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Results of German Investigations on Damage Due to Material Ageing and Corrosion

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1. Introduction

A solid basis of knowledge of the possible ageing mechanisms is necessary for preventive measures.

Two main investigation programes have been performed on behalf of Federal Office for Radiation Protection (Bundesamt für Strahlenschutz, BfS) to counteract material ageing.

The results of these programes will be reported in the following.

2. Report about German Investigation Projects

2.1 Report about Investigation Project SR 2501 Work Package

„Influence of the Reactor Coolant on the Fatigue Behaviour of Austenitic Stainless Steels"

Material fatigue is an ageing process which can occur in particular at highly stressed components like pipes, T-adapters and pressure vessel nozzels.

This material fatigue often results from thermal transients for instance caused by thermal shock, stratification or thermal striping.

The US Boiler and Pressure Vessel Code Section III and der German Nuclear Safety Standard Commission (KTA) provide fatigue curves.

They can be used to design components of nuclear power plants subjected to cyclic loading.

The current ASME Section III Code fatigue curves are derived from best-fit curves based on fatigue tests of small polished specimens in air at room temperature.

To account for effects on data scatter, size effects, surface finish and atmosphere, the data were lowered by a factor of 2 on stress and 20 on cyles but the design curves do not cover the corrosive influence of the reactor coolant environment.

Today it is generally accepted that the Light Water Reactor (LWR) environment has to be involved in cumulative fatigue life considerations.

The U.S. Nuclear Regulatory Commission has recently issued the Regulatory Guide 1.207 ”Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors” .

This Guide is based on the research work and publications by Argonne National Laboratory (ANL). The fatigue life prediction curves for austenitic stainless steels in air and LWR environment are shown in Fig. 1.

