow curves under compression over a wide temperature and strain rate regimes and identification of safe and unsafe domains based on dynamic materials model. The processing maps and instability maps have been generated for 304, 316, 304LN, 316LN, 9Cr–1Mo and Mod. 9Cr–1Mo steels [16–18]. Mechanical processing of these alloys in the dynamic recrystallisation regime (Fig. 6) with appropriate selection of processing parameters resulted in high-quality rolled and forged products with significant reduction in the rejection rates. This type of scientific approach is essential for processing defect-free components of fusion reactors employing RAFM steels.

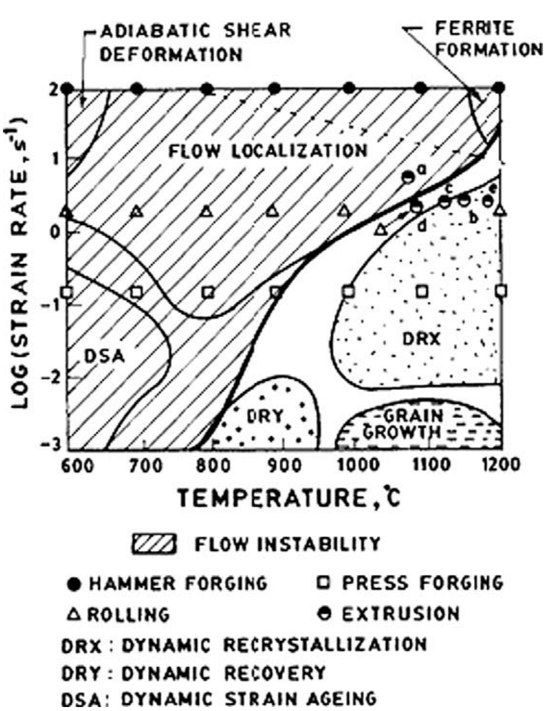

Fig. 6. Processing map of 316L(N).

5. Oxide dispersion strengthened alloys

In order to have extended life of fuel cladding tubes of SFRs, materials that exhibit superior resistance to high-neutron exposure beyond 200 dpa as well as high-creep strength at 973 K are required. Oxide dispersion strengthened (ODS) ferritic/martensitic steels which have more swelling resistance and creep strength than Alloy D9 are currently under development by controlling the composition, number density and stability of Y–Ti–O nanoclusters [19]. Detailed investigations conducted on the microstructural aspects of 9Cr–ODS alloy clad tubes are of significant interest towards the development of optimum composition for the blanket system plates of the fusion reactors. Low fracture toughness, high-ductile-to-brittle transition temperature (DBTT) [20] and finding suitable welding process (fusion welding is undesirable) for ODS alloys continue to be cause of concern. In case of mass production of clad tubes and plates, the initial billets are generally produced by hot isostatic pressing (HIP) of mechanically alloyed powders. In HIP processing the impurity particles present on powder particles get embodied on grain boundaries as agglomerated particles, leading to accelerated creep cavity formation and associated reduction in life. Though the creep resistance of ODS alloys could be improved marginally by choosing coarse particles initially, there could be other modifications needed in powder production and processing conditions to eliminate PPBs.

6. Weldability of stainless steels and ferritic steels

Solidification cracking is a significant problem during welding of fully austenitic stainless steels and their stabilized compositions. Hot cracking is caused by low melting eutectics containing impurities such as S, P and alloying elements such as Ti, Nb and N. Weld metal and heat-affected zone (HAZ) cracking behaviour of Alloy D9 and 316L(N) SS, have been investigated specifically for understanding the roles of Ti in Alloy D9, N in 316L(N) SS, and the impurity elements [21]. In Alloy D9, cracking increased with increase in

Ti/C ratio, but significant contribution to cracking resulted from the ~200 ppm nitrogen picked up during welding. In modified 316 weld metals with ferrite numbers of 3–7, nitrogen in the range 0.06–0.12% had no detrimental effect on weldability. The metallurgical propensity to solidification cracking is determined by elemental segregation, which manifests itself as a brittleness temperature range (BTR) that can be determined using varestraint test. Excellent correlation was obtained between chemical composition (chromium equivalent to nickel equivalent, Cr_{eq}/Ni_{eq} ratio) and BTR for various austenitic SS of interest (Fig. 7) [21]. A *generalised Bayesian Neural Network (BNN) model for estimating ferrite content* in austenitic and duplex SS welds from their chemical composition [22] and robust ANN models have been developed for alloy development, thermo-mechanical processing, creep–fatigue life prediction and prediction of radiation damage.

