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NRI experimental facility for the testing of irradiation assisted stress corrosion cracking

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Summary

IASCC influencing reactor internals of both BWR and PWR reactors is a complex phenomenon covering influences of material structure, neutron fluence, neutron flux, chemistry of environment, gamma radiation and mechanical stress. To evaluate such degradation, tests should be performed under conditions similar to those in real structure. Nuclear Research Institute has built several experimental facilities in order to be able to test IASCC degradation of materials. Basically, reactor water loops, both PWR and BWR, could be used to model environmental conditions including gamma and neutron irradiation. Pre-irradiation can be done in irradiation channels under well controlled temperature conditions. During the experiment, in-pile conditions can be compared with those out of pile. It enables to clarify pure influence of irradiation. For testing of irradiated specimens, hot cell facility has been developed for slow strain rate tests. The paper will show all above mentioned facilities as well as some of results observed with them.

1. Introduction

Irradiation assisted stress corrosion cracking is an complex phenomenon covering simultaneous influence of neutron irradiation, gamma irradiation, coolant environment, stress and deformation state and material microstructure on the sensitivity to intergranullar cracking. Except of that, high irradiation doses could lead to substantial changes of mechanical properties due to irradiation hardening [1].

To describe such changes, specific tests should be performed with material irradiated to the fluences which are similar than in operating power reactor. Another influence, the impact of radiation on the water chemistry and possibly local processes in the material, could be modeled only if such radiation is present during the experiment. Usage of in pile conditions is a necessity in such cases.

Between the experiments which could describe response of materials on the reactor core conditions, long term exposures are dominant. They are represented by constant load or constant displacement type of tests or, in some more accelerated manner low displacement rate tests. Among them, slow strain rate test (SSRT) of tensile specimens and rise displacement test (RDT) of compact tension specimens should be mentioned.

Nuclear Research Institute Řež, plc. is running the research reactor LVR-15 which is equipped with the water loops, loading facilities and irradiation channels.

Research reactor at NRI is under operation since 1962. Since the half of eighties, PWR water loop is working in both active and comparative channels of reactor. In early nineties, first work was started in connection with the design of testing facility for IASCC type of testing. This work was based on the experiences with the stress corrosion cracking testing with non-irradiated specimens. In 1994, in pile slow strain rate testing facility was introduced. The first in pile experiment was performed in 1995 in the framework of WWER 1000 reactor pressure vessel and internals testing program. In 1995, the first BWR water loop was started followed by the second one in 1997. In the same year, the loading facility for the in pile testing of up to 2T CT specimens was installed.

At present, the complete matrix of irradiation experiments with materials is being performed in order to separate particular influences:

- in pile testing of pre-irradiated specimens

- in pile testing of non-irradiated specimens
- out of pile testing of irradiated specimens
- out of pile testing of non-irradiated specimens.

The goal of the paper is presenting these facilities as well as to showing as an illustration some results achieved with them.

2. Research reactor and water loops

Reactor LVR 15 is light water moderated and cooled tank nuclear reactor with forced cooling. The maximum reactor power is 10 MW. In the reactor core, there are vertical channels for both pre-irradiation of materials and material testing as well as 9 horizontal channels. Some of characteristics of LVR-15 reactor are listed below:

maximum reactor power	10 MW
max. thermal neutron flux	$1.5x10^{18}n/m^2s$
max. fast neutron flux	$3x10^{18}n/m^2s$
irradiation channel in the fuel	$1x10^{18}n/m^2s$
irradiation channel in reflector	$3\text{-}5x10^{18}n/m^2s$.

The water loop RVS-3, Fig. 1, is a PWR / WWER environment conditions modeling facility with one in pile channel, one comparative channel and an out of pile test section. Loading facility for testing corrosion fatigue under out of pile conditions is a part of the system. The design pressure is 16.5 MPa, design temperature is 345 °C. Maximum water flow is 10 tons / hour. Neutron flux in the in pile channel is approximately $1x10^{18}$ n/m^2s (E>0.5 MeV). Water environment is prepared in the make up and dosing water systems. A sampling system allows to sample water environment during the experiment. Between monitoring parameters, conductivity measurement, high temperature pH and electrochemical potential are the most important examples.

The boiling water loop is designed for the testing of materials properties and radioactivity transport under BWR conditions (Fig.2). The highest design parameters are 12 MPa of pressure, temperature of 300 °C, flow rate 8 tons/hr and neutron flux in the in pile channel $1x10^{18}$ n/m^2s. The generation channel creates hydrogen peroxide which is then forced to the in pile testing channel. Out of pile comparative channel allows a comparison between in pile and out of pile conditions under the same circuit water chemistry.

