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Test facilities and on-line instrumentation capabilities for core component materials investigations at the HALDEN reactor project

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Abstract

This paper describes experimental facilities at the Halden Reactor Project, dedicated to studying the behaviour of LWR core component materials in environments simulating those of operating nuclear power plants in terms of thermal hydraulic, neutronic and water chemistry conditions. The majority of the materials investigations make use of in-pile measurements. On-line monitoring techniques, such as the reversing dc potential drop method for crack propagation monitoring and the use of Linear Variable Differential Transformers (LVDTs) for crack initiation and creep and stress relaxation studies, are described and results from studies employing these instrumentation methods are presented.

The development of cracks due to the mechanism of irradiation assisted stress corrosion cracking (IASCC) is a process that affects the lifetime of nuclear power plants and there is a need both for the industry and safety authorities to have reliable materials data for use in safety assessments. IASCC of in-core components is a cause for concern for both BWRs and PWRs as reactors age, with components such as the core shroud and top guide in BWRs and the baffle former bolts in PWRs having experienced intergranular cracking attributed to IASCC. The main objective the crack growth studies that have been performed at Halden for a number of years are to generate long-term crack growth rate data for irradiated materials in typical LWR conditions. The effects of fluence, radiation hardening and applied stress intensity level on cracking are also addressed. Between four to six Compact Tension (CT) specimens, equipped with pressurised bellows for load application and instrumented for crack propagation monitoring with the reversing DC potential drop (DCPD) method, are accommodated in the test assemblies. The specimens are prepared from irradiated 304, 316 and 347 SS (with fluences ranging from $7 \times 10^{19}$ to $2.5 \times 10^{22}$ n/cm$^2$ (> 1 MeV)) taken from commercial reactor core components. Examples of crack growth rates measured in BWR (normal and hydrogen water chemistry (NWC and HWC)), and in PWR conditions are presented.

In addition to the crack growth investigations, a crack initiation (integrated time-to-failure) study is being conducted in a BWR loop system. The main objective of the investigation is to determine the number of specimen failures that occur in irradiated tensile stainless steel specimens as a function of the water chemistry (NWC versus HWC), with the aim of providing information on the effectiveness of hydrogen additions in reducing the susceptibility to the initiation of cracks in high dose material. A total of 30 miniature, irradiated tensile specimens, prepared from a 304 L SS control blade (fluence $8 \times 10^{21}$ n/cm$^2$)
are installed in the test assembly. Constant load (corresponding to 76-97% of the irradiated yield strength of the material) is applied to the specimens by means of bellows which place the specimens in tension. LVDTs are used to detect sample failures.

Another mechanism that may be detrimental or beneficial to core materials long-term performance is irradiation enhanced stress relaxation. Many bolts in reactor internals are stressed to high initial loads and when exposed to high temperature and radiation over a long time, the bolts may become loose due to stress relaxation. In a study that is currently under preparation, the effects of irradiation on the creep / stress relaxation in tensile specimens made from three common structural materials (solution annealed 304 and cold worked 316 SS and Alloy 718) will be evaluated. Load (stress) is applied to the specimens by means of bellows and constant displacement conditions are maintained by monitoring sample elongation with LVDTs and adjusting applied load on-line.

1. Introduction

The OECD Halden Reactor Project is a joint undertaking of organisations in 18 countries sponsoring a jointly financed programme under the auspices of the OECD Nuclear Energy Agency. The research and development work conducted at the Project reflects the needs of the nuclear industry and addresses a number of issues related to high performance fuel (in normal operating conditions and in response to transients), cladding corrosion, water chemistry, and the ageing and degradation phenomena of reactor vessel and internals materials.

The studies are performed in the Halden Boiling Water Reactor (HBWR), a test reactor with a maximum power of 20 MW that is cooled and moderated by boiling heavy water (normal operating temperature 235°C and pressure 34 bar). At any given time, around 30 test assemblies are in operation in the reactor, including studies on fuel behaviour and on materials performance, ageing and degradation issues. Typically, the reactor operates for two ~100-day reactor cycles each year. Depending on reactor position, fast fluxes in the range from $5 \times 10^{12}$ to $7 \times 10^{13}$ n/cm²s can be achieved. Typical fluxes are $3 \times 10^{13}$ n/cm²s, equivalent to an accumulated fluence of $6 \times 10^{20}$ n/cm² in a calendar year.