6.1. Activated-flux TIG (A-TIG) welding of austenitic stainless steels

TIG welding process has been developed in which a thin coating of activated flux is applied to the base metal surface prior to welding [23]. The activated flux causes constriction in the welding arc, thereby increasing the current density at the anode root and the arc force acting on the weld pool. Activated fluxes have led to a drastic improvement in the operational characteristics of the TIG process, notably greater depths of penetration in a single pass without addition of any filler metal, higher welding speeds and reduction in the sensitivity to cast-to-cast variation of the base metal. A-TIG welding process could be considered for fusion reactor thick components such as vacuum vessel.

6.2. Dissimilar metal joining

In SFRs dissimilar metal welds (DMWs) between austenitic SS and Cr–Mo ferritic steels, used in steam generators, are prone to extensive premature service failures due to large thermal stresses induced at the weld fusion line as a result of difference in coefficients of thermal expansion (CTE). A unique trimetallic transition joint between austenitic SS and Cr–Mo steel, with an intermediate Alloy 800 piece has been designed, developed and characterized [24]. Concepts used in predicting long-time creep behaviour from short-term creep tests were extended to thermal fatigue situations in a novel thermal cycling performance test that simulates the life-limiting damage mode of these DMWs. The trimetallic transition joint, 316LN SS/Alloy 800/Mod. 9Cr–1Mo steel adopted for the steam generator circuit of SFRs has shown a factor of four improvements in life under thermal cycling conditions. Since the usage of Ni-base alloys are restricted in fusion reactor systems, significant efforts are needed to develop a suitable DMW joints.

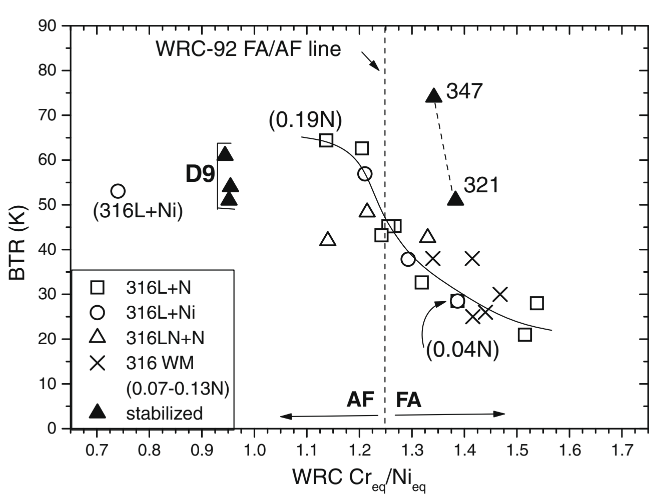

Fig. 7. Weldability of austenitic SS-base and weld metals as function of composition (WRC Cr_{eq}/Ni_{eq} ratio), valid for a narrow range of <0.04% P + S contents.

7. Miniature specimen testing

In fission and fusion programmes, there is an urgent need for the rapid measurement of properties from small volume of material for assistance in the design and development of new alloy systems, assessment of mechanical properties of small heat-affected zones of welded components, to make effective use of the limited volume available in the test reactors and particle accelerators for irradiation of test samples, and for testing of active sections of material where the activity levels of small specimens will allow economic handling of test material within limited protection areas and in the fields of failure analysis and remaining life assessment. Considerable research and development work has been carried in out on SFR materials for establishing miniature specimen test techniques, sample design and establishing correlations between mechanical properties of small specimens and conventional test specimens [25]. A linear correlation between conventional tensile and shear punch tests has been demonstrated for a wide variety of materials such as 304 SS, 316 SS, 2.25Cr–1Mo and 9Cr–1Mo ferritic steels etc. Using ball-indentation tests, the degradation in mechanical properties of Mod. 9Cr–1Mo steel on thermal and creep exposures have been studied [26]. Creep tests were carried out on Mod. 9Cr–1Mo steel at 923 K under different stress levels and interrupted at various strains corresponding to different life fractions. Small specimens extracted from the head and the gage portions of the creep exposed samples were used to estimate mechanical property changes of the thermal and creep exposures, respectively, using ball-indentation techniques. The changes in the strength property were related to time, temperature and applied stress through Larson–Miller (LM) type parametric relationships.