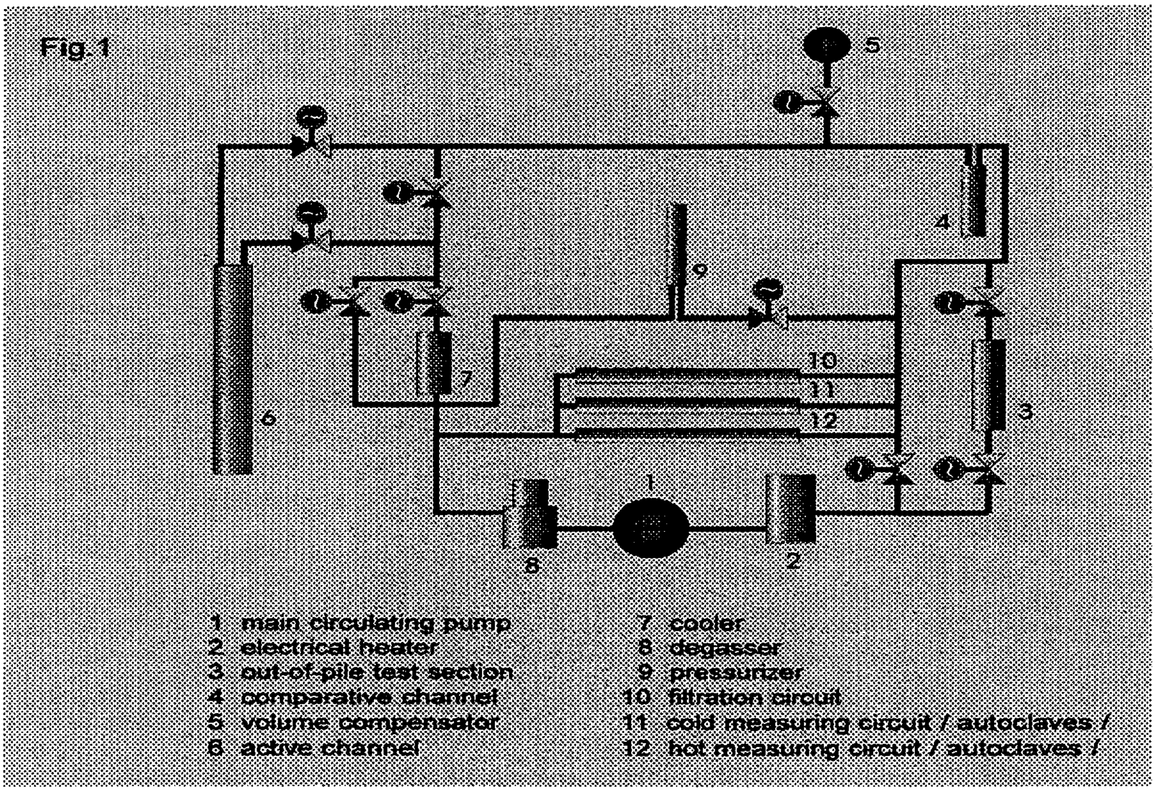


Fig.1: Pressure water loop RVS-3

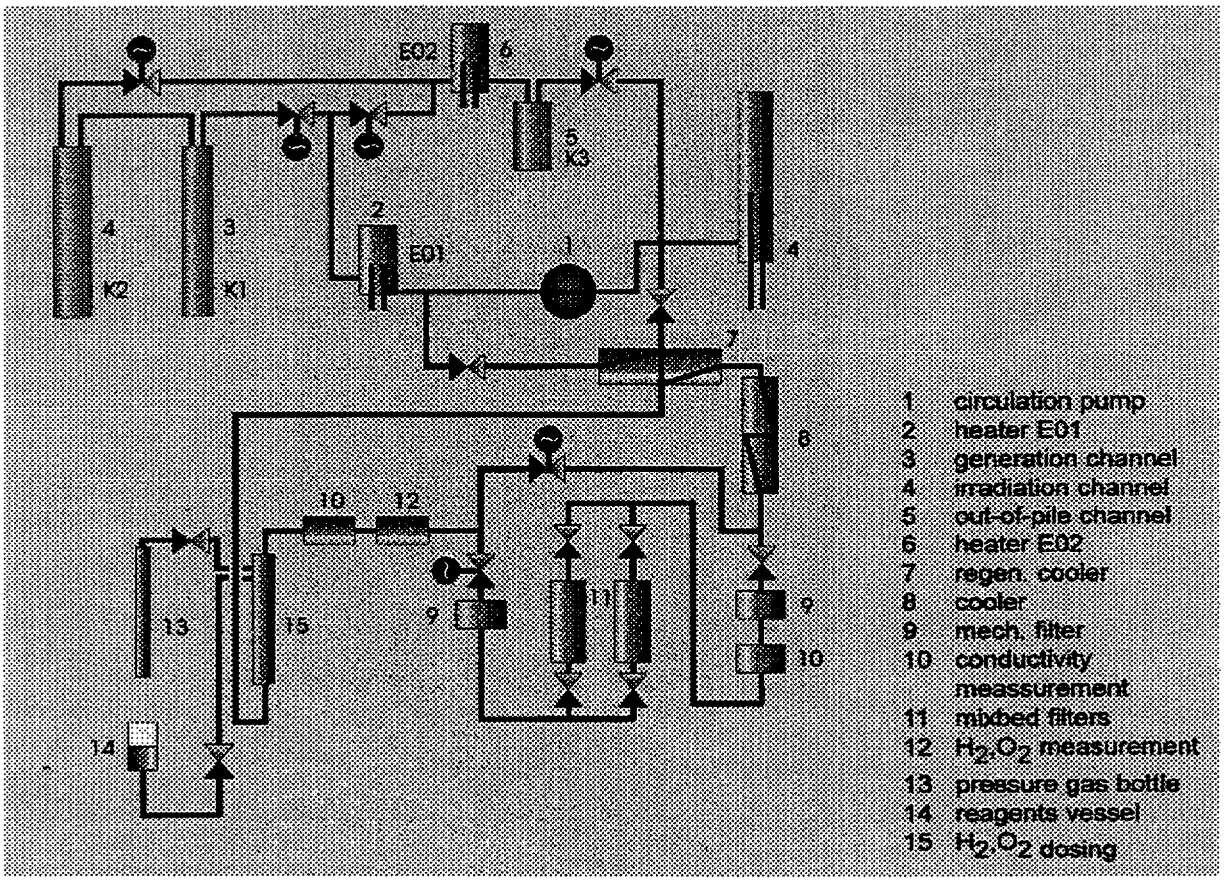


Fig.2: Boiling water loop

3. Pre-irradiation of structural materials

Pre-irradiation of test specimens is carried out in the irradiation channels. Two types of rigs are used. Chouca rig is used for the irradiation of tensile and Charpy specimens, mostly ferritic, for the programs of reactor pressure vessel testing. NRI has developed