SAFE: Safe LTO in the context of environmental effects on fracture, fatigue & EAC

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The SAFE-I (2012 – 2014) and SAFE-II (2015 – 2017) projects are supported by the Swiss Nuclear Safety Inspectorate ENSI and aim to fill selected important knowledge gaps in the field of environmentally-assisted cracking (EAC) and environmental effects on fatigue and rapid fracture in pressure boundary components in the primary coolant circuit of light water reactors (LWR).

Background & motivation: Pressure boundary components in the primary coolant circuit of LWRs are made of low-alloy and stainless steels and are very critical components with regard to safety and lifetime. During service, toughness and ductility of these materials can decrease with time, due to irradiation induced embrittlement (RPV only), thermal ageing or potential environmental (hydrogen) effects. Under simultaneous effect of the reactor coolant, thermo-mechanical operational loads and irradiation, cracks can initiate and grow by environmentally-assisted cracking (EAC) and thermo-mechanical fatigue (TMF), which finally could lead to a large leak or component failure. Several EAC and TMF cracking incidents occurred in both boiling water (BWR) and pressurised water reactors (PWR) in a wide range of stainless steel, nickel-base alloy, carbon and low-alloy steel components in the last three decades. Critical components are thus periodically inspected by non-destructive examination to detect defects before they reach a critical size necessary for rapid fracture.

An accurate knowledge on the degradation of the toughness and fracture properties of these materials during service and of the system conditions which may lead to EAC initiation and growth is thus evidently indispensable to ensure the safe and economic long-term operation in this context. Reliable quantitative experimental data on these phenomena and a basic knowledge on the underlying mechanisms are essential to evaluate their possible effects on structural integrity/safety and lifetime of components, to identify critical component locations/operating conditions and to define and qualify possible mitigation, repair and maintenance actions.

Within various sub-projects of SAFE-I & SAFE-II the following unexplored aspects and concerns are currently evaluated in this field:

1. Environmental effects on fracture toughness and tearing resistance of RPV steels
2. Stress corrosion cracking in Alloy 182 dissimilar metal welds
3. Stress corrosion cracking initiation in Alloy 182 weld metal
4. Environmental effects on fatigue in stainless steels

1. Environmental effects on fracture toughness and tearing resistance

Fracture toughness and tearing resistance are material properties, which not only depend on microstructure or loading conditions (e.g. strain rate or constraints) but are also strongly influenced by the environment in which the cracking occurs. Except for temperature and irradiation, the effect of environment on fracture behaviour has not been taken into account in the nuclear power industry. There is now growing experimental evidence that the fracture resistance of most structural materials might be degraded by reactor coolant (hydrogen) effects in the LWR operating regime.

This sub-project aims to establish and quantify the unexplored role of the high-temperature water environment and hydrogen on the fracture and mechanical behaviour of low-alloy RPV steels in the LWR temperature regime and identify the underlying mechanism and the critical combinations of metallurgical, environmental and loading conditions that may result in significant environmental and hydrogen effects. Synergies with other related degradation modes like EAC and dynamic strain ageing are also explored. For these reasons, an experimental parameter study on environmental & hydrogen effects on the fracture behaviour in the upper shelf & brittle to ductile transition region in different RPV steels with systematic variation of environmental, material and mechanical loading parameter is performed. In preliminary investigations, no clear evidence for environmental reduction of upper shelf fracture toughness in RPV steel in high-temperature water at 288 °C was found although a clear change in fracture morphology was observed. In a next step, the possibility of ductile to brittle transition temperature (DBTT) shifts and reduction of resistance to brittle failure by environmental and hydrogen effects shall be explored. If such effects should occur, the evaluation of synergies with irradiation and temper embrittlement should be checked also.

Recent publications:

S. Roychowdhury, H.P. Seifert, P. Spätig, Z. Que, S. Ritter, Effect of high temperature water and hydrogen on the fracture behaviour of a low-alloy reactor pressure vessel steel, in Proceedings of 17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Ottawa, Ontario, Canada, August 9-13, 2015.

