ERIC VAN WALLE

REACTOR MATERIALS RESEARCH

Background and Objectives

The Reactor Materials Research department evaluates the integrity and behaviour of structural materials used in nuclear power industry and research as a function of several parameters such as temperature, pressure, neutron fluence, burn-up, material composition and water chemistry. The finality of this research is to contribute to the interpretation, the modelling of the material behaviour and to develop and assess strategies for optimum life management of nuclear power plant components. We give commercial projects high priority and we primarily use their outcome to generate new research that can give more added values to upcoming projects.

The department runs the LHMA (Laboratory for High and Medium level Activity) in order to perform state-of-the-art PIE (Pre and Post Irradiation Examination), operates the BR1 (Belgian Reactor 1) and BR2 hot-cells and makes elaborate use of the BR2 MTR (Materials Testing Reactor) for material irradiation. The programs within the department concentrate on four distinct disciplines:

- fusion (mostly mechanical testing, reported in another contribution to this annual report);
- Irradiation Assisted Stress Corrosion Cracking (IASCC);
- nuclear fuel;
- Reactor Pressure Vessel Steel (RPVS).

As LHMA is central to all work related to hot-cell infrastructure, we are moreover heavily solicited by other departments of SCK•CEN and by external companies.

Programme

The Reactor Materials Research (RMR) programme covers a vast number of projects that are of commercial and of research origin. Some projects are almost continuous, some have clear deadlines. Typical for hot-cell work is that the occurrence or continuation of projects can be hampered by transport problems (mainly fuel from external laboratories or nuclear power plants - NPPs) or inactivity of hot-cells due to interventions. This is for example the reason why some projects, scheduled to be finalised in 1999, are still unfinished in 2000.

In 2000, the main topics in the RMR programme were:

- the extension of the numerical simulation of the IASCC phenomenon;
- the preparation for the installation of a hot-cell for IASCC experiments, the dedicated irradiation of specimens susceptible to IASCC and the development of specific instrumentation for these complicated irradiation programmes;
- the fundamental study of the grain boundary behaviour of cesium in nuclear fuel;
- the installation of an X-ray diffractometer in SCK•CEN's thermochemistry laboratory;
- the continuation of the fission gas release modelling activity;
- the feasibility study of an experimental irradiation program THOMOX involving instrumented irradiation of advanced nuclear fuels;
- the continuation of the BACCHANAL fuel irradiation;
- the evaluation of the operational behaviour of fuel rods with different cladding irradiated up to very high burnup;
- the continuation of the Electrabel Convention with major contributions in the field of RPVS and IASCC;
- the finalisation of the European Union sponsored RESQUE project for which SCK•CEN was the co-ordinator;
- the kick-off of the FP5 European projects INTERWELD, LIRES, PRIS (IASCC) and FRAME (RPVS);
- the continuation of the TACIS-projects in support of VVER-1000 surveillance;
- the finalisation of the VVER-440 IAEA programme on irradiation, annealing and re-embrittlement;
- the successful organisation of the 9th IGRDM-Symposium (International Group on Radiation Damage Mechanisms);
- the organisation by SCK•CEN of a Belgian workshop on "Fuel for Power Reactors" in Brazil with the participation of AVN (AIB Vinçotte Nuclear), BELGONUCLEAIRE, Tractebel Energy Engineering;
- the studies on RAFM (Reduced Activation Ferritic-Martensitic) materials and Chromium alloys in the context of fusion research.

Achievements

Corrosion

Components close to the reactor core, such as the pressure vessel internals, accumulate a high fast neu-

