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Radiation response of the overlay cladding from the decommissioned WWER-440 Greifswald Unit 4 reactor pressure vessel

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HIGHLIGHTS

- Tensile and crack extension testing of the austenitic overlay cladding from decommissioned reactor pressure vessels.
- Engineering crack initiation fracture toughness values according to ASTM E1820 were evaluated.
- Due to the inhomogeneous structure of the welded overlay cladding a significant scatter was observed in the initiation values.
- The J–Δa values show discontinuities caused by fast crack propagation because of low tearing strength or crack jumps.
- The measured fracture toughness values indicate that the cladding would remain intact during pressurized thermal shock transient.

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ABSTRACT

The Results of tensile and crack extension testing conducted on irradiated austenitic overlay cladding material are presented. The specimens were machined from three trepans sampled from the decommissioned WWER-440/V-230 reactor pressure vessel of the nuclear power plant Greifswald Unit 4. Crack extension curves were measured with Charpy size SE(B) specimens using the unloading compliance technique according to ASTM E1820-11 at different temperatures. Provisional crack initiation fracture toughness values J_Q and K_{J_Q} are determined. The highest K_{J_Q} values were found in the temperature range from 20 to 80 °C. A significant scatter was observed in the initiation values. This is due to the inhomogeneous structure of the welded overlay cladding. During the loading the crack moves through regions with different tearing strength, thus the J–Δa values show discontinuities caused by fast crack propagation or crack jumps. The comparison of the measured K_{J_Q} values and conservatively estimated stress intensity factors at an assumed surface crack shows that the cladding would remain intact during pressurized thermal shock transient.

1. Introduction

Austenitic overlay cladding has been originally designed to protect the low alloyed RPV base and weld metals against corrosion. Usually, the overlay cladding remains out of consideration in the current RPV integrity assessment regulatory codes (Brumovsky et al., 1997; Gillemot et al., 2007). Some codes allow the inclusion of the overlay cladding on condition that its properties are known during the RPV lifetime (Brumovsky, 2003; Gillemot et al., 2007). The fracture behaviour of an operating nuclear RPV, particularly during certain overcooling transients, may strongly depend on the properties of the irradiated overlay cladding (Haggag et al., 1990). Investigations have shown that the integrity of the overlay cladding strongly influences the loading of a crack in the base or weld metal during an emergency PTS event. An intact overlay cladding clamps an under clad crack and results in a lower stress intensity factor K_I at the crack tip. In contrast a surface crack is additionally loaded due to the tension stress in the overlay cladding caused by different thermal properties with respect to the base or weld metal (Abendroth and Altstadt, 2007). Therefore, knowledge about the properties of the overlay cladding, mainly of its fracture toughness, is also necessary for a precise evaluation of the RPV integrity during a PTS event. Overlay cladding material is currently not included in the RPV surveillance specimen programmes. Therefore, mechanical and fracture toughness data are not available from operated RPVs. In the literature there are test results from the overlay cladding investigated within the acceptance tests and irradiations experiments performed in research reactors. Results on the RPV overlay cladding and especially on their irradiation response are limited compared to base and weld metal. Irrespective

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Nomenclature

a	crack depth (small half axis of the ellipse)
a_i	actual crack length determined by unloading compliance
a_0	initial crack length
a_{0q}	initial crack length estimated according to ASTM E1820 with the a_i values up to maximum load
bcc	body centred cubic
c	large half axes of the ellipse
COD	crack opening displacement
E	Young's modulus
FEM	finite element method
J_{Ic}	J_Q becomes J_{Ic} when properly qualified against the criteria proposed in the test standard procedures ASTM E1820
J_Q	J integral a at 0.2 mm crack extension excluding blunting (according to ASTM E1820)
K_I	stress intensity factor
$K_{J_{Ic}}$	J_{Ic} based fracture toughness according to ASTM E1820
K_{J_Q}	J_Q based fracture toughness according to ASTM E1820
NPP	nuclear power plant
$R_{p0.2}$	proof strength at 0.2% plastic deformation
R_m	ultimate tensile strength
RPV	reactor pressure vessel
PTS	pressurized thermal shock
s	thickness of the cladding
SE(B)	single edge bend specimen
SIF	stress intensity factor
T	temperature in K
UC	unloading compliance technique
WWER	Russian type pressurized water reactor (water operated and moderated)
Φ	neutron fluence
σ	uniform hoop stress

of the various cladding technologies the following main features can be summarized (Haggag et al., 1990; Alekseenkov et al., 1997; Brumovsky et al., 1997; Timofeev and Karzov, 2006; Gillemot et al., 2007; Margolin et al., 2010)