8. High-temperature design approach for SFR components and design codes

The Code RCC-MR developed by CEA provides design rule, material property data and analysis guidelines for the possible failure modes relevant to SFRs. In the structural design of 500 MWe PFBR, all the possible failure modes based on structural integrity considerations and key parameters influencing the failure mechanisms are identified and understood. The failure mechanisms are caused by high-temperature operation over prolonged period (>40 years) with thermal transients, vibration, seismic and events of very low probability resulting in accident conditions. The failure modes have been analysed systematically using numerical and experimental techniques in compliance with the applicable design codes, such as RCC-MR (2002) [27,28] and ASME-Sec III (2003) [29]. In the components, the allowable stress (S_m) is 2/3 of yield strength at the specified temperature to prevent gross deformation. The stresses in the component subjecting to design loads have been determined by finite element analysis. In PFBR both hot and cold sodium pools co-exist within the main vessel with a large temperature difference (150 °C) which is the source of high-temperature gradient during steady as well as transient conditions. In view of low design pressure, high-thermal stress and further economic considerations, thin walled shell structures are generally chosen for main vessel, inner vessel and thermal baffles. Buckling, particularly under seismic-induced loads, is the critical failure mode for the thin walled structures. Sodium has excellent heat transfer properties and poses many challenging structural mechanics problems, originating from thermal stress and thermal shocks. Under higher operating temperature over a prolonged period (>40 years) in association with temperature variations following thermal transients, the life-limiting thermo-mechanical related failure modes such as creep deformation, creep rupture, fatigue, creep–fatigue interaction, ratchetting and progressive buckling are analysed specifically for the components made of austenitic steel that has low thermal conductivity and high co-efficient of thermal expansion. Some of the special types of failure modes considered include thermal striping and thermal ratchetting that occur in components subjected to temperature-induced fluctuations associated with thermal hydraulic phenomena. All the possible failure modes are identified systematically reflecting the operating experiences accumulated over 300 reactor years worldwide. Fig. 8 shows the failure modes considered in the design of reactor assembly components.

The combination of steady stresses sustained during normal operation (responsible for creep damage) in association with high-cyclic stresses developed during reactor scram (responsible for fatigue damage) results in creep–fatigue damage interaction at the stay plate-outer shell junctions of control plug (A), stand pipe-conical shell (B) and top tube sheet-outer shell junction of IHX (C). Since, about 860 shutdowns are considered conservatively in the design, the same number of load cycles, each with the hold period of ∼300 h (to account for the 40 years design life with 75% load factor) were applied for creep–fatigue damage evaluation. At some specific locations in the hot pool components, especially on CP parts, the accumulated creep–fatigue damage (CFD) is enhanced due to the presence of high-cycle thermal fatigue caused by thermal striping mechanism. The CFD is further enhanced due to presence of welds at the junctions. By limiting the CFD to a value, specified by the design code, the crack initiation is prevented at the junction.

8.1. Creep–fatigue damage assessment as per RCC-MR

For the main vessel and inner vessel the stresses and strains are computed using viscoplastic analysis, since it is not possible to respect the creep–fatigue damage limits through elastic analysis route for these components without any sympathetic safety actions. However, sympathetic safety actions are introduced following the reactor scrams that introduce thermal shocks. Through this it is possible to analyse the creep–fatigue damage by using elastic route. For the viscoplastic analysis, the Chaboche model was employed [30]. For other components viz. control plug, IHX and SG shell-skirt support, elastic analysis route was followed. The summary of computed creep–fatigue damage assessment of hot pool components is presented in Table 3. These results indicate that the maximum creep–fatigue damage estimated based on design load cycles (D_{eff}) is less than 50% of code allowable value (unity). Hence structural mechanics considerations permit a design life of 45 years with a comfortable margin. This analysis has formed a basis for the selection of higher operating temperature (820 K) and longer plant life (>40 years) for PFBR.