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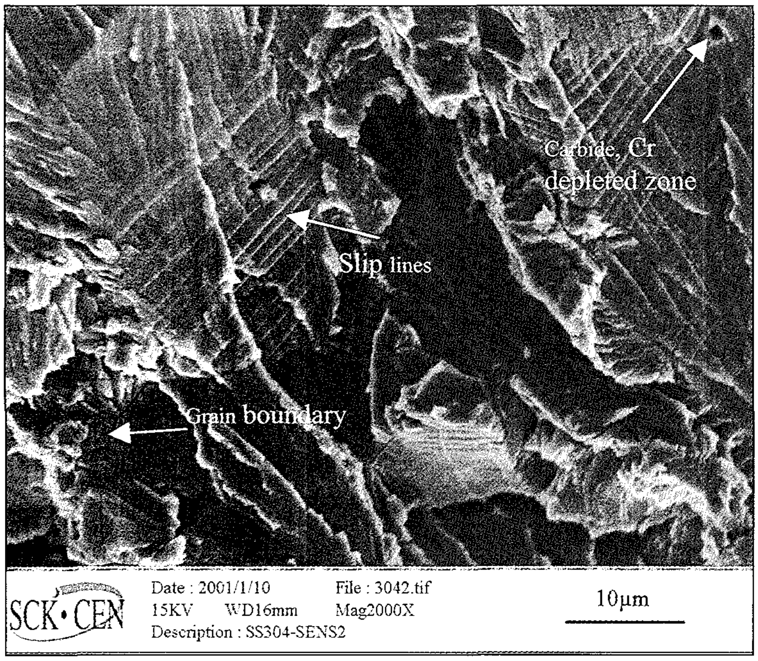
tron fluence during the lifetime of a nuclear power plant. It is well known that irradiation by neutrons changes the properties of the materials. These changes affect both the mechanical properties (strength, ductility and fracture resistance) and the microchemistry of the material (which results in redistribution of the alloying elements inside the microstructure). Moreover, the presence of intense radiation fields causes radiolytic decomposition of the coolant and influences the environment to which the material is exposed.

These effects (individually or combined), may render a material-environment system (such as stainless steel in high-temperature water) susceptible to irradiation assisted stress corrosion cracking (IASCC).

The corrosion research programme of SCK•CEN aims to clarify the mechanisms contributing to IASCC in order to formulate improved inspection schemes and mitigating measures. Therefore, we have three complementary activities under development:

- *modelling of IASCC and related phenomena*: a computer code takes into account chemical and electrochemical reactions at the surface of the corroding material and in the environment. It models transport phenomena and the coupling between mechanical loading and stress corrosion crack propagation. We consider complex geometric systems and we include the influence of irradiation on the (stress) corrosion process;
- *experimental study of IASCC in Light Water Reactor (LWR) conditions*: LWR-conditions include high temperature (250-350°C) and high pressure (up to 15MPa), and make the study of IASCC non-trivial. Presently, tests on non-irradiated materials generate reference data for modelling and future experiments on irradiated specimens. Presently, we also construct a hot laboratory at LHMA to perform stress corrosion tests at LWR conditions. We take irradiated specimens from dedicated experiments in BR2 or from components extracted from power reactors;
- *development of dedicated instrumentation*: no 'off-the-shelf' instrumentation is available to study the phenomenon of IASCC. A programme to develop, test and validate instruments and methods for detection of (IA)SCC in LWR conditions, with emphasis on high temperature reference electrodes, is currently underway.



Typical stress corrosion crack morphology, obtained in sensitised AISI 304, tested at 300°C in oxygenated PWR water.

Modelling of IASCC

In 2000, the numerical simulation of corrosion was taken one step ahead. This is shown in a number of applications and by the further development of the computational electrochemistry code, ELEMD, in collaboration with the Vrije Universiteit Brussel (VUB), the Elsyca n.v. and the Von Karmann Institute (VKI).

Predicting crack growth rate versus temperature behaviour of Type 304 stainless steel in dilute sulphuric acid solutions. The coupled-environment fracture model for stress corrosion cracking of Macdonald is extended to incorporate the effects of sulphuric acid additions and to include thermal activation of the crack tip strain rate. These extensions allow, after calibration, to compare theoretically estimated and experimentally determined crack growth rates over a considerable temperature range.

Measuring and modelling the IR drop and the resulting active-passive transition of iron in an acetic acid solution. The IR (I = corrosion current, R = ohmic resistance of the electrolyte) drop model of Pickering is used to model the active-passive transition and the potential drop in a creviced iron sample in a buffered acetate solution. The numerical calculation, based on solving a Laplace equation with appropriate boundary conditions, is in good agreement with the experimental results.

The influence of conducting walls on electrochemical measurements. The correct measurement of the corrosion potential is essential, as it is a fundamental parameter when investigating aqueous corrosion. Electrochemical measurements under high pressure and high temperature are usually carried out in stainless steel autoclaves. We showed through numerical calculations and laboratory experiments that the presence of electrically conducting walls influences electrochemical measurements. We proposed methods for correcting the measurements.

Development and application. The mixed potential model for high temperature and pressure LWR environments is implemented in ELEMD. The model accounts for oxygen reduction, hydrogen evolution, hydrogen-peroxide reduction and metal dissolution. Analytical and computational results are compared for a rotating disc electrode. The computational approach allows extrapolation to a more complicated geometry. Limiting currents are obtained numerically by solving the full diffusion-migration-convection equations.