irradiation rigs for compact tension specimens irradiation. The specimens of sizes up to 50 mm thickness are irradiated under controlled conditions of temperature and environment. The irradiation is performed in helium. An example of irradiation rig for irradiation of four CT specimens is in Fig. 3. Irradiation temperatures are up to 320 °C, the flux is up to $1x10^{18}$ n/m^2s.

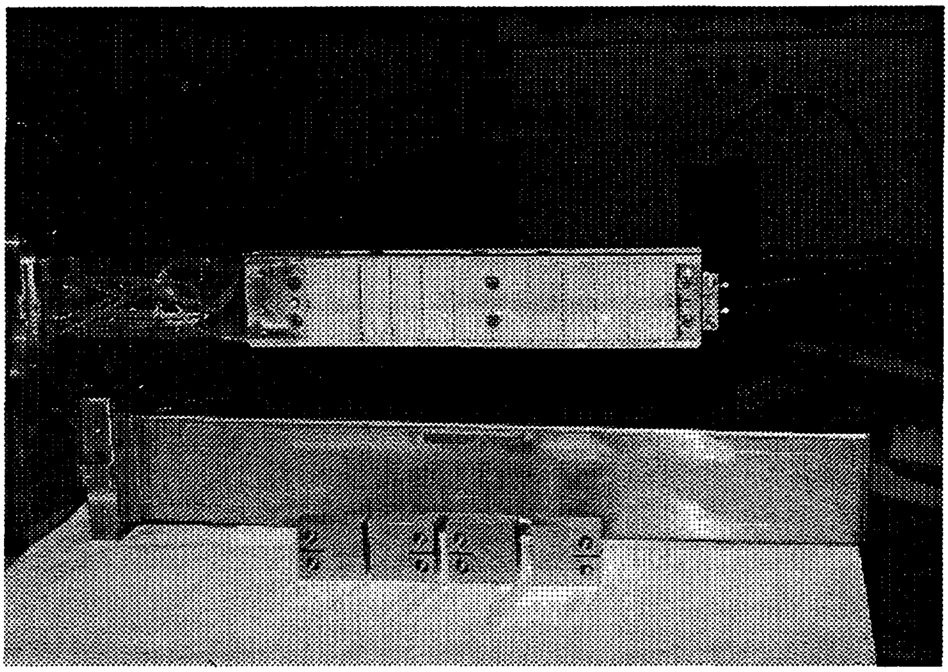
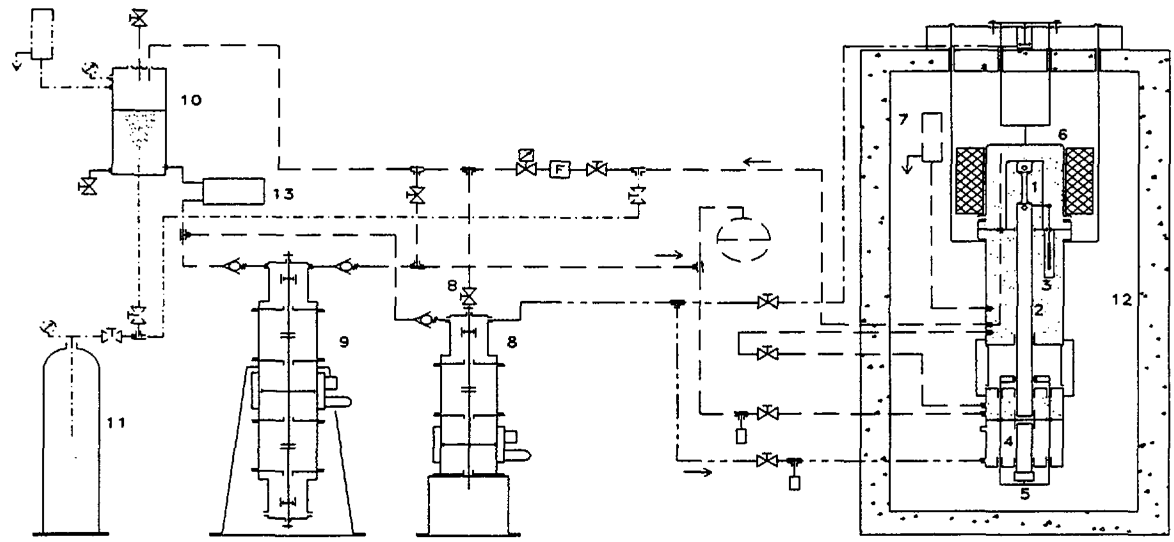