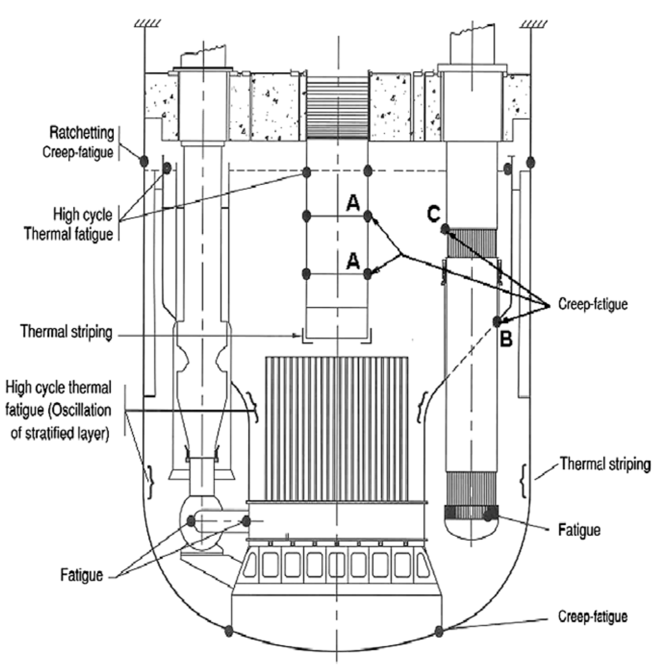

Fig. 8. High-temperature failure modes in prototype fast breeder reactor.

In order to predict the life of steam generator accurately under combined mechanical and cyclic thermal loadings. '20 parameter Chaboche Viscoplastic model' has been developed for Mod. 9Cr–1Mo steel. This model can simulate the complex mechanical behaviour, viz. softening/hardening under monotonic loadings, insignificant primary creep, cyclic softening behaviour, strong strain rate dependency of mechanical properties, over the long range of temperatures. This can be directly applied to life prediction of fusion reactor blanket system components.

8.2. Determination of creep rupture strength of steam generator tubes

Steam Generator tubes of Mod. 9Cr1Mo are welded to the spigots of the tube sheet by in-bore welding technique. In the absence of experimental data, a strength reduction factor of 0.9 on creep rupture strength of the weld is recommended as an 'ad-hoc' measure for the European fast reactor (EFR). A systematic test programme on SG tube mock-ups has been undertaken to quantify the reduction in creep life of welded tubes compared to the parent tubes with and without considering the thickness reduction due to corrosion and under-tolerance. A complex test setup has been set up to maintain high-internal pressure (20 MPa) in the tube specimen at high-temperature and accelerated creep tests were performed deriving the test conditions based on Larson–Miller Parameter. Creep tests conducted on tubes without any weld under internal pressure of 20 MPa at 950 K revealed a hoop strain of 1.5% compared to the computed creep strain of 2.1% estimated using RCC-MR: Appendix Z. This phase of the test programme confirms the use of RCC-MR data for design analysis [31]. High-temperature tests conducted for SG (G91) and IHX tubes (SS 316 LN) are useful in providing robust weld design rules for the coolant channel in the first wall of the fusion reactor.

8.3. Design codes and environmental effects

Currently, the construction of mechanical components of PFBR is being done as per RCC-MR code. PFBR construction provided many useful feed backs. Assessment of conservatism and suggestions of improvements were made based on systematically planned experiments executed through in-house and collaborative works, particularly in the domain of weld design, creep crack growth, thermal buckling and ratcheting assessments. These improvements are planned jointly with CEA, who developed this code. RCC-MR is judged to be suitable code for the design and construction of most of the fusion reactor components.

Design codes do not provide guidelines for the treatment of environmental effects, i.e., chemical interactions between structural materials and coolants, effects of neutron displacement damage and transmutant gases. In addition to temperature and dose, the radiation-induced phenomena are sensitive to dose rate, transmutant gas generation rate and operating temperature history. In some instances there could be a strong interaction between corrosion phenomena and radiation effects. Meeting the performance

Table 3
Creep–fatigue damage for PFBR components [33].

Component	Analysis	Load cycle/a	Hold time/cycle (h)	D_c	D_f	D_{eff}
Main vessel	Viscoplastic	4 SGDHR cycle	024	0.02	0.007	0.036
Inner vessel	Viscoplastic	19 scrams	350	0.05	0.015	0.086
Control plug	Elastic	19 scrams	350	0.36	0.005	0.370
IHX	Elastic	22 shutdowns	305	0.45	0.003	0.460
SG	Elastic	22 shutdowns	305	0.30	0.030	0.310

goals of fusion and fission systems would require long-term efforts to expand understanding of these phenomena and to develop sound scientific basis for development of improved properties in existing alloys through compositional and microstructural modification and development of entirely new alloys specifically designed to resist environmentally related property degradation in its many forms.