Many of the tests require representative light water reactor (LWR) conditions, which are achieved by housing the test rigs in pressure flasks that are positioned in fuel channels in the reactor and connected to dedicated water loops, in which boiling water reactor (BWR) or pressurised water reactor (PWR) conditions are simulated. An important aspect of the materials tests is the use of on-line instrumentation situated within the reactor core. For crack growth investigations, the reversing dc potential drop (DCPD) method is employed, while Linear Variable Differential Transformers (LVDTs), widely used for fuel performance investigations, have also been adopted for crack initiation and stress relaxation studies.

2. Irradiation Assisted Stress Corrosion Cracking Studies

The key objectives and priorities of the Irradiation Assisted Stress Corrosion Cracking (IASCC) test programme at Halden, which was initiated in 1991, are to i) predict behaviour, in particular the cracking response of irradiated materials; ii) assess possible countermeasures (e.g. changes in coolant chemistry, post irradiation annealing) and determine the limits of
operation for existing materials; iii) generate data that provide a fundamental mechanistic understanding of IASCC.

IASCC, the term used to describe the intergranular cracking of austenitic iron and nickel based alloys exposed to high temperature reactor coolant and irradiation environments, is widely recognised as one of the most important degradation phenomena affecting the long-term integrity of both BWRs and PWRs. Several reviews [1-4] have been published describing the factors influencing IASCC initiation and propagation processes, but despite extensive study over many years the mechanisms controlling the phenomenon are not yet fully understood.

Exposure to neutron irradiation results in changes in the microstructure, mechanical properties and microchemistry of the material, while ionising radiation modifies the coolant chemistry through radiolysis. The formation of small point defect clusters and faulted dislocation loops results in radiation hardening in the materials, with large increases in yield strength together with loss of ductility and fracture toughness. Radiation induced segregation results in the redistribution of major alloying and impurity elements at the grain boundaries (e.g. segregation of Ni and Si and depletion of Cr and Mo).

In BWR oxidising environments, for materials with low fluence, radiation hardening and Cr depletion are considered important factors in promoting IASCC. Decreasing the corrosion potential by the addition of hydrogen is effective in mitigating cracking. However, materials with higher fluence are found to be susceptible to IASCC also in reducing environments (such as BWR hydrogen water chemistry and PWR primary water) and while radiation hardening continues to play an important role, it is believed that other controlling mechanisms, still to be identified, also become important.

2.1 Crack Growth Studies

An important area of research in the field of IASCC is the determination of cracking behaviour in components where cracks already exist and assessing, for different fluence levels, the benefits of countermeasures such as the addition of hydrogen to the coolant (in BWRs). Also of importance is the possibility of gaining quantitative information on the rates of crack growth that can be expected in various core component materials, particularly as a function of varying stress levels and increasing fluence (where radiation induced segregation and radiation hardening play a significant role).

An essential requirement for the crack growth tests is the ability to monitor cracking response on-line in conjunction with the possibility for varying the load that is applied to the samples. Continuous crack monitoring allows the effects of changing chemistry environments (e.g. NWC vs. HWC) to be assessed directly, as well as enabling the contributions of loading on cracking response to be evaluated.

The geometry of the CTs that are used in the crack growth studies at Halden is shown in Fig. 1(a) and (b). The specimens, with width W=16 mm, and thickness B=5 mm, have 8 mm long machined chevron notches and 10% side grooves such that B_eff was 4.47 mm. Depending on material availability, either the entire specimen (including “arm” extensions for the attachment of wiring for the DCPD crack length measurements) is prepared from irradiated
material (Fig. 1(a)), or only the CT itself is prepared from irradiated material (Fig. 1(b)). In the latter case, the arm extensions, in the form of unirradiated material, are EB welded to the specimen. After machining, the CTs are fatigue pre-cracked in air and after installation in the test assemblies the leads for crack propagation monitoring are spot-welded to the specimen “arm” extensions. The specimens are instrumented with two pairs of potential leads and one pair of current leads.

Dynamic load was applied to the CTs by means of individually calibrated loading units which are equipped with bellows that are pressurised with helium gas through an outer system. During irradiation, the specimens may be subjected either to constant load or to cyclic loading conditions. The cyclic loading, with $R = 0.5$, $0.6$ or $0.7$ is implemented 1, 2 or 3 times every 24 hours. Typically, the duration of an unloading-reloading cycle is ~500 s.