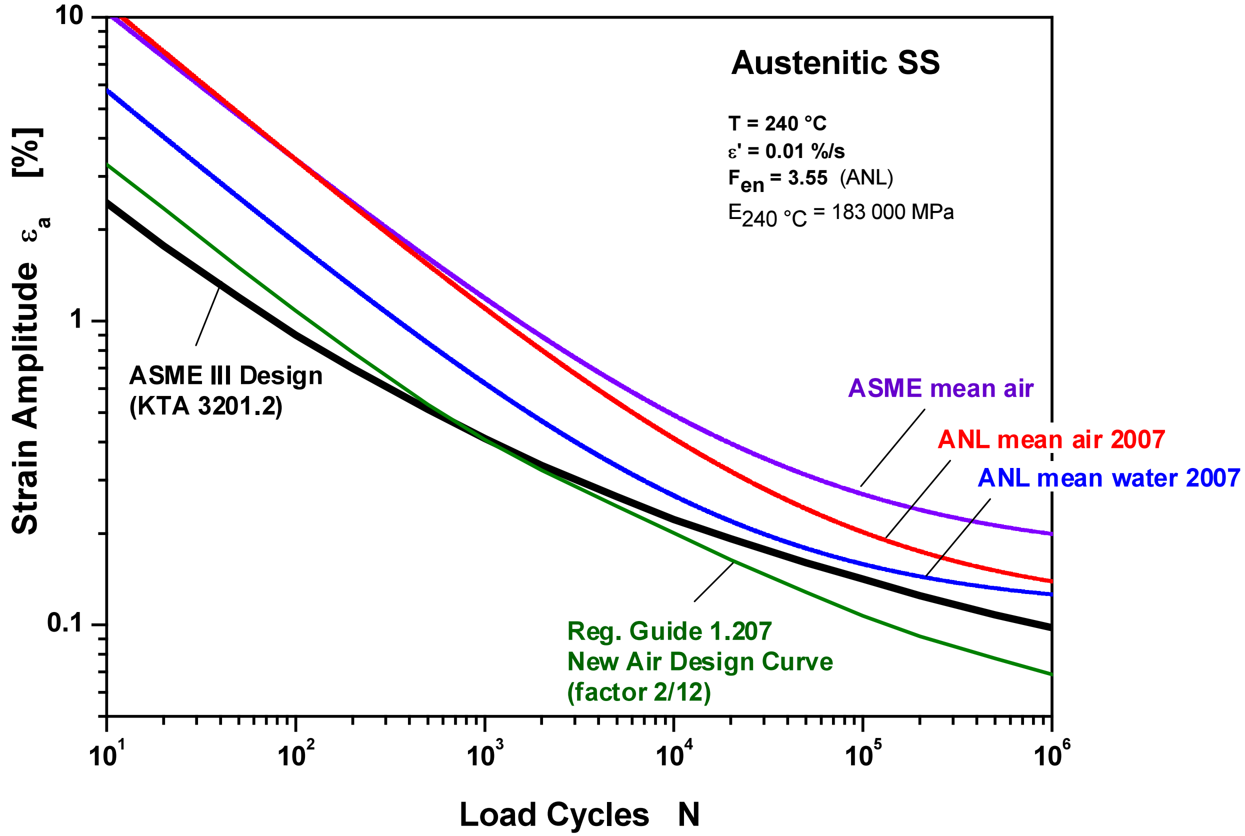

FIG. 1. Fatigue life curves for stainless steels

The ANL fatigue life model for LWR environments includes parameters for the effects of temperature, strain rate, dissolved oxygen content in water and, in case of ferritic steels, sulphur content of the steel.

In a second approach the environmental effects are expressed in terms of an environmental correction factor F_{en} as the ratio of fatigue life in a room temperature air environment to fatigue life in LWR coolant at operating temperature

$$F_{en} = N_{air} / N_{water} \qquad (1)$$

also depending on the above-mentioned parameters, which is used to estimate the environmental fatigue usage

$$U_{en} = \Sigma\, U_i \cdot F_{en\,i}. \qquad (2)$$

This procedure is conform with the methods proposed in Japan (i. e. Higuchi et al.) whereas the parameters to determine F_{en} are somewhat different in the Japanese approach.

In Regulatory Guide 1.207 a new fatigue design curve in air is developed, see Fig. 1, based on the new ANL mean air curve, lowered by a factor of 2 on stress and a reduced factor of 12 on cycles, whichever is more conservative.

Again this curve does not include safety margins and the environmental effects have to be considered on the basis of F_{en} as stated above.

The investigations of environmental effects on fatigue life of stainless steels performed in USA and Japan are predominantly based on unstabilized austenites.

The aim of the current project is to investigate whether the fatigue behaviour of the niobium or titanium stabilized austenitic stainless steels used in German nuclear power plants in contact with oxygenated high temperature water (boiling water reactor coolant) can be predicted by the curves developed in USA and Japan.

For details of the used autoclave system I refer to the fulltext version of the paper.

Table 1 Specification of the tested materials:

Test materials
Stainless steel 1.4550 Nb – stabilized (pipe, s = 35 mm) X 10 CrNiNb 18-9 C = 0.06 %
Stainless steel 1.4541 Ti – stabilized (pipe, s = 35 mm) X 10 CrNiTi 18-9 C = 0.06 % 1. as delivered (solution annealed) EPR-value = 0.3% 2. sensitized 620°C/8h EPR-value = 1.3 – 14 % (EPR: Electrochemical Potentiocinetic Reactivation)

Table 2 Test matrix of the LCF experiments:

work package	material	material state	ε_a [%]	environment	number of specimens
3.1.1	1.4541	as delivered	0,6 0,9 1,2	air 240 °C	3 x 2
3.1.2	1.4550	as delivered	0,6 0,9 1,2	air 240 °C	3 x 2
3.1.3	1.4541	as delivered	0,6 0,9 1,2	water 240 °C	3 x 3
3.1.4	1.4541	as delivered	0,6 0,9	sulphate 240 °C	2 x 2
3.1.5	1.4550	as delivered	0,6 0,9 1,2	water 240 °C	2 x 2
3.1.6	1.4541	sensitized	0,6 0,9	water 240 °C	2 x 2

ε_a = strain amplitude

Results: The fatigue life tested at 240 °C in air is quite similar for both austenitic stainless steels. The fatigue behaviour is in quite good accordance with the ANL mean air curve and better represented by this curve than by the ASME mean air curve, see Fig. 2.