Fig.3: Irradiation rig for four CT specimens

4. Out of pile testing

Slow strain rate tests with irradiated specimens are performed in both PWR and BWR environments in the autoclave with active loading system and recirculation water loop, see Fig.4. While loading facility and autoclave are placed in the hot cell, the water loop pump, water chemistry plant and control and regulation system is placed outside the hot cell.



1 – specimen
2 – pull rod
3 – transducer of displacement
4 – chamber of the loading medium
5 – dynamometer
6 – autoklave lid
7 – safety valve
8 – pressurizing unit
9 – water loop system
10 – storage tank
11 – gas bottle (H, Ar)
12 – shielding of the semi-hot cell
13 – oxymeter

Fig. 4: Hot cell SSRT testing system

The hot cell is connected with other hot cells with the transportation system. For the necessary cleaning, grinding, manipulation, hydrogen charging and other supporting activities, the shielded box connected with the main hot cell was constructed. The data acquiring system is also part of the system.

The facility was designed for loading of tensile specimens of small diameter, typically 2-4 mm, at extremely low deformation rates, down to $2x10^{-8}$ /s in the high pressure, high temperature environment.

The autoclave is connected with the recirculation water loop, see Fig. 4. The specimen is fixed by pins to the grips. The grips are electrically insulated from the rest of autoclave by ceramic insulation on the interfaces. The LVDT transducer Hottinger WSS-L-010W is used. The transducer is fixed between the lower grip and the autoclave body. This way, high stiffness was reached. The lower grip is connected to the pull rod.

The autoclave is heated by a furnace with the electronic control system. The maximum temperature is 330°C.

The water loop is of the recirculation, refreshed type. The water flow of 0.125 l/hour is made by the water loop pressure pump. To achieve stable pressure in the loop, the membrane autoclave pressurizer is included into the system. The flow rate is controlled by a control valve with very fine adjustment. A microfilter is located upstream of it to keep its proper function. The water flows to the lower chamber and to the autoclave. This way, the impact of the loop pressures to the pull rod are balanced and no change of pressure in the loop can cause any unexpected shift of the specimen. The loop itself contains also a storage vessel with the volume of 11 liters.

The oxygen content in the water is measured by an oxymeter. The range of measurement is 10 - 10000 ppb. The oxymeter is placed on the input part of the high pressure pump.

All tubing of the loop is made of stainless steel, SWAGELOCK connections and valves have been used.

The hot cell GOLEM is a three-chamber facility. The input chamber serves as a check point for the specimen delivery from the transportation system connecting the GOLEM with other hot cells. The loading facility is located in the middle chamber. The third one is used as an interface between the GOLEM and an auxiliary shielded box. The total activity which can be stored in the hot cell is 37.6 GBq.

Another possibility for out of pile testing of irradiated specimen represents in pile channel used without reactor running. Out of pile experiments with non-irradiated specimens can be done in non active autoclave laboratories or in the out of pile PWR and BWR loops channels.

5. In pile testing

The facility for the SSRTs in the active channel was developed originally in 1994. The facility enables testing of specimens with diameter up to 3 mm and length up to 18 mm. The specimen is loaded by monotony tension at a defined deformation rate which can be as low as $2x10^{-8}$ /s.