9. Non-destructive evaluation

Non-destructive evaluation (NDE) has been extensively employed for characterizing various key microstructural features, mechanical properties, deformation and damage mechanisms as a primary step towards ensuring structural integrity of SFR components. Ultrasonic parameters (such as attenuation and velocity) and micro-magnetic parameters (coercive force and Magnetic Barkhausen Emission, MBN) have been developed for determination of grain size and size distribution of second phases that occur during thermal ageing. Creep and fatigue damage has been characterized by in situ metallography, acoustic emission (AE), ultrasonic attenuation and velocity measurements, acoustic harmonic measurement, MBN, acoustic Barkhausen emission, laser interferometry, positron annihilation, X-ray diffraction and small angle neutron scattering. The correlation between progressive deformation and damage during LCF of 9Cr–1Mo and MBN peak height are illustrated in Fig. 9 [32]. The cyclic hardening which occurred in the early stage of LCF decreased the MBE peak height value. The reduction in MBE peak height value increased with increase in the amount of hardening, which in turn depended on strain amplitude. The progressive cyclic softening stage displayed a reversal in MBE response. In the saturation stage, where the stress value remained constant for a large number of cycles due to formation of stable dislocation substructure, resulted in a constant value of MBE. The rapid stress drop and cusp formation in stress–strain hysteresis loop, which indicates surface crack initiation and propagation, exhibited a rapid increase in MBN peak value. This is attributed to the movement of additional reverse domains produced at crack surface. The correlation of MBE with dislocation substructure evolution in different stages of LCF has also been established. These studies demonstrated that the potential of MBN technique in monitoring fatigue damage and prediction of the remaining service life. Ultrasonic spectral parameters have been used for evaluation of grain size in a wide range of microstructures, i.e. ferrite, ferrite + martensite and martensite in stainless steel 316 and Mod. 9Cr–1Mo steel [33].

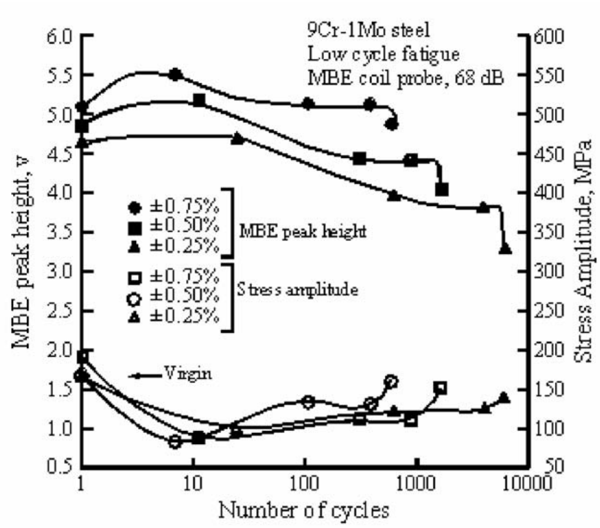

Fig. 9. Variation in MBE peak height and stress amplitude in low cycle fatigue tests at different strain amplitudes in 9Cr–1Mo steel.

The AE technique has been extensively used to study the deformation mechanisms, and to determine LCF and HCF damage and fatigue crack propagation rate, from which severity of flaws can be estimated. Different stages of crack growth were characterized by AE technique [34]. During NDE-based materials characterization, the interaction of the probing medium with various microstructural features may lead to complex signals, thus making interpretations very difficult and requiring separation of the signal to obtain feature specific information. Under such conditions, the use of multiple NDE techniques and NDE parameters in a complimentary manner helps in isolating the influence of one or a combination of specific microstructure features. A comprehensive review has been written recently on several important aspects of NDE [35]. As the interrogating medium is different in each NDE technique, the sensitivity, speed and detection capability of techniques are different. Choice of NDE technique depends on material, geometry, defect type and location, applicability, accessibility and suitability [35].