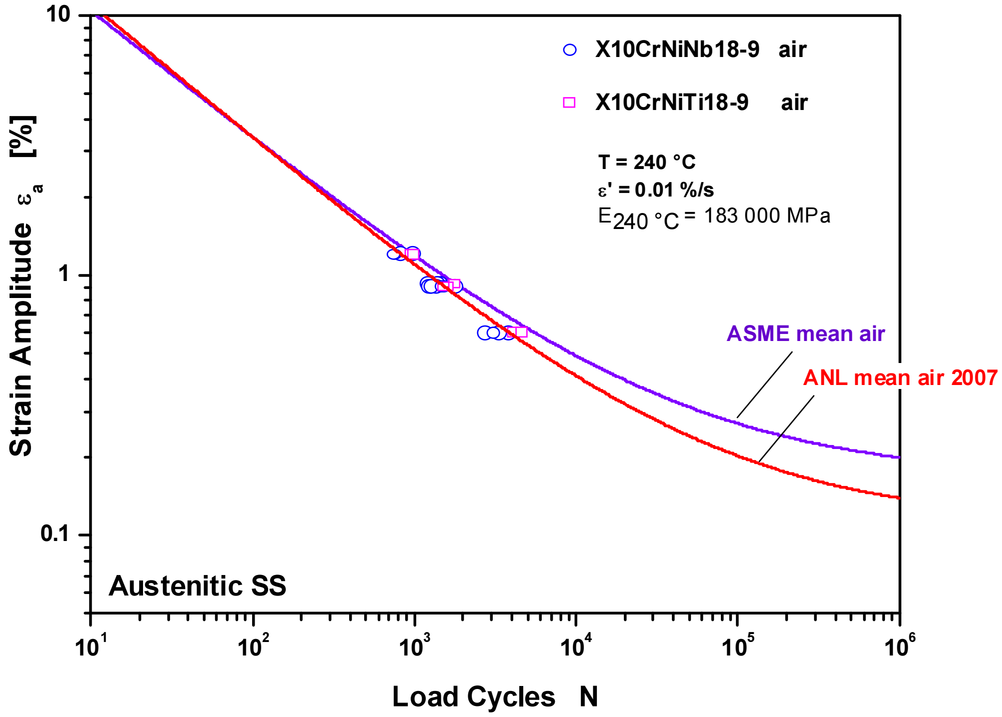

FIG. 2. Fatigue life of stabilized austenitic stainless steel in air compared with ASME and ANL mean air curves

The fatigue life in simulated BWR environment is conservatively covered by the ANL mean water curve, see Fig. 3. The influence of the simulated BWR environment (N_{air}/N_{water}) tends to decrease with increasing strain amplitude.