2. Stress corrosion cracking in Alloy 182 dissimilar metal welds

The nickel-base Alloy 182 has been widely used as a weld filler and attachment pad metal to join the low-alloy steel reactor pressure vessel (RPV) and pressure vessel nozzles to both wrought nickel-base alloys (Alloy 600) and austenitic stainless steel (304L, 316L, 316NG) components in LWRs by manual shielded metal arc welding. The recent stress corrosion cracking (SCC) incidents in control rod drive mechanisms and core shroud support welds in Japanese and European BWRs represent a serious safety concern. In these highly constrained Alloy 182 welds with very high residual stresses, the stress intensity factors of SCC cracks with crack-tips in the interface region between the weld metal and adjacent low-alloy RPV steel can reach high values. Under these conditions, the possibility of fast SCC into the RPV in BWR normal water chemistry (NWC) environment cannot be excluded, in particular in high-sulphur RPV steels.

The goal of this sub-project is thus to characterise the SCC crack growth perpendicular to the interface region between the Alloy 182 weld metal and adjacent RPV steel in BWR environment in the high K_I region and to quantify the thresholds for K_I and chloride content for fast SCC crack growth into the RPV steel. This project is performed in collaboration with the Tohoku University and Japanese Nuclear Energy Safety Organization (JNES).

Critical system conditions for fast SCC into the adjacent low-alloy RPV steels were identified for the stress intensity factor K_I in the fusion boundary region, the chloride content and corrosion potential ECP (Figures 1 & 2). The observed SCC cracking behavior correlates excellently with the field experience, where SCC cracking was confined to the Alloy 182 weld metal and no cases of SCC were observed in low-alloy steel primary pressure boundary components. Under static loading conditions in chloride-free high-temperature water, there seems to be little risk that a fast growing interdendritic SCC crack may cross the fusion line and significantly propagate into the adjacent low-alloy RPV steel. On the other hand, tests in BWR/NWC water with addition of small amounts of chloride revealed that sustained crack growth across the fusion line into the RPV steel with very high crack growth rates of several cm/year is easily possible. A modification of the current Action Level 1 or chloride of the EPRI water chemistry guidelines should therefore be pursued. In BWR hydrogen water chemistry (HWC) or PWR environment at lower ECP, the chloride tolerance with respect of fast SCC is much higher.