10. Issues in the development of fusion reactor materials and technologies

Development of plasma facing and breeding blanket materials which are capable of withstanding high-neutron and heat fluxes (30–75 dpa/year and 10–15 MW year/m^2 for two to five years) and appropriate fabrication methods is crucial for successful realisation of fusion power. Structural materials of current interest for first wall and blanket systems are RAFM steels, ODS alloys, V-base alloys and SiC/SiC$_f$ composites [36]. The most promising structural material appears to be RAFM steel with greatest maturity in technology for applications in the range 623–823 K. Though considerable improvements have been made in the specification and fabrication of RAFM steels, immediate actions are necessary for the optimization of fabrication with respect to the homogeneity of thick product forms and composition adjustment for reduction of the irradiation-induced activation of large scale heats.

In spite of considerable research work on TIG, EB, Laser, plasma-MIG, friction stir welding and diffusion welding for joining RAFM steels, optimization of a welding technique that could be used for majority of the components has not yet been established. TIG has very high potential in this regard. The optimization of TIG weld wire compositions that are resistant to hot cracking, evaluation of

Table 4
Comparative evaluation of the current status of knowledge in fission and fusion systems.

	Fission (SFRs)	Fusion
Mechanical property data base	Fairly well developed	Considerable gaps (C, F, CFI and fracture properties)
Radiation effects	Void swelling	Radiation-induced segregation and its implications on phase instability
	Helium embrittlement	Effects of relevant amounts of helium coupled with neutron dose on various phenomena and mechanical properties
	Irradiation creep (extensively examined)	Radiation-induced precipitation and defects, flow localization etc.
Mass transfer and corrosion	Well established in liquid Na for core and steam generator materials	Several TBM concepts and coolants Effects of T, flow velocity, microstructure etc. on mass transfer and corrosion in various coolant media to be established Critical issues to be solved dissolution and mass transfer is greater in lead than Li oxygen forms protective layers Li is a reducing agent oxide layers for coating are not permitted
Models	Predictive models for swelling are inadequate	Predictive models for swelling and segregation are inadequate
High-temperature design	ASME, RCC-MR (conservative design criteria)	ITER structure design criteria (ISDC) Development of codes for high temperatures (DEMO, etc.)
	Bilinear damage rule for creep–fatigue interaction (materials specific)	Incorporation of environment, irradiation and creep–fatigue interaction effects Creep properties during irradiation (beneficial or detrimental: in situ testing techniques
Industrial production of materials	Well established (stainless steels, carbon steels, ferritic steels, superalloys)	Little industrial production experience (RAFM steels, V-alloys, SiC composites etc.) Information on Tubes & forgings for RAFM
Improvement in ductile to brittle transition temperature	Being pursued by clean steel production and heat treatments (FMS)	DBTT in fusion environment (?) (14 MeV neutrons and He)
Welding techniques	Dissimilar welding of 316L(N) – Mod. 9Cr–1Mo well established (Ni-base fillers)	To be established (?)
Component assembly and joining technologies	Well established at component level	HIP, EBW, laser welds, friction stir welding (laboratory level)
Construction of reactor	Full scale	Mock-ups for TBM

multi-pass and post weld heat treatment and irradiation effects in fusion relevant environment are some of the key issues to be addressed [37]. There is a need to improve the fabrication development knowledge base using hot isostatic pressing (HIP) for structures with internal cooling channels. Since fabrication of thick section components by HIP is a formidable task, too much dependence on HIP shall be avoided. Since the design, construction and operation of DEMO will be the next important milestone after ITER the timely availability of sound materials database, modeling/simulation/prediction methods of materials behaviour with and without irradiation, and development of a design methodology by incorporating fracture and rupture properties of the irradiated materials for ensuring the integrity of the components assume greater significance. A comparative evaluation of the current status of knowledge in fission and fusion systems is presented in the Table 4.