Typically, four to six CTs may be accommodated in the crack growth test assemblies (Fig. 2). Of these, four specimens are located in a high $3\times10^{13}$ n/cm²s (> 1 MeV) fast flux while additional specimens are accommodated in the upper test section, where fast neutron flux is low. Examples of results obtained from two crack growth rate studies, one performed in BWR conditions and the other in PWR conditions appear below.

The BWR experiment was conducted over four ~100-day irradiation cycles. During the first three irradiation cycles, long-term crack growth rates were measured in oxidising conditions (~5 ppm O₂) and in the final cycle, the response of the specimens to the introduction of 2 ppm H₂ was evaluated.

Two of the CTs in the test matrix were prepared from Wurgassen NPP 347 SS top guide material with a fluence of ~1.5 x 10²¹ n/cm² (irradiated yield strength (YS) 948 MPa). One specimen was prepared from Oskarshamn 2 304 SS control blade handle material with a fluence of ~9 x 10²¹ n/cm² (irradiated YS 745 MPa), and the fourth CT was prepared from irradiated 316 NG with a fluence of 0.9 x 10²¹ n/cm² (irradiated YS 650 MPa).

An example of the crack growth rate data generated for the 316 NG SS specimen over one cycle of exposure to oxidising conditions appears in Fig. 3. Figs 4 and 5 show the response of the low fluence 316NG SS specimen and the high dose 304 SS specimen to the introduction of hydrogen. For the 316 NG specimen (Fig. 4), a clear reduction in growth rate was observed on shifting to low corrosion potential by increasing the hydrogen content.

For the 304 SS CT (Fig. 5), an apparent increase in crack growth rate accompanied the addition of hydrogen due to a “staircase” effect; i.e. each unloading/reloading cycle was accompanied by a step or “jump” in crack length as shown in detail in Fig. 6. Similar increases in crack length have also been observed in other investigations on materials with high yield strength due either to cold work or to irradiation hardening, and can occur in both oxidizing and reducing environments [5]. It is still unclear whether the crack increments are the result of real, rapid crack extension or due to the development of non-uniformities (uncracked ligaments) along the crack front, followed by breaking of these ligaments as load is increased (for example during the reloading part of a cycle). In contrast to the clear reduction in growth rate for the low fluence 316NG SS CT, the crack growth rate for the
304SS remained high even on switching to constant load, indicating that in some cases hydrogen may have limited benefit in reducing crack growth rates in very high dose materials.

On completion of the investigation, the fracture surfaces of the specimens were examined in a Scanning Electron Microscope (SEM). The fracture surfaces showed that transgranular cracking occurred during the fatigue pre-cracking, with intergranular cracking during in-pile testing under both constant and cyclic loading conditions. For all specimens the intergranular SCC had transitioned along the entire fatigue pre-crack front.

The PWR investigation was performed at a temperature of 335 °C (versus the 280 °C employed in the BWR study) and the test assembly was connected to a loop system operating with 2-3 ppm H₂, 2-3 ppm Li and 1000-1200 ppm B. Three of the CTs in the matrix were prepared from Chooz A centre filler assembly material with fluences of 1.2 and 2.5 x10²² n/cm² (irradiated YS 890 MPa), and the fourth specimen was prepared from the Oskarshamn 2 304 SS material that was also included in the BWR study. An example of the crack growth rate measured on the Oskarshamn 2 304 SS CT over one irradiation cycle is shown in Fig. 7.

As in the case of the BWR study, post-test SEM examination of the fracture surfaces showed transgranular fatigue pre-cracks while the environmentally assisted cracking was completely intergranular.

In Fig. 8, some of the crack growth rate versus stress intensity (K) data generated in the BWR investigation (in oxidising conditions) and in the PWR study are summarised. All the CTs show similar crack growth dependency on increasing K level. The crack growth rates recorded for the irradiated CTs in O₂ are ~4-5X higher than the disposition curve for sensitised 304 SS in 8 ppm O₂ (NUREG 0313) [6], while the crack growth rates measured on the CTs in the PWR study are comparable to those of the unirradiated sensitised 304 SS.

In Fig. 9 the crack growth rates measured in the irradiated CTs as a function of yield strength are compared with the crack growth rates measured in unirradiated specimens with increased yield strength due to cold work [7,8]. Increases in yield strength due either to cold work or radiation hardening result in elevated crack growth rates [7,8] both at high and low corrosion potential. The growth rates measured for the irradiated materials in 5 ppm O₂ were higher than for the solution annealed, cold worked-only materials, as would be expected due to the combined contributions of radiation hardening and radiation induced segregation (in particular Cr depletion). In reducing environments where the contribution of radiation-induced segregation is limited and radiation hardening is the primary factor in producing enhanced crack growth, the crack growth rates measured in the PWR study are comparable to those recorded for the unirradiated specimens.