The scheme of the loading facility is in Fig. 5, photograph in Fig. 6. The facility consists of a reactor irradiation channel with inserted Field tube, inner loading frame, outer loading unit, control unit and data acquisition system.

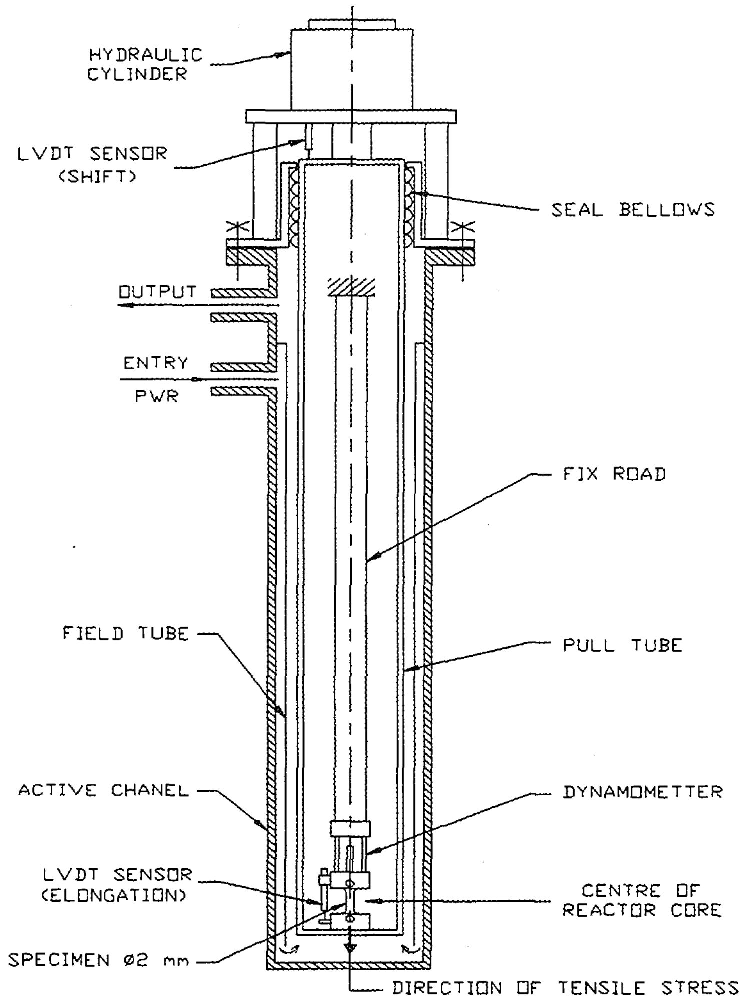


Fig. 5: Scheme of in-pile loading facility

The inner loading frame is one compact unit of the length of 4500 mm loaded into the active channel by the crane. The specimen is fixed to the loading frame in the position of maximum neutron flux in the reactor. Parallel to the specimen, the LVDT HOTTINGER transducer is placed to measure the displacement of the specimen as well as to control the deformation rate. The transducer is of the ± 5 mm type. The accuracy guaranteed by the producer is ± 0.2%. Above the specimen, the load cell is placed. The sensitivity of measurement of the load is ± 1 %. The pull rod is fixed to the upper deck of the channel. There is another LVDT transducer to check the total displacement of the system. The fixation of the specimen is made through electrically insulated pins.

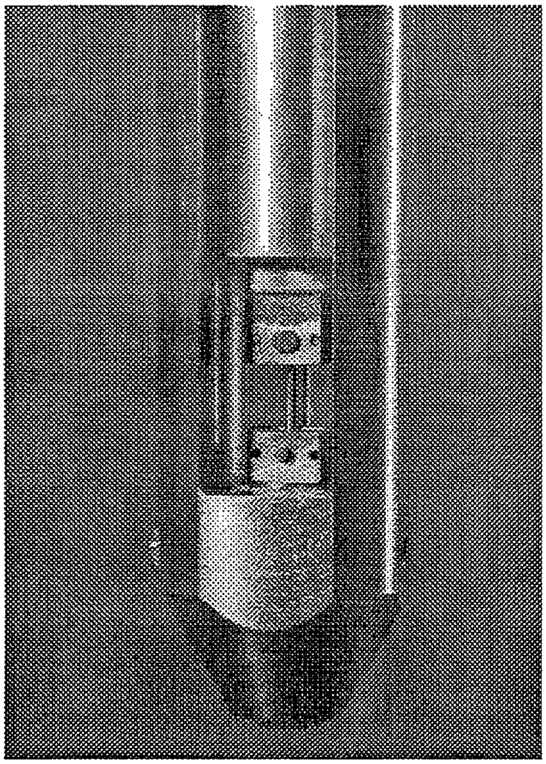


Fig. 6: Detail of in pile loading facility

The loading is made by pressurizing the thick-wall tube fixed to the hydraulic cylinder connected with the outer pressure unit. The design of the facility enables to keep the load constant (zero) during the start up of the loop operation. After the parameters of the environment are stable, the specimen is loaded with small load to fix the specimen. Then the loading according the planned experimental conditions can be started.