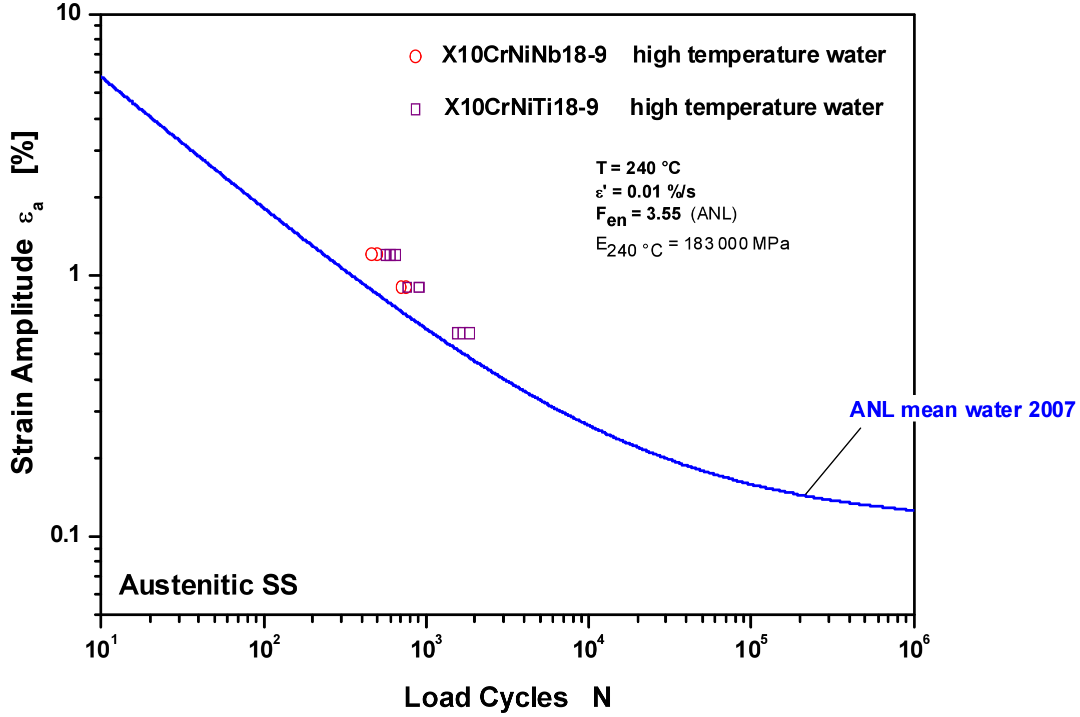

FIG. 3. Fatigue life of stabilized austenitic stainless steel in water compared with ANL mean water curve

The results so far suggest that the fatigue life of the stabilized austenitic stainless steels used in German nuclear power plants is similar to that of the unstabilized austenites of United States and Japan and can be reliably predicted by the existing methods.

2.2. Report about Investigation Project SR 2501 Work Package ”Improvement of ultrasonic testing of bi-metallic welds”

A further result of the project is the improvement of ultrasonic testing of bi-metallic welds by using a testblock with a realistic transverse flaw in the circumferential pipe weld.

The flaw was generated by loading of a half pipe after saw cutting of the test specimen (3-point bending specimen, load 60 tons = 600 kN) with corrosive chemistry on stressed inner surface.

Fig. 4 shows the bi-metallic test specimen, re-welded from half pipes A and B to original pipe shape. The flaw is a crack of about 6 mm depth, generated by genuine intergranular stress corrosion cracking (IGSCC), similar to the well-known SUMMER crack.

FIG. 4. Bi-metallic test specimen (OD 332 mm / 32 mm wall thickness)

Fig. 5 shows a radiography of half pipe A, showing the transverse crack at the inner surface of the bi-metallic weld (buttering and weld root area, crack size 20 mm length / 6 mm depth).

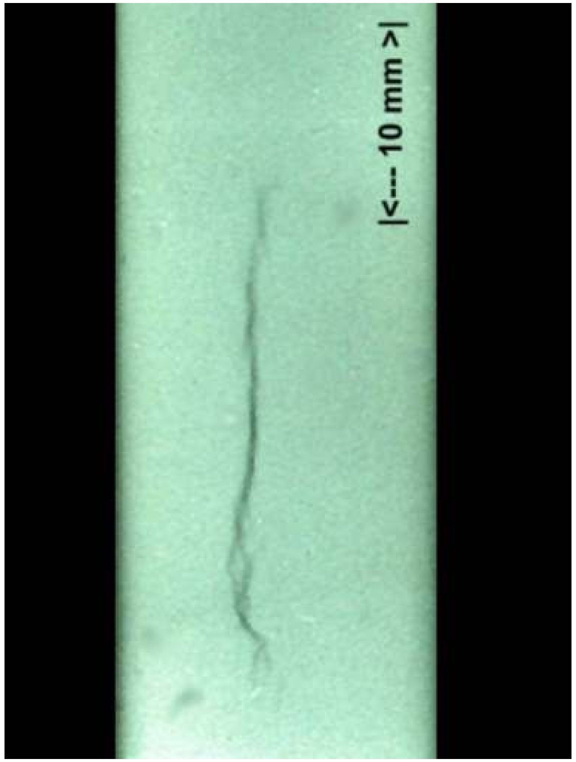

FIG. 5. Radiography of half pipe A

2.3 Report about Investigation Project SR 2521

„Analysis and Assessment of the Risk Potentials caused by Corrosion in German Light Water Reactor Plants“

Following the reports about events in nuclear facilities between 1995 and 2004 that were caused by corrosion, it was the purpose of this project to systematically analyze the risk potentials of the existing types of corrosion.