### 2.2 Crack Initiation (Integrated Time-To-Failure) Studies

While most of the IASCC investigations carried out at Halden have concentrated on measuring crack growth rates, the initiation of cracks in representative core component materials has also been investigated in two separate in-pile studies employing two different specimen geometries.
Given the stochastic nature of initiation processes, critical features considered essential to obtaining useful initiation data include an adequate number of repeat specimens, active loading, and continuous monitoring of individual specimens. To this end use was made, in both the studies, of LVDTs for continuous monitoring of specimen performance, and efforts were made to maximise the number of specimens that were incorporated in the test matrices.

In the first initiation study, the effects were addressed of accumulated fluence, stress level and material type on the initiation of cracks in pressurised tube specimens exposed to BWR operating conditions.

Thirty-four thin-walled unirradiated tube specimens (Fig. 10), pressurised with argon gas to different levels of hoop stress, were included in the test matrix. Selected tubes were equipped with external gas lines which enabled on-line variation of the stress level while the remainder were pre-pressurised prior to installation in the reactor. Twenty four of the tubes, with stress levels ranging from 1.2 to 2.75 $\sigma_y$ (of the unirradiated material), were prepared from sensitised 304 stainless steel, thereby enabling the separate effect of stress on the initiation of cracks to be evaluated. The remaining ten tubes were prepared from solution annealed 304 and 316 L and cold worked 347 stainless steel. In the case of these specimens, the primary objective is to compare the behaviour of the different materials as affected by dose accumulation, and to this end, all the tubes are pressurised to the same level (1.2 $\sigma_y$). The thirty-four tubes were arranged in six strings, and each string was equipped with an LVDT for on-line monitoring of total string length. Three of the strings were located in the high fast neutron flux region of the facility and the remainder were installed in a low flux position (Fig. 10).

During irradiation, the specimens were exposed to a BWR operating temperature of 288 °C. In order to encourage tube rupture, a comparatively aggressive coolant environment was created by operating the loop with an inlet oxygen content of 3 ppm and an inlet solution conductivity of 0.5 $\mu$S/cm (increased by the addition of $\text{H}_2\text{SO}_4$ to the feed-water). As described above, tube integrity was monitored by means of the LVDTs attached to the lower end of each specimen string and also by means of a gamma monitor installed in the outer loop system. In the event of crack initiation and subsequent propagation as a through-wall crack, the monitor was activated by the release of Ar-41 from the tube and into the coolant. An example of the on-line signal changes accompanying the rupture of a tube is illustrated in Fig. 11. The rupture is clearly detected by increases in the coolant activity levels and by corresponding changes in the LVDT signals. In post irradiation examination of the fracture surfaces of various specimens from this study, clear intergranular stress corrosion cracks, which initiated on the outer surface of the specimens, have been observed.

In a second crack initiation investigation, which is currently in progress, the main objective is to record the number of specimen failures that occur in irradiated austenitic stainless steel tensile specimens as a function of BWR water chemistry (NWC versus HWC), with the aim of providing information on the effectiveness of hydrogen additions in reducing the susceptibility to the initiation of cracks in high dose material.
A total of 60 irradiated miniature tensile test specimens, 30 to be exposed to NWC conditions and 30 to be exposed to HWC conditions were prepared for the investigation. The specimens, which have a total length of 20mm and a cylindrical (1 mm diameter, 4 mm long) gauge section, were all prepared from a 8 x 10²¹ n/cm² 304L SS control rod material from the Barsebäck 1 BWR. The fluence of this material is close to the maximum end of life fluence of BWR components such as the top guide and, because of the high dose, is more likely to exhibit susceptibility to stress corrosion cracking. In addition, the material will have reached saturation in terms of mechanical and microstructural property changes and is expected to behave in a more consistent manner than lower dose materials where saturation has not yet been reached.

Thirty of the specimens are currently being exposed to the NWC conditions. The specimens are arranged in pairs (Fig. 12) and constant load is applied by means of bellows that are compressed by the system pressure, thereby placing the specimens in tension. The load level is determined by a combination of bellows pre-fill pressure, the system pressure and the operating temperature. Constant load, producing stresses corresponding to 76 to 97 % of the irradiated yield strength (718.5 MPa) of the material has been applied to the specimens.