11. Summary

High-temperature design is generally based on experimental data created on materials of interest, a set of design rules, codes and standards that incorporate specific loading conditions and safety margins into consideration. Complex service conditions that are encountered in fusion systems such as high-temperature, irradiation, and environment, monotonic and alternating stresses promote damage interactions which cannot be combinedly simulated under laboratory conditions and on scaled down component testing. Materials used in fusion energy systems must be fully code qualified for high-temperature applications that include creep, low cycle fatigue, thermo-mechanical fatigue and creep–fatigue interaction. Irradiation effects on these properties of structural materials also need to be considered. Reliable small specimen testing methods are to be developed to establish reliable data base for validating structural design of fusion reactors. The simulation testing of materials under extreme operating conditions necessitates understanding the damage phenomena over a wide range of length and time scales, calling for a multiscale modeling approach that encompass the abinitio, molecular dynamics and kinetic techniques at the atomistic level, dislocation dynamics at meso-scale, finite element methods and continuum models at macro scale. Multiscale modeling has the potential to provide physical inputs into constitutive equations, design rules and code cases for damage interactions [37]. However the current state of modeling is not adequate for direct applications to design procedures and there is a large scope for international collaborations. Enhancing the cross-fertilization of fission and fusion R&D programmes would mutually benefit the development of both kinds of nuclear reactors for sustainable nuclear energy.

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References

[1] W.R. Crowin, Nucl. Eng. Technol. 38 (2006) 591.
[2] Generation-IV Road Map-Materials Cross-Cutting Opportunity (Online Resource).
[3] M.D. Mathew, S. Latha, G. Sasikala, S.L. Mannan, P. Rodriguez, Nucl. Technol. 81 (1988) 114.
[4] M.D. Mathew, G. Sasikala, K. Bhanu Sankara Rao, S.L. Mannan, Mater. Sci. Eng. A148 (1991) 253.
[5] G. Sasikala, S.K. Ray, M.D. Mathew, K. Bhanu Sankara Rao, S.L. Mannan, in: S.L. Mannan, M.D. Mathew (Eds.), Proceedings Seminar, Materials R&D for PFBR 1–2 January 2003, Kalpakkam, p. 105.
[6] B.K. Choudhary, K. Laha, A. Nagesha, E. Issac Samuel, V.S. Srinivasan, K.S. Chandravathi, M. Valsan, R. Kannan, K. Bhanu Sankara Rao, S.L. Mannan, in: S.L. Mannan, M.D. Mathew (Eds.), Proceedings Seminar, Materials R&D for PFBR, 1–2 January 2003, Kalpakkam, p. 121.
[7] K. Laha, K.S. Chandravathi, P. Parameswaran, K. Bhanu Sankara Rao, S.L. Mannan, Met. Mater. Trans. A 38A (2007) 58.
[8] H. Tanigawa, T. Hirose, K. Shiba, R. Kasada, E. Wakai, H. Serizawa, Y. Kawahito, S. Jitsukawa, A. Kimura, Y. Kohno, A. Kohayama, S. Katayama, R.L. Klueh, M.A.

