ENVIRONMENTAL DEGRADATION TESTING OF ALLOY 709

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INTRODUCTION: Alloy 709 is an advanced austenitic stainless steel developed to meet the demanding service conditions of sodium fast reactors (SFRs) but may also be of use for lead fast reactors, high temperature gas reactors, and molten salt reactors. Understanding environmental degradation modes and quantifying degradation rates are crucial to ensure the long-term, reliable operation of these reactors. This paper discusses recent neutron irradiation results, future neutron irradiation test plans, and sodium corrosion testing to communicate the current best understanding of the performance of Alloy 709 in SFR environments.

1. OVERVIEW

Structural materials for fast reactor components require good high temperature mechanical properties in the unirradiated and irradiated states as well as adequate compatibility with coolants such as sodium or lead. The high operating temperatures of sodium fast reactors make the time-dependent mechanical properties essential for structural materials, while the corrosive coolant and the intense fast neutron spectrum present additional challenges. The overall efficiency of the power conversion cycle is improved by higher coolant outlet temperatures, driving interest in developing new structural materials that can perform at higher temperatures while maintaining compatibility with the chemical and radiation environments.

The United States Department of Energy Office of Nuclear Energy (DOE-NE) has developed a comprehensive plan to develop and deploy Alloy 709 (A709) as a structural material for high temperature reactors under the Advanced Reactor Technologies - Fast Reactor Program. A709 is an advanced austenitic stainless steel derived from the Japanese alloy NF709 (ASTM A213), which was developed by Nippon Steel for seamless tubing. Crucially, A709 possesses enhanced high temperature strength compared to existing austenitic stainless steels within the American Society of Mechanical Engineers (ASME) Section III, Division 5 code for Class A construction, which covers materials for high temperature nuclear reactors. Specifically, A709 has been optimized with respect to its chemical composition and heat treatment to enable higher operating temperatures than conventional grade 316H stainless steel [1].

Due to its superior high temperature strength, A709 is a promising candidate structural material for sodium cooled fast reactors (SFRs) as well as other advanced reactor concepts such as lead fast reactors, molten salt reactors (MSRs), and high temperature gas reactors (HTGRs). Deployment of A709 will enable reactor vendors to reduce plant capital costs, expand the thermal design window, and increase safety margins. To date, the A709 research and development program has acquired the design data needed for the 100,000 h ASME Section III, Division 5 Code Case, and it is on schedule to submit the draft Code Case to ASME for approval in 2026.

As part of the material qualification effort for A709, the compatibility of the material with sodium and the neutron irradiation response are being tested. Regulatory acceptance requires additional understanding on the service performance and degradation mechanisms in an application-specific environment. A709 is being tested in flowing sodium loops at Argonne National Laboratory and is being irradiated in both the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory and the Advanced Test Reactor (ATR) at Idaho National Laboratory within the United States.

2. BASIC METALLURGICAL CONSIDERATIONS

The nominal A709 composition is 20Cr-25Ni-1.5Mo-Nb-N-C (in weight percent) with trace elements of Ti, B, Co and impurities of Si, Cu, Al, P and S. Thus, A709 has higher nickel and chromium contents than 316H stainless steel and can be hardened with precipitation treatment. A709 is more resistant to sensitization than 316H stainless steel due to its higher Cr content and the fact that Ti and Nb are stronger precipitate formers than Cr. The increased Ni content also helps stabilize the austenite phase to counteract Cr, which is a ferrite phase stabilizer. The exact specification for the A709 composition can be found in the ASTM A240 standard under UNS number S31025, while the solution annealing specification is listed in standard ASTM A480. The precipitation-hardened condition will be added as a supplementary requirement in the specification, but it entails heat treatment at 775 °C for 10 hours.

A709 will experience radiation damage during its deployment for sodium fast reactors and other advanced reactors. Expected phenomena include hardening, swelling, radiation-induced segregation and precipitation, irradiation-induced creep, and the generation of helium via the transmutation of constituent alloy elements. These degradation phenomena will likely lead to the loss of tensile ductility, creep strength and creep elongation to failure, creep-fatigue resistance and fatigue cycles to failure [2]. Because A709 is a precipitation-hardening alloy with multiple second phases that may form, the effect of extended elevated temperature and irradiation on the precipitate structure and resulting mechanical properties is difficult to predict. Radiation-induced segregation to grain boundaries and dislocations is likely, which may affect precipitation. Helium generation, especially due to its high nickel content, is likely to induce high temperature helium embrittlement, though the amount of helium generated per unit dpa depends on the neutron spectrum.