Compact tension specimens testing facility has been developed to test crack growth in specimen. Static, quasi-static and cyclic loading can be applied. Two specimens can be tested in one frame simultaneously, Fig.7. They are electrically insulated from the rest of the frame by ceramic coated pins. The maximum load is 170 kN. The reverse direct potential drop (RDCPD) measurement is applied in order to measure crack growth in the specimens. The accuracy of measurement is better than 50 µm.

Another in pile testing represents constant load and constant deformation measurement of C-rings and CT specimens. The holder is placed in the active channel of PWR water loop. After required time interval, the surfaces of C-rings are checked for the presence of cracks. The initiation stresses of SCC damage have been tested that way.

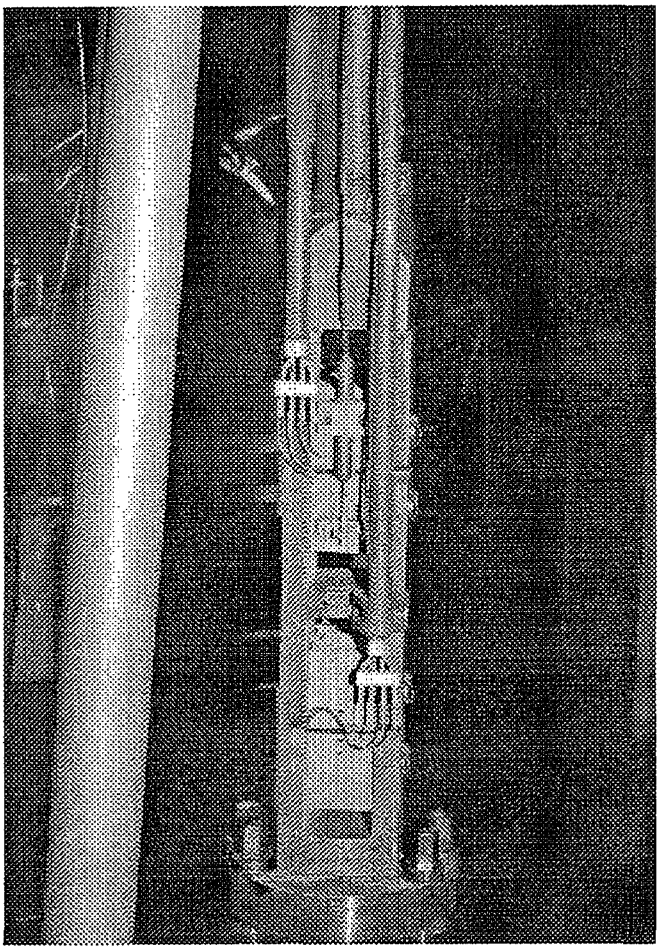


Fig.7: Testing frame for two CT specimen

6. Principle tests performed for IASSC assessment

Five principle experiments have been performed in the in pile and out of pile facilities:

- Slow strain rate test of tensile specimens. Deformation rates of 10^{-7}/s and lower are used. As a result, changes of mechanical properties and fracture feature is evaluated. From the existence of brittle cracks, even transgranular or intergranular, the crack growth rate can be estimated. General sensitivity to SCC or IASCC is determined by this test.
- Constant loading of CT specimens with continuous on-line monitoring of crack length.
- Rising displacement tests with CT specimens. From these tests, both threshold values of J-integral and crack growth rates are evaluated.
- Cyclic loading of CT specimen. Crack growth rate is evaluated after the test.
- Constant loading of C-rings. Initiation stresses and time to initiation is determined for pre-irradiated specimens.

7. Post experimental treatment

After the experiment is finished, fractographic analysis is a standard procedure with both non-irradiated and irradiated specimens. The hot cell with the SEM has been designed and prepared, see Fig.8.

Hydrogen can be accumulated in the specimen during the test or, influence of hydrogen can be tested by pre-charging of test specimens. To determine hydrogen content in the specimens, Balzers analyzer EAH 220 was adopted for the measurements of irradiated specimens.