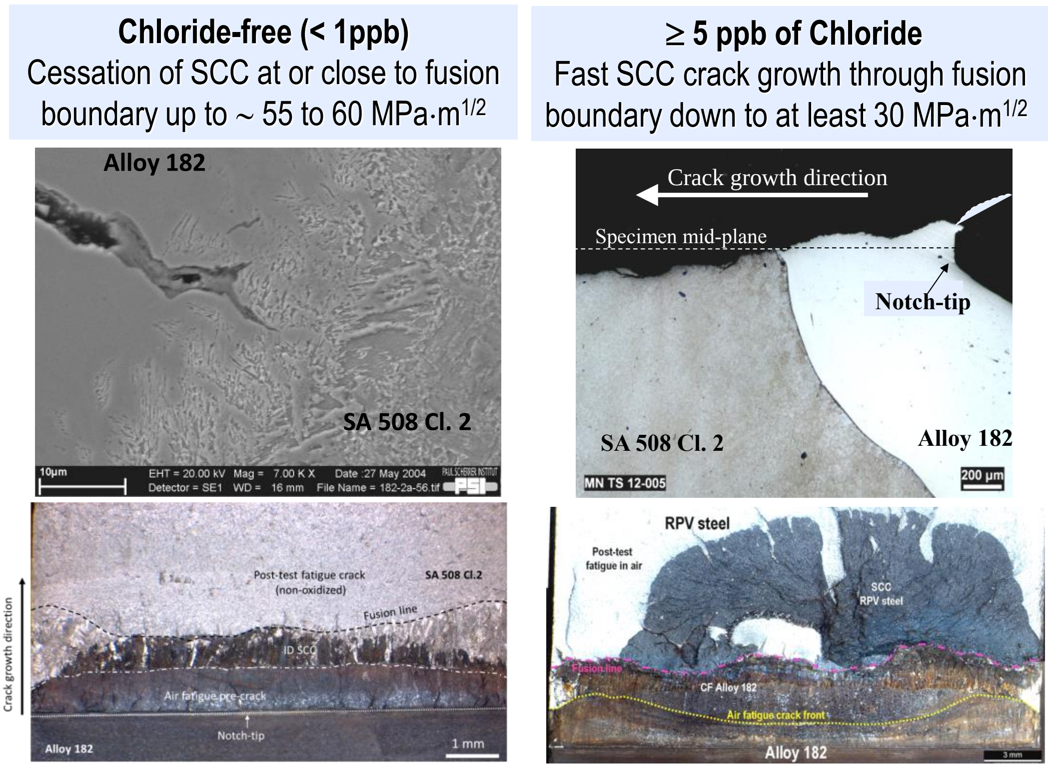


Figure 1: SCC behaviour perpendicular to fusion boundary in BWR/NWC environment.

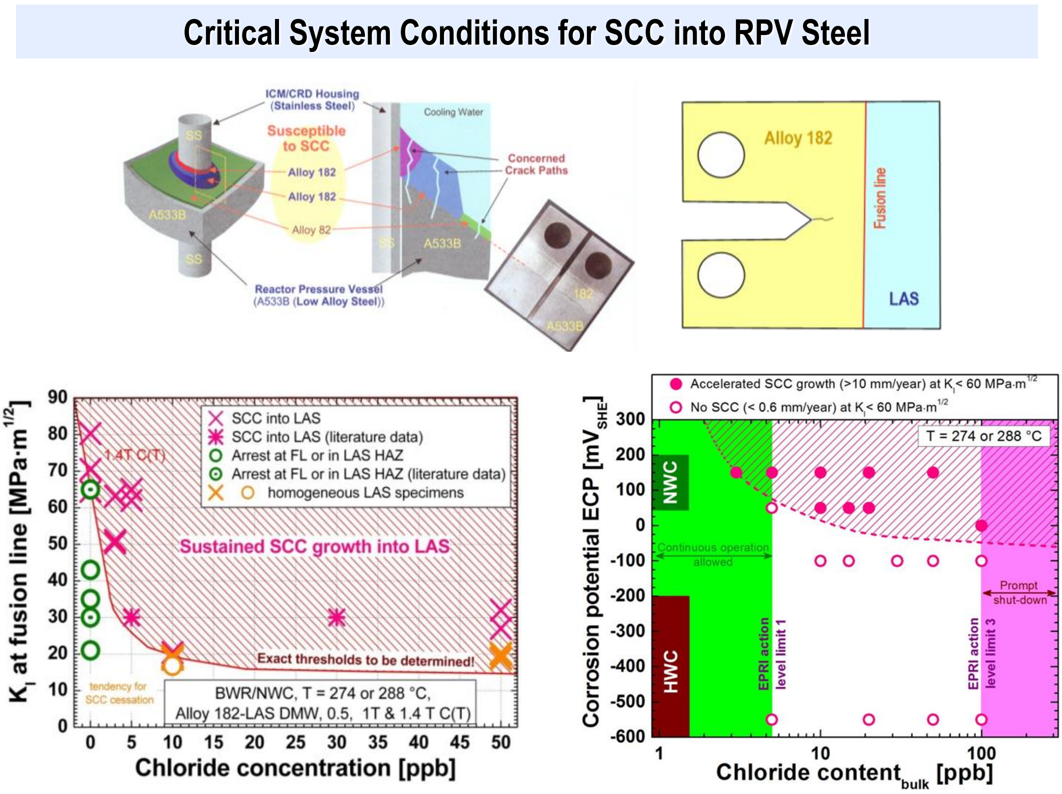


Figure 2: Critical system conditions for SCC into RPV steel.