Each pair of specimens is equipped with an LVDT that enables on-line detection of failure, and the specimens within each pair are identified by means of spacers placed internally in the bellows. In the event of specimen failure, the bellows collapse, with the extent of movement being recorded by the LVDT, which enables identification, on-line, of the failed sample. A typical example appears in Fig. 13.

In total five failures have been detected during ~10000 hours of in-reactor exposure. The failures occurred after times ranging from ~550 to 8000 hours, at stress levels between 77% - 93 % of the irradiated yield stress (Fig.14). While these results are in contradiction to those reported by Jacobs et al [9], in which irradiated (0.12-3 x 10²¹ n/cm²) tensile specimens, with applied stresses ranging from 3 to 90% of yield strength, typically failed within several hundred hours of testing, they are in part supported by observations from another uniaxial constant load study [10], where irradiated (0.5-1x10²¹ n/cm²) specimens with 75-85% of yield stress either failed early or did not fail during the ~2000 hour test period.

Operation for an additional two irradiation cycles under oxygenated conditions is planned, after which the second set of thirty replacement specimens will be exposed to hydrogen chemistry for a similar period of time.

3. Stress Relaxation and Creep Investigations

Many bolts in reactor internals are stressed to high initial cold pre-loads, which when subjected to high operating temperatures and irradiation over time, may result in loosening of the bolts and the pre-load can be lost. In a study that is currently under preparation, the effects of irradiation on the creep / stress relaxation in austenitic stainless steels commonly employed in commercial PWR reactors is to be evaluated.

As for the integrated time-to-failure study, use will be made of small tensile specimens (2.5 mm diameter and a gauge length of ~50 mm), which are prepared from unirradiated, solution annealed 304 and cold worked 316 stainless steel and Alloy 718 (see Table below).
Table 1 Test Matrix Stress Relaxation / Creep Investigation

<table>
<thead>
<tr>
<th>Material</th>
<th>No.</th>
<th>Temp. (°C)</th>
<th>Stress (MPa)</th>
<th>Dose (dpa)</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Instrumented specimens</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CW 316</td>
<td>1+2*</td>
<td>330</td>
<td>345</td>
<td>2.0</td>
<td>Replacement baffle bolt + split pin material</td>
</tr>
<tr>
<td>CW 316</td>
<td>1</td>
<td>330</td>
<td>275</td>
<td>2.0</td>
<td>*2 specimens to be operated in creep mode</td>
</tr>
<tr>
<td>CW 316</td>
<td>1</td>
<td>330</td>
<td>205</td>
<td>2.0</td>
<td>Qualification sample</td>
</tr>
<tr>
<td>CW 316</td>
<td>1</td>
<td>330</td>
<td>--</td>
<td>2.0</td>
<td></td>
</tr>
<tr>
<td>CW 316 LN</td>
<td>1</td>
<td>330</td>
<td>345</td>
<td>2.0</td>
<td>Low irradiation creep material</td>
</tr>
<tr>
<td>CW 316 N lot</td>
<td>1</td>
<td>370</td>
<td>345</td>
<td>2.0</td>
<td>EBR II irradiation creep test archive</td>
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<tr>
<td>SA 304L</td>
<td>1</td>
<td>290</td>
<td>90</td>
<td>2.0</td>
<td>EBR II irradiation creep test archive</td>
</tr>
<tr>
<td>SA 304L</td>
<td>1</td>
<td>290</td>
<td>72</td>
<td>2.0</td>
<td></td>
</tr>
<tr>
<td>Alloy 718</td>
<td>2</td>
<td>330</td>
<td>345</td>
<td>2.0</td>
<td>PWR irradiation stress relaxation data</td>
</tr>
<tr>
<td><strong>Uninstrumented specimens</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>CW 316</td>
<td>6</td>
<td>330</td>
<td>275</td>
<td>0.4, 1.6, 2.0</td>
<td>Second phase ppt densification data</td>
</tr>
<tr>
<td>SA 304</td>
<td>6</td>
<td>290</td>
<td>90</td>
<td>0.4, 1.6, 2.0</td>
<td>Baffle former plate material</td>
</tr>
<tr>
<td>Alloy 718</td>
<td>6</td>
<td>330</td>
<td>345</td>
<td>0.4, 1.6, 2.0</td>
<td>Second phase ppt densification data</td>
</tr>
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</table>