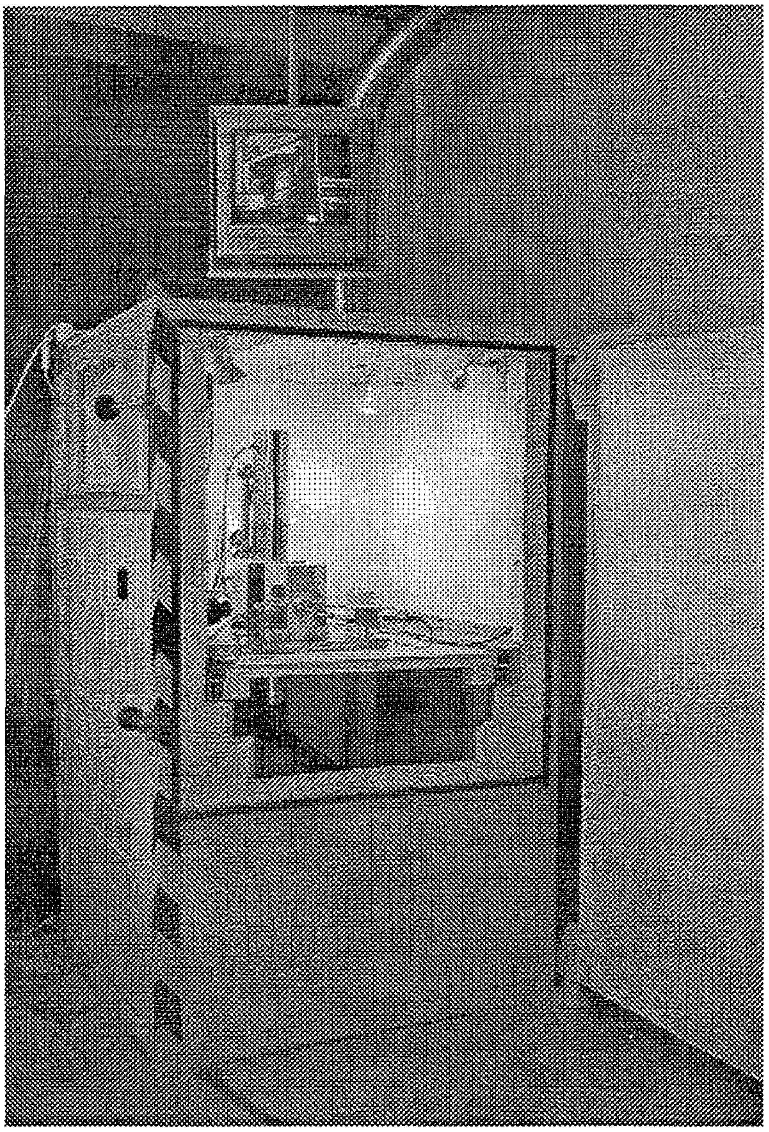


Fig.8: Hot cell with SEM

8. Examples of testing

As an example of IASCC testing, the experiments performed during the program of assessment of reactor pressure vessel and internals of WWER 1000 reactors will be used. Two types of experiments were performed: slow strain rate tests of pre-irradiated specimens and constant loading of CTs and C-rings.

In Fig. 9, there is the stress - deformation line of the loading of 08Kh18N10T steel preirradiated to the fluence of $1x10^{20}$ n/cm^2 in the comparison with non-irradiated steel. Even only small changes are present, the effect of preirradiation on yield strength and slight decrease of total elongation are seen.

In Fig. 10 there is the surface of C-ring tested in pile after preirradiation. The material of C-ring is 14Kh17N2.

In the present time, in the field of IASCC the tests for the Cooperative IASCC Research Program lead by EPRI are performed. As well, the national program of the reactor internals assessment has been started.

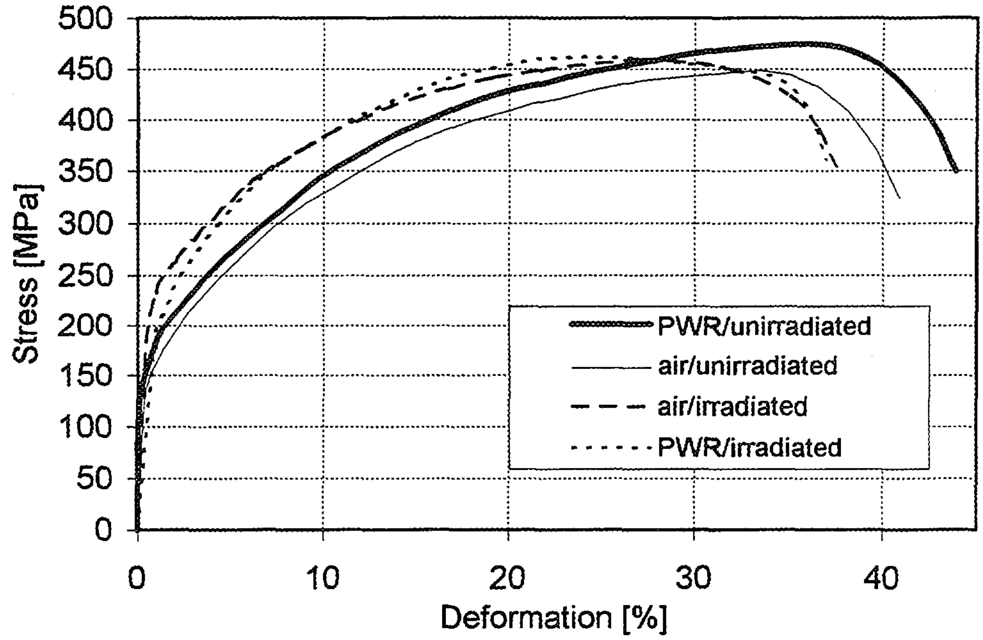


Fig.9: Stress - deformation line of the loading of 08Kh18N10T steel preiradiated to the fluence of $1x10^{20}$ n/cm^2 and non - irradiated.