The evaluation of the residual stresses in such dissimilar metal welds, e.g. by mock-up welds and weld residual stress simulations, and the accurate estimation of stress intensity factor profiles at the fusion boundary for different weld configurations and procedures is a critical issue and key task for reliable integrity and safety assessments in this context.

Recent publications:

H.P. Seifert, S. Ritter, H.J. Leber, S. Roychowdhury, "Stress Corrosion Cracking Behavior in the Transition Region of Alloy 182/Low-Alloy Reactor Pressure Vessel Steel Dissimilar Metal Weld Joints in Light Water Reactor Environments", Corrosion, Vol. 71, No. 4, 2015, pp. 433-454.

H.P. Seifert, S. Ritter, T. Shoji, Q.J. Peng, Y. Takeda, Z.P. Lu, "Environmentally-Assisted Cracking Behaviour in the Transition Region of an Alloy 182/SA 508 Cl.2 Dissimilar Metal Weld Joint in Simulated Boiling Water Reactor Normal Water Chemistry Environment ", Journal of Nuclear Materials, 378(2), 2008, pp. 197 – 210.

3. Stress corrosion cracking initiation in Alloy 182

In recent years several SCC cracking incidents occurred in Alloy 182 dissimilar metal welds in BWRs and PWRs, which seriously challenged the integrity of the primary coolant circuit in some cases. Components such as different reactor pressure vessel nozzle safe ends and reactor pressure vessel penetration or attachment welds suffered from SCC. In 2012, such an SCC crack in Alloy 182/82 was observed in the feedwater nozzle of nuclear power plant Leibstadt (KKL, BWR), which penetrated almost the full wall thickness and has resulted in a long shut-down period. An overlay cladding repair welding with a more resistant weld metal was applied. Because of the significant SCC risk in Alloy 182 weldments of reactor pressure vessel head penetrations in PWRs, nuclear power plant Beznau (KKB) plans to proactively exchange the reactor pressure vessel heads in both reactors in 2015. Both cases thus further confirm the relevance of this topic also in the Swiss context.

SCC crack growth and possibly initiation are strongly affected by the dissolved hydrogen (DH) content in PWR environment in Ni-base alloys specifically. A temperature-dependent peak in SCC growth rate has been observed at a certain critical DH concentration, which corresponds to the thermodynamic Ni/NiO phase boundary in the corresponding Pourbaix diagram and is close to typical primary-side PWR operating conditions. A weak maximum in initiation susceptibility was observed in recent experiments with wrought Ni-base Alloy 600, whereas the very limited amount of tests with Alloy 182 weld metal do not allow a clear conclusion so far. As most BWRs (including the two Swiss reactors) are injecting hydrogen into the feed water (hydrogen water chemistry (HWC) or HWC combined with NobleChem^{TM}) to mitigate the high SCC susceptibility in austenitic stainless steels and Ni-base alloys under oxidising normal water chemistry conditions, this maximum in initiation susceptibility represents a certain concern. There is no evident reason, why the same effects of DH that have been reported in PWR primary water should not also apply to pure BWR water.

The main scientific goal of this PhD thesis is to evaluate the unexplored effect of DH contents on the SCC initiation and short crack growth in Alloy 182 weld metal under BWR/HWC conditions at 274 °C. The results will help to identify optimal DH levels for SCC mitigation in BWRs. For this purpose, SCC crack initiation and the subsequent short crack growth shall be studied with sharply notched fracture mechanics and smooth tensile specimens in modern autoclave high-temperature water loop systems. Controlled active loading of multiple specimens and on-line initiation and crack growth monitoring are the two main advantages of the selected novel and unique approach, which will provide more reliable initiation data and a better separation of initiation and short crack growth contributions. New insights into the SCC initiation mechanisms and typical initiation sites are expected from the metallographical and metallurgical post-test investigations. Based on the test results and post-test observations a tentative mechanistic interpretation of the SCC initiation and short crack growth process shall be elaborated.