Sokolov, R.E. Stoller, S.J. Zinkle, Technical Issues of Reduced Activation Ferritc/Martensitic Steels for the Fabrication of ITER Test Blanket Modules, ISFNT-8, 2007.
[9] A. Nagesha, M. Valsan, R. Kannan, K. Bhanu Sankara Rao, S.L. Mannan, Int. J. Fatigue 24 (2002) 1285.
[10] V.S. Srinivasan, M. Valsan, R. Sandhya, K. Bhanu Sankara Rao, S.L. Mannan, D.H. Sastry, Int. J. Fatigue 21 (1999) 11.
[11] A. Nagesha, M. Valsan, K. Bhanu Sankara Rao, R. Kannan, S.L. Mannan, Trans. Ind. Inst. Met. 58 (2005) 373.
[12] A.F. Armas, C. Petersen, R. Schmitt, M. Avalos, I. Alvarez Armas, J. Nucl. Mater. 307–311 (2002) 509.
[13] C. Petersen, D. Rodrian, Int. J. Fatigue 30 (2008) 339.
[14] B.K. Choudhary, K. Bhanu Sankara Rao, S.L. Mannan, B.P. Kashyap, Int. J. Fatigue 14 (1992) 219.
[15] Utpal Borah, Shreedhar Venugopal, R. Nagarajan, P.V. Sivaprasad, S. Venugopal, Baldev Raj, Int. J. Mech. Sci. 50 (2007) 511.
[16] S. Venugopal, P.V. Sivaprasad, S.L. Mannan, Trans. Ind. Inst. Met. 58 (2005) 1.
[17] P.V. Sivaprasad, S.L. Mannan, Y.V.R.K. Prasad, Mater. Sci. Tech. 20 (2004) 1545.
[18] M.D. Mathew, S. Venkadesan, P.V. Sivaprasad, M. Vasudeven, C. Narayanan, V. Shanmugam, S. Venugopal Jr., K.G. Samuel, R. Sandhya, S. Latha, M. Nandagopal, S. Govindarajan, K. Bhanu Sankara Rao, S.L. Mannan, in: S.L. Mannan, M.D. Mathew (Eds.), Proceedings of Seminar, Materials R&D for PFBR, 1–2 January 2003, Kalpakkam, p. 59.
[19] S. Ukai, S. Mizuta, M. Fujiwara, T. Okuda, T. Kobayashi, J. Nucl. Sci. Technol. 39 (2002) 778.
[20] .R. Lindau, A. Möslang, M. Rieth, M. Klimiankou, E. Materna-Morris, A. Alamo, A.-A.F. Tavassoli, C. Cayron, A.-M. Lancha, P. Fernandez, N. Baluc, R. Schäublin, E. Diegele, G. Filacchioni, J.W. Rensman, B.V.D. Schaaf, E. Lucon, W. Dietz, Fusion Eng. Des. 75 (2005) 989.
[21] V. Shankar, S.K. Albert, G. Srinivasan, C.R. Das, V. Ramasubbu, A.K. Bhaduri, S.K. Ray, S.L. Mannan, in: S.L. Mannan, M.D. Mathew (Eds.), Proceedings Seminar, Materials R&D for PFBR, 1–2 January 2003, Kalpakkam, p. 233.
[22] M. Vasudevan, A.K. Bhaduri, Baldev Raj, International Patent PCT/IN06/00442 dated 8 November 2006.
[23] A.K. Bhaduri, S.K. Albert, C.R. Das, M. Vasudevan, G. Srinivasan, V. Ramasubbu, S. Venugopal Jr., P.R. Sreenivasan, S.K. Ray, in: S.L. Mannan, M.D. Mathew (Eds.), Proceedings Seminar, Materials R&D for PFBR, 1–2 January 2003, Kalpakkam, p. 263.
[24] M. Vasudevan, M. Murugananth, A.K. Bhaduri, Baldev Raj, K. Prasad Rao, Sci. Technol. Weld. Joining 9 (12) (2004) 109.
[25] V. Karthik, K. Laha, P. Parameswaran, K.V. Kasiviswanathan, Baldev Raj, J. Test. Eval. 35 (2007) 438.
[26] V. Karthik, A. Vijayaraghavan, K. Laha, K.V. Kasiviswanathan, Baldev Raj, Trans. Ind. Inst. Met. 70 (5) (2007) 487.
[27] RCC-MR Subsection NB for Class 1 Components, Design and Construction Rules for Mechanical Components of FBR Nuclear Islands, 2002, AFCEN, Paris, France.
[28] RCC-MR subsection Z, Technical Appendix A16, Guide for Leak before Break analysis and Defect assessment, 2002 AFCEN, Paris, France.
[29] ASME Sec III, Appendix II -1000: Experimental Stress Analysis, 2001.
[30] P. Chellapandi, A. Biswas, R. Srinivasan, S.C. Chetal, S.B. Bhoje, Trans. Ind. Inst. Met. 58 (2005) 207.
[31] P. Chellapandi, S.C. Chetal, Baldev Raj, Int. J. Press. Vessels Piping 82 (2005) 739.
[32] V. Moorthy, B.K. Choudhary, S. Vaidyanathan, T. Jayakumar, K. Bhanu Sankara Rao, Baldev Raj, Int. J. Fatigue 21 (1999) 263.
[33] Anish Kumar, T. Jayakumar, Baldev Raj, K.K. Ray, Mater. Sci. Eng. A 362 (2003) 58.
[34] V. Moorthy, T. Jayakumar, Baldev Raj, Mater. Sci. Tech. 12 (1996) 56.
[35] Baldev Raj, V. Moorthy, T. Jayakumar, K. Bhanu Sankara Rao, Int. Mater. Rev. 48 (2003) 273.
[36] E.E. Bloom, J.T. Busby, C.E. Duty, P.J. Maziasz, T.E. McGreevy, B.E. Nelson, B.A. Pint, P.F. Tortorelli, S.J. Zinkle, J. Nucl. Mater. 367–370 (2007) 1.
[37] M. Samaras, W. Hoffelner, M. Victoria, J. Nucl. Mater. 371 (2007) 28.