The specimens will be subject to constant displacement conditions during irradiation in an inert environment to fluences of ~0.25, 1 and 1.4 x 10^21 n cm^{-2} (> 1 MeV). Load (stress) is applied to the specimens via bellows that are compressed by gas pressure that is introduced into the chamber housing the bellows (Figure 15). Constant displacement of the tensile specimens is maintained by monitoring sample elongation with LVDTs and adjusting (reducing) the applied load (stress) on the specimens on-line, by decreasing the pressure in the bellows housing units. Selected specimens will be operated in creep mode by maintaining constant load on the specimens during irradiation. In addition to the bellows gas lines, the test units are equipped with gas lines that enable the specimen temperature to be varied in the range from 240 to 400°C, by altering the composition of helium-argon gas mixture surrounding the specimens. In order to reduce the amount of scatter in the data, the number of samples in the matrix has been maximised, incorporating 30 tensile specimens in total (12 will be instrumented as illustrated in Figure 15, and 18 will be un-instrumented and subject to post irradiation measurements).

4. Summary

The objectives of the materials testing programme at the Halden Project are to improve the understanding of materials ageing and degradation processes as well as to practically demonstrate methods designed to increase component lifetime. The focus is on in-reactor experiments addressing core and vessel materials and, to this end, a range of different studies are being conducted.

The major focus is on the generation crack growth rate data for irradiated CT specimens, fabricated from commercial reactor component materials. The effects of stress intensity, irradiated yield strength and operating conditions (BWR and PWR) on in-core cracking behaviour are addressed.

The crack growth studies are complemented by studies on the effects of fluence, material type and stress level on the initiation of cracks in thin-walled pressurised tube specimens and on
the benefits of HWC in suppressing crack initiation in high dose tensile specimens subjected to constant load.

A third area of study addresses stress relaxation and creep behaviour of cold worked 316 SS, 304 SS and Alloy 718 as a function of stress, temperature and fluence.

References


Fig. 1 (a) CT specimen including arm extensions prepared from irradiated material and (b) irradiated CT specimen with unirradiated arm extensions.
Fig. 2 Example of test assembly for crack growth rate investigations. Four instrumented CT specimens may be accommodated in the high flux region, which is surrounded by high enrichment fuel rods (booster rods). A further two instrumented specimens may be accommodated in the upper test section, where fast flux is low.
Fig. 3 Example of crack growth rates measured on a CT specimen prepared from irradiated 316 NG stainless steel during exposure to BWR operating conditions (280 C) with high (5 ppm) O₂

Fig. 4 Response of 316 NG SS CT (initial fluence 0.9 x10²¹ n/cm²) to the addition of hydrogen. A clear reduction in growth rate is apparent.
Fig. 5 Response of 304 SS CT (initial fluence $9 \times 10^{21} \text{n/cm}^2$) to the addition of hydrogen. There is no significant change in growth rate for the specimen.

Fig. 6 Detail from Fig 5, showing response of specimen to unloading / reloading cycles. Each cycle is accompanied by a step-like increment in crack length.
Fig. 7 Example of crack growth rates measured on a CT specimen prepared from irradiated 304 stainless steel during exposure to PWR operating conditions (335°C) with the addition of Li and B and ~3 ppm H₂.

Fig. 8 Crack growth rates measured on irradiated CT specimens in BWR oxidizing and PWR reducing environments compared with crack growth rates measured on unirradiated sensitised 304 SS in high O₂ (NUREG-0313).
Fig. 9 Comparison of the effect of yield strength on crack growth rates measured in oxidizing environments on unirradiated cold worked stainless steels 5,7,8 and on irradiated CTs exposed to BWR oxidizing and PWR reducing environments in Halden experiments.
Fig. 10 Arrangement of pressurised tube specimens in in-core crack initiation test
Fig. 11 Example of change in LVDT signal and increase in gamma activity levels accompanying the rupture of a pressurised tube specimen.

Figure 12. Schematic illustrating on-line monitoring technique used in crack initiation (integrated time-to-failure) study on irradiated tensile specimens
Figure 13. Irradiated tensile specimen failure recorded on-line in crack initiation / integrated time-to-failure study

Fig. 14 Summary of specimen failures recorded on-line in time to failure investigation. The time to failure and the stress levels on the specimens are indicated in the figure.
Figure 15. Principle of technique employed for irradiation creep/stress relaxation studies