4. Environmental effects on fatigue in stainless steels

The accumulated excellent field experience of fatigue-designed primary pressure boundary components is excellent. The few fatigue cracking incidents in LWRs in recent years were mainly related to either high-cycle fatigue (HCF) through flow-induced vibrations from power up-ratings (e.g., BWR steam dryers or socket welded small diameter instrument lines) or to thermo-mechanical fatigue (TMF) caused by thermal-hydraulic phenomena like thermal stratification and striping or turbulent mixing (e.g., pressuriser surge lines or residual heat removal system in PWRs). In contrast to the HCF damage due to high-frequency vibrations, where strain rates are too high for significant environmental effects, the low-cycle fatigue (LCF) or combined LCF and HCF damage due to thermal loadings may have been aggravated by corrosion effects due to exposure to the reactor coolant, although the extent and contribution of environmental effects remain unclear.

Environmental effects on fatigue are fairly well established under lab conditions and there is no evident reason, why such environmental effects should not occur in the field. Although the possibility of environmental effects on fatigue initiation in LWR environments is undisputed, there is no international consensus about the practical consequences and their adequate implementation in fatigue design codes. Based on laboratory investigations, different proposals were established for incorporating environmental effects into the fatigue design and evaluation procedures according to

Section III and IX of the American Society of Mechanical Engineers (ASME) boiler and pressure vessel (BPV) code or even implemented in certain national codes (e.g., in Japan). The practical application of these procedures is complex and also related to some uncertainties. Furthermore, there are relevant differences between fatigue design, lab tests and component conditions in the field (e.g., thermo-mechanical fatigue, mean stress, static load hold times, …). The suggested procedures have thus not (yet) found full acceptance by the industry.

To close some knowledge and data gaps in this field, the fatigue crack initiation behaviour of different wrought low-carbon and stabilised austenitic stainless steels is currently characterised under simulated BWR and primary PWR conditions in LNM. The special emphasis in SAFE is hereby placed to unexplored plant-relevant aspects, which may result either in undue conservatism or in a lack of conservatism in fatigue evaluation procedures. The following aspects are currently evaluated in high-temperature water:

- in-phase (IP) and out-of-phase (OP) TMF initiation behaviour
- effect of mean stress and stress state on fatigue initiation
- effect of load history/load sequence on fatigue initiation
- effect of long static load hold times on fatigue initiation

The adequacy of typical mean stress correction (e.g., SWT, …) & damage accumulation methods (e.g., Miner, …) for fatigue in high-temperature water shall be evaluated. The generated results shall help to improve current Code proposal for environmental fatigue analysis.

For this purpose, load-controlled fatigue tests with sharply notched fracture mechanics specimens in autoclaves and stress- or strain-controlled isothermal LCF and IP or OP TMF tests with tubular smooth specimens are performed in world-wide unique facilities (Figures 3 & 4). The latter systems allow fatigue experiments with very small strain amplitudes or complex stress/strain profiles with superimposed rather rapid temperature changes (100 – 340 °C) under flowing conditions.