Excepted were strain-induced corrosion cracking, transgranular stress-corrosion, intergranular stress corrosion with sensitization and the corrosion of nickel-based alloys that have been observed and analyzed in considerable depth in other investigation projects.

The share of events due to corrosion in the total number of reportable events in German LWR plants amounted to approx. 12.5 %. This is the annual average over 10 years, as shown in Fig. 6.

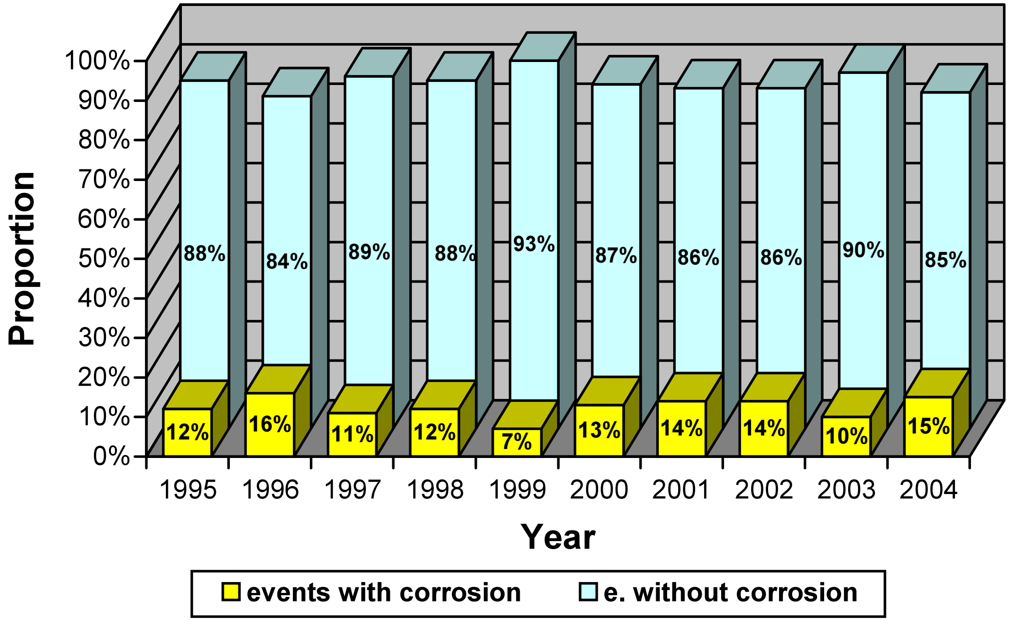

FIG. 6 Reportable events in German LWR Plants with and without corrosion

Some 90 % of the events caused by corrosion both in Germany and abroad were identified during scheduled maintenance, inspection or test measures, which include regular inspections.

The remaining events due to corrosion were identified by the internal monitoring equipment.

A deterministic assessment of systems and components affected by corrosion showed that for a summarized analysis it was impossible to exclude specific corrosion types from any further considerations merely for system-related reasons.

The assessment with probabilistic methods did not lead to any other results.

Fig. 7 shows the observed corrosion types in German nuclear power plants.