A709 is expected to be generally compatible with sodium, but the corrosion of alloys in liquid sodium is primarily governed by the solubility of the alloy's constituents in liquid sodium and, to a lesser extent, by chemical reactions. Key corrosion mechanisms include metal dissolution or sodium alloying, mass transfer driven by temperature and compositional gradients, and corrosion caused by non-metallic impurities such as oxygen, carbon, and nitrogen in liquid sodium. The operating temperature, temperature gradient, and flow rate of the sodium all influence the corrosion extent. In the case of A709, preferential leaching of Ni is expected, as well as corrosion by oxygen impurities and carburization/decarburization by carbon transfer. Together with the elevated temperature, these degradation mechanisms may lead to microstructural evolution, especially precipitation. Thus, the effect of sodium on precipitates, especially carbides, must be understood.

3. NEUTRON IRRADIATION OF A709

To date, several A709 sub-sized tensile [3] specimens and miniature notched bend bar [4] fracture toughness specimens have been irradiated in HFIR at target temperatures of 400 °C and 600 °C up to approximately 2 displacements per atom (dpa). Specimens irradiated up to 10 dpa will be removed in the coming months. Tensile testing of unirradiated and irradiated specimens was performed at a strain rate of 10^{-3} s^{-1} at 23 °C, 400 °C, and 600 °C. Some reduction in uniform elongation and total elongation was observed (from approximately 40% to 30% following irradiation at 400 °C and 20% at 600 °C) when tensile tested at the irradiation temperature. Note that the vast majority of the A709 deformation occurred via uniform elongation. Fracture toughness testing and transmission electron microscopy are in progress and will be completed in the future.

A more extensive neutron irradiation campaign is planned for A709 that will use both HFIR and ATR [2]. This campaign will include neutron irradiation with the thermal neutron spectrum of HFIR and will include the use of cadmium shrouds for thermal neutron filtering in ATR to achieve a spectrum more representative of a fast reactor. Specimens will be irradiated to 1 and 2 dpa in HFIR and ATR and 10 dpa in HFIR in the solution-annealed and precipitation-treated condition at target temperatures between 300 °C and 800 °C to investigate nominal as well as lower- and upper-bound deployment temperatures. The material will be characterized for fracture toughness, tensile properties, creep, creep-fatigue, and creep crack growth. The initial phase of the irradiation campaign will involve post-irradiation testing of material, but successive phases intend to perform in-reactor testing of creep properties.

4. SODIUM EXPOSURE OF A709

Solution-annealed and precipitation-treated specimens from a single heat of A709 have been exposed to flowing sodium in forced-convection loops at 550 °C, 600 °C, and 650 °C for 3000 h, 10,000 h, and 20,000 h [5]. The same material has also been subjected to thermal aging experiments for comparison. Following the sodium exposure tests, no elevated corrosion damage was observed and the obtained steady-state corrosion rate was 0.1 μm/yr at 550 °C and 0.6 μm/yr at 650 °C. There were observed differences in the precipitate microstructure and composition for specimens exposed to flowing sodium versus those that are only thermally aged, suggesting some effects of sodium exposure on the microstructure of A709. However, these differences were insignificant for the test conditions, and the mechanical properties, including the yield stress, ultimate tensile stress, uniform elongation, and total elongation, were largely similar for the thermally-aged specimens and the sodium-exposed specimens. This contrasts sharply with observations of 316H stainless steel, where sodium exposure significantly affects the post-exposure tensile properties at 650 °C. It is evident that, under the current test conditions, A709 outperforms 316H stainless steel with no noticeable impact of sodium on its mechanical properties at temperatures below 650°C. In addition, thermodynamic calculations indicate that A709 is less prone to decarburization in a typical SFR environment compared to 316H stainless steel, making it more likely to retain adequate performance at elevated temperatures.

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