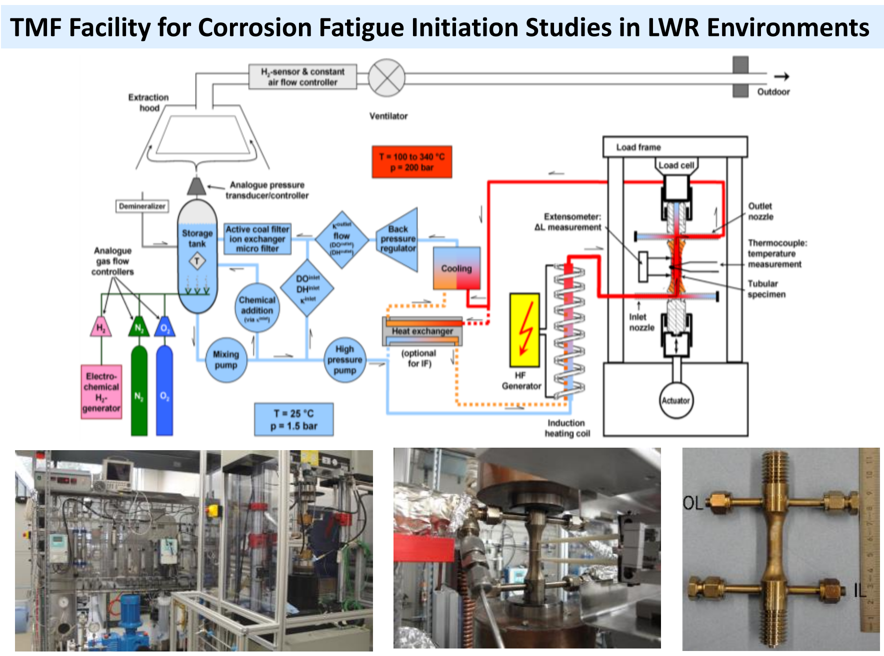
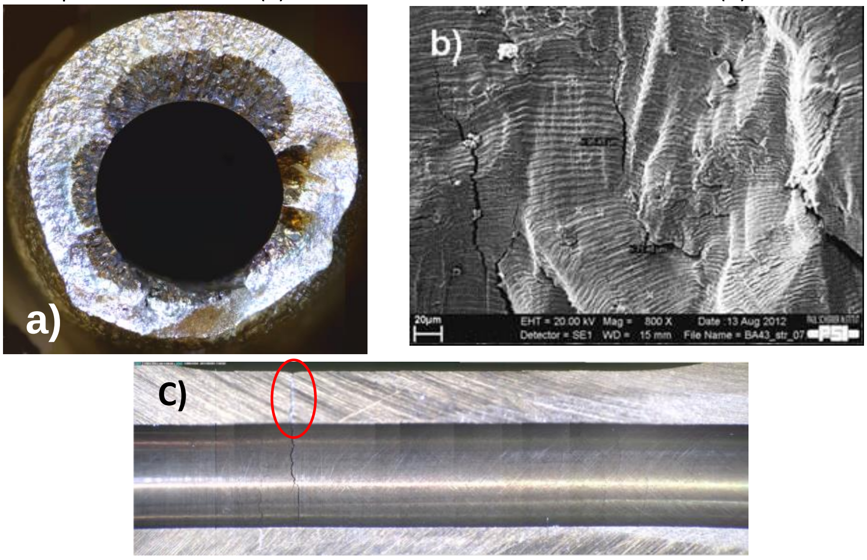


Figure 3: Facility for TMF initiation studies in LWR environments.

TMF & Isothermal LCF in Hydrogenated High-Temperature Water

Multiple crack initiation (a) and fracture surface with clear striations (b) from IP-TMF



Leak from isothermal LCF test at 220 °C (c)

Figure 4: TMF and isothermal LCF damage in hydrogenated high-temperature water.

Recent publications:

H.J. Leber, S. Ritter, and H.P. Seifert, "Thermo-Mechanical and Isothermal Low-Cycle Fatigue Behavior of 316l Stainless Steel in High-Temperature Water and Air", Corrosion, 2013, 69(11).

H.P. Seifert, S. Ritter, and H.J. Leber, "Corrosion Fatigue Initiation and Short Crack Growth Behaviour of Austenitic Stainless Steels under Light Water Reactor Conditions", Corrosion Science, 2012, 59, pp. 20-34.

H.P. Seifert, S. Ritter, and H.J. Leber, "Corrosion Fatigue Crack Growth Behaviour of Austenitic Stainless Steels under Light Water Reactor Conditions", Corrosion Science, 2012, 55, pp. 61-75.