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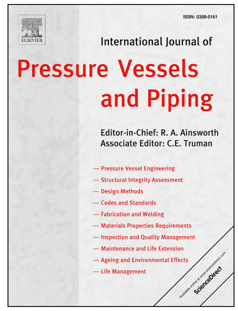

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Abstract

The performance of fast reactor (FR) working under high temperature, moderate pressure and intense neutron irradiation depends on its structural materials. Type 316LN austenitic stainless steel with 0.02-0.03 wt.% carbon and 0.06-0.08 wt.% nitrogen is generally used for structural components of FR along with modified 9Cr-1Mo ferritic steel for steam generator. With an aim of enhancing the life of FR, efforts towards the enhancement of strength of the materials are been described. Nitrogen content in the 316LN steel has been increased for optimizing the tensile, creep, fatigue and creep-fatigue interaction strengths. Both the tensile and creep strength were found to increase with nitrogen content; whereas the fatigue and creep-fatigue interaction strength have optimum values at around 0.12 wt. % of nitrogen. Based on the study, nitrogen content in 316LN steel has been optimized at 0.11-0.13 wt. % for optimum combination of strengths. Weldability study has been performed and matching welding consumable has been developed.

Creep strength enhancement of modified 9Cr-1Mo steel and its fusion welded joint has been addressed through microalloying the steel with boron along with the restriction on nitrogen content. The microalloying not only improves the creep strength of the steel also reduces the type IV cracking susceptibility of the fusion welded joint of the steel. The optimum contents of boron and nitrogen in the steel have been identified as around 60 ppm boron and 100 ppm nitrogen for better creep strength and type IV cracking resistance. The paper describes the challenges in developing the materials.

Keywords: Sodium-cooled fast reactors; austenitic stainless steel; Nitrogen-enhanced; Reactor structural; Ferritic steel; Boron-added; Type-IV cracking; Steam generator

1. Introduction

India has envisioned a three-stage nuclear energy programme to utilize its vast thorium resources to reduce the carbon foot print [1]. Sodium-cooled fast reactor (SFR) technology represents second of the three-stage nuclear programme. The Fast Breeder Test Reactor (FBTR) of 40 MWth (thermal) is on operation for over 25 years at the Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, which helped to master the complex FR technology [2]. Based on this experience, a 500 MWe Prototype Fast Breeder Reactor (PFBR) has been designed indigenously and is at advanced stage of construction at Kalpakkam. For the future reactors, the design is being further optimised for enhancing economy with respect to cost of electricity production.

Type 316 austenitic stainless steel and its variants are the preferred candidate materials for high-temperature reactor structural components operating above 700 K, due to their adequate high-temperature strengths, compatibility with liquid sodium coolant, ease of fabrication, weldability and commercial availability. For application in nuclear reactor, close control on chemical composition of the steel has been imposed to avoid the scatter in mechanical properties. However, the austenitic stainless steels suffer from intergranular stress-corrosion cracking (IGSCC) in chloride and caustic environments, especially in heat affected zone and deposited weld metal of fusion welded components due to the combined influence of sensitization and presence of residual stresses introduced during weld thermal cycle. Hence, for the fast reactor application, carbon content in the 316 steel has been restricted to less than 0.03 wt.% and nitrogen (0.06-0.08 wt.%) has been added to compensate the strength loss. The low carbon steel with added nitrogen is designated as 316LN. The 316LN steel has higher

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