- The austenitic cladding shows a ductile-to-brittle transition behaviour similar to bcc steels because of the weld structure containing 3–10% of δ ferrite. Specimens which failed in brittle mode contain areas of cleavage associated with the δ ferrite phase and ferrite–austenite interphases fractured by interphase separation.
- The increase of the ultimate tensile strength with decreasing temperature is more pronounced than it is for the yield strength.
- The crack initiation fracture toughness vs. temperature curve runs over a peak at room temperature, with unstable crack propagation at low temperature and low tearing strength above 200 °C.
- There is a susceptibility to neutron irradiation, which leads to an increase of the yield and ultimate strength, the appearance of brittle crack extension and a decrease of the tearing strength.
- Depending on the test temperatures, unstable failure of specimens or crack jumps after preceding ductile crack extension were observed. Margolin et al. (2010) suggests the introduction of a limit temperature above which intercrystalline fracture and brittle crack jumps are not observed. In addition, a critical numerical fracture toughness value, J_c^{jump}, for the brittle crack initiation or crack extension was specified with 65 kJ/m^2 for WWER RPVs.
- The progression of crack extension curves measured on the cladding from WWER-440 and WWER-1000 RPVs exhibits discontinuities because of the heterogeneous weld structure (Margolin et al., 2010).

Low fracture toughness and the occurrence of brittle failure at elevated temperatures raise concerns over the potential adverse impact of the austenitic overlay cladding on the integrity of the RPV during a PTS transient. The growth of a surface flaw can lead to the failure of the cladding and significantly increase the probability of vessel failure (Haggag et al., 1990; Abendroth and Altstadt, 2007; Margolin et al., 2010). The initiation of brittle crack extension in the cladding below a threshold fracture toughness value has to be avoided for temperatures above 20 °C (Margolin et al., 2010). This threshold value has to be specified depending on the operation and emergency conditions of the NPP to be assessed. Therefore knowledge about the fracture toughness within the temperature range of a PTS transient and at operation temperature of the RPV are of high importance.

The most realistic evaluation of the toughness response of RPV overlay cladding to irradiation may be achieved by directly studying RPV wall samples taken from decommissioned RPVs. Such a possibility is available with the investigation of samples taken from the RPVs of the Greifswald NPP. Four WWER-440/V-230 nuclear reactors representing the first generation of this reactor type were operated between 11 and 15 years and were decommissioned in 1990. The RPVs were designed and manufactured in the former Soviet Union at the end of the 1960s and the beginning of the 1970s. The operation history, the expected neutron fluences and the material conditions of the units 1–4 are presented elsewhere (Konheiser et al., 2006; Viehrig et al., 2009, 2010, 2012a,b). While the RPVs of the units 1 and 2 do not contain an austenitic anticorrosive overlay cladding, the RPVs of the units 3 and 4 are cladded. Trepans were extracted from the RPV of the units 1, 2 and 4. The description of the trepanning device and the sampling procedure and the results of investigations on the base metals and the weld materials of the RPVs were published elsewhere (Konheiser et al., 2006; Viehrig et al., 2009, 2010, 2012a,b, 2015). This paper presents the test results measured on the austenitic overlay cladding of trepans sampled from different locations within the reactor core region of the Greifswald Unit 4 RPV, which was shut down after 11 operating cycles. The main focus is on the measurement of the crack extension curves and the evaluation of initiation fracture toughness values according to the test standard ASTM E1820-11.

2. Material and specimens

The RPV of Greifswald Unit 4 is overlay cladded by austenitic stainless materials using automatic submerged welding methods. The overlay cladding is welded with several passes using ribbon material of different composition, and hence represents a multilayer structure. Following requirements have to be met (Alekseenkov et al., 1997; Brumovsky et al., 1997; Timofeev and Karzov, 2006; Gillemot et al., 2007; Margolin et al., 2010):

- Prevention of hot cracks and a martensitic structure in the first layer diluted with base metal,
- Prevention of the overlay cladding embrittlement due to the formation of brittle secondary carbides and intermetallic phases,
- High resistance against intercrystalline corrosion, and
- Prevention of the embrittlement in adjoining zones of base metal because of high temperature thermo strained cycles during the welding of the cladding.