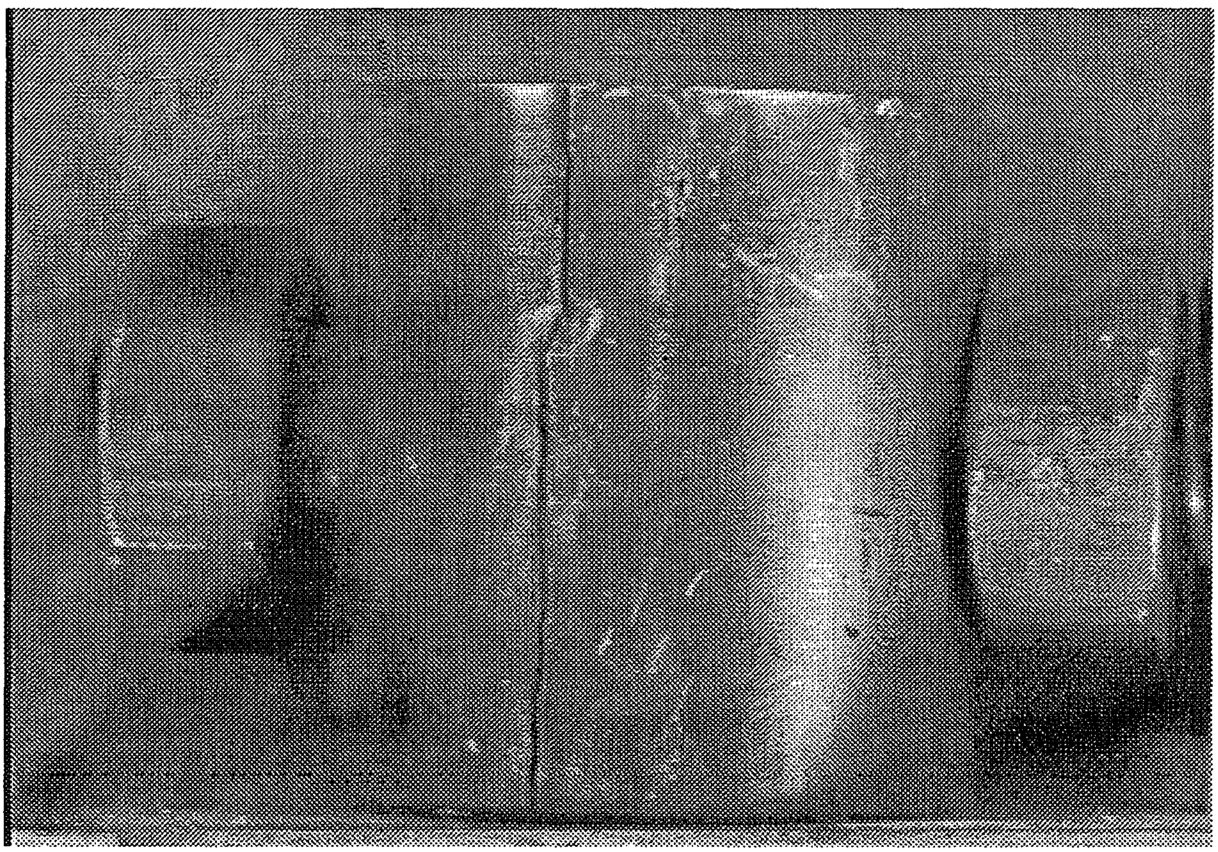


Fig.10: Surface of C - ring tested in pile after preirradiation

9. Conclusions

Nuclear Research Institute has developed extended system of testing facilities for the IASCC testing. Both in pile and out of pile testing are performed with non-irradiated and irradiated specimens. As the typical tests, slow strain rate tests and CT tests are performed. Al the work is supported by the hot cells capabilities and design experiences.

10. Reference

[1] P.L. Andresen, P.P. Ford, S. Murphy and J.M. Perks. "State of Knowledge of Radiation Effects on Environmental Cracking in Light Water Reactor Core Materials". Proc. 4th Int. Conf. on Environmental Degradationof Materials in Nuclear Power Systems - Water Reactors. Jekyll Island, GA, August 1989, p.1-83