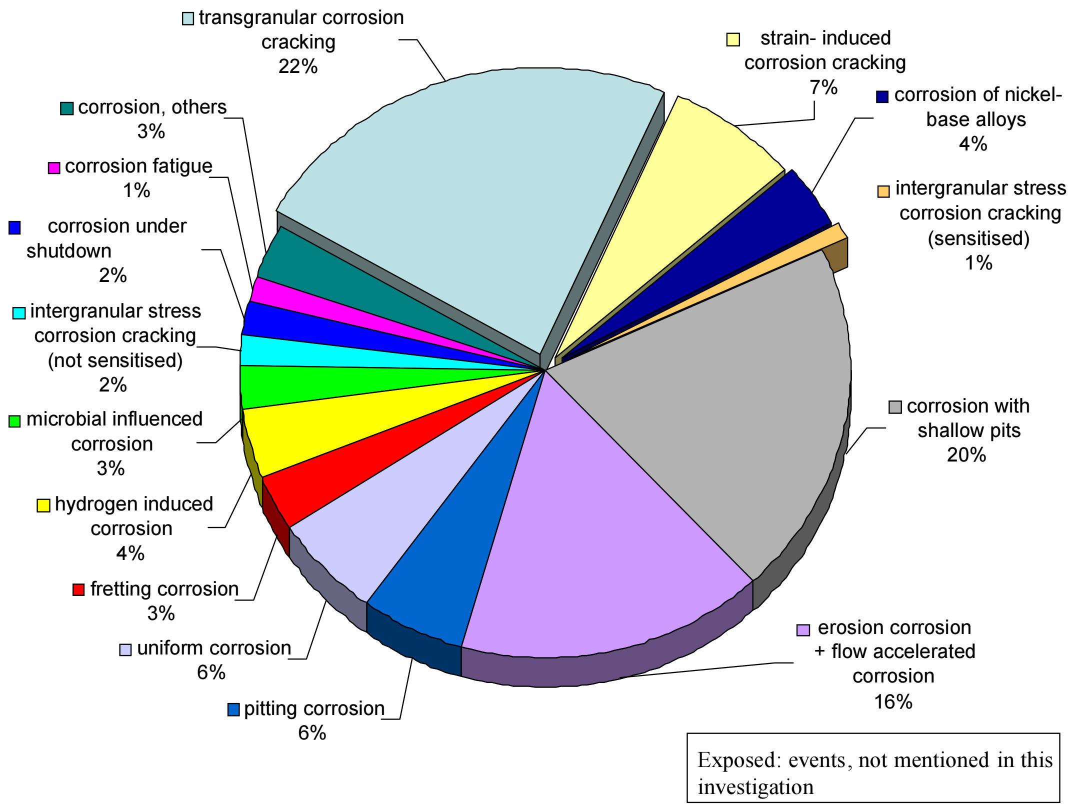

FIG. 7. Observed corrosion tpyes in German nuclear power plants

The following aspects were taken into account in the subsequent assessments:

- characteristic features, peripheral conditions and the propagation of the individual types of corrosion,
- possible influences of deterioration of the material caused by corrosion on the leak-before-break postulate,
- provisions in the German set of nuclear rules and regulations about preventive measures to be taken against corrosion,
- capability of non-destructive test methods and of visual inspections to recognize and identify damage caused by corrosion,
- status of the recurrent tests on pressurized components in the nuclear facilities
- internal monitoring measures for identifying the causes and consequences of damage due to corrosion.

The following conclusions have been derived for the German LWR plants from the results thus gained:

• Corrosion damage during operation can be largely prevented by a selective choice of the materials used, by well-defined manufacturing processes and by an optimized design of the safety-relevant components with regard to the stress they are subjected to.

• The fracture-mechanical assessments have confirmed that pipes dimensioned with regard to the leak-before-break criterion have to be free from systematic corrosion effects.

• The system of recurrent tests, internal monitoring as well as preventive inspections and maintenance will ensure the early identification of corrosion that may occur despite the preventive measures taken.

• The continuous assessment of damage that has occurred both in Germany and abroad will raise new aspects and deepen the knowledge, so that existing prevention efforts and monitoring measures can be further optimized.

• As a tendency less events caused by corrosion occured in plants of a more advanced design.

• Not thermally treated, cold-formed components made of stabilized austenitic steel are showing susceptibility to inter-granular stress corrosion. This components should therefore be proved in periodic reviews with greater intensity.

• The effects of medium on corrosion behaviour of not thermally treated, cold-formed components should be further analyzed.

3. Conclusions

The first results of fatigue experiments (in project SR 2501) show a good correlation between fatigue life of German nuclear power plant material and international fatigue mean curves. The experiments will be finished November 2007.

A good step forward to improve ultrasonic testing of bi-metallic welding was done with the build up of a new test specimen with realistic flaws.

As a result of assessing the events caused by corrosion (project SR 2521) and the peripheral conditions for their occurrence a recommendation for the periodic review of not thermally treated cold formed components was made.

Also, the susceptibility to inter-granular stress corrosion of not thermally treated, cold-formed components made of stabilized austenitic steel should be further analyzed.