Experimental studies of SCC in LWR environment

The characterisation of the SCC behaviour of austenitic stainless steels in LWR conditions, requires test methods to be applied in autoclave environments. We produced a first data set on the SCC propagation rate in stainless steel in simulated primary PWR environment. We evaluated the cracking behaviour of AISI 304 after cold work or/and sensitisation by SSRT (slow strain rate tensile) and crack growth rate (CGR) measurements on CT (compact tension) specimens. The magnitude of and trends in the data agree with published information in literature and other international programs. The qualitative accordance between SSRT and CGR measurements is good: the cracking mechanisms are comparable for the different materials and testing conditions. It is important to note that the results of the SSRT tests as such, can lead to erroneous conclusions on the relative difference in cracking resistance when only the fractographic results are considered (as is often the case in literature). In our work, it is shown that, under the same conditions, the sensitised material (which demonstrated a considerably higher fraction of SCC in the SSRT tests) has a lower crack propagation rate than the cold worked material. This is due to the lower ductility of the cold worked material, which limits the extend to which a stress corrosion crack can propagate in an SSRT test before overload fracture occurs. In the constant load tests on CT specimens, the crack propagation is stable, so the true difference in stress corrosion cracking resistance of the material can be revealed.

The yield stress is the main determining factor for SCC in PWR water. The presence of oxygen as an environmental factor, is found to strongly accelerate cracking. The latter is also observed when a very low potential is applied (corresponds to strong hydrogen generation). The cracking is predominantly transgranular and is different from the intergranular cracking generally observed in irradiated stainless steels. A better reproduction of the behaviour of irradiated materials, requires a more accurate simulation of the irradiated microstructure.

Technically, the method is improved by applying alternating current potential drop measurements, with better signal to noise ratio at constant current level than direct current methods. Nevertheless, in the test periods applied, the lower limit for measuring crack propagation rates of the testing method is about 10^{-8} mm/s.

INTERWELD, an EU sponsored R&D program, focuses on the evolution of residual stress and microstructure in large welded internal reactor structures and investigates the relation with IASCC. The programme considers welds that are irradiated in-service and model welds, irradiated in an MTR. We will study the relation between residual stress, microstructure, microchemistry and IASCC susceptibility experimentally with neutron diffraction, electron microscopy, Auger and analysis by secondary ion mass spectroscopy. This will be combined with SSRT tests in a simulated reactor environment. The aim is to study to what extend the residual stress relaxation during irradiation interacts with the threshold dose for IASCC.

The objective of PRIS, also an EU sponsored R&D programme, is to produce material data for irradiated austenitic stainless steel LWR internals – both BWR and PWR – as a function of fluence. We will use the data for structural integrity studies and lifetime assessment. The information consists of validated initiation fracture toughness data, fracture resistance curves, tensile properties, as well as data on microstructure changes caused by irradiation.

Instrumentation development for SCC detection

An important parameter for IASCC - and for any other aqueous corrosion process - is the corrosion potential. Hence, when monitoring the corrosion potential with a reference electrode, we can evaluate the likeliness of IASCC to occur. We developed a high temperature reference electrode that can be used for electrochemical corrosion studies in an autoclave under relevant PWR conditions. The electrode is an External Pressure Balanced Reference Electrode (EPBRE) connected to the autoclave by means of a (cooled) salt bridge. Our laboratory developed a potential correction algorithm to calculate the potential of the EPBRE against the Standard Hydrogen Electrode (SHE) scale. During the design emphasis was put on easy handling of the hot-cell EPBRE with manipulators and on the resistance of the reference electrode to irradiation.

The main objective of the LIRES, also an EU-sponsored R&D programme, is to develop a reference electrode, which is robust to be used inside a LWR. A four-step development trajectory is foreseen:

1. To set a test standard for a Round Robin exercise;
2. To develop different reference electrodes;
3. To perform Round Robin tests of these reference electrodes, followed by the selection of the best reference electrode(s);
4. To execute irradiation tests under appropriate LWR conditions in a MTR. The kick-off meeting and the first progress meeting on item (1) have been held.

Fuel

Nuclear fuel research at SCK•CEN combines both applied and fundamental research. The aim is to develop the competence to respond to the rapidly evolving demands in advanced research. The fundamental part focuses on solid state research of nuclear fuel, on modelling of fuel behaviour and on the definition, technical preparation and execution of in-pile instrumented irradiation experiments. We took the first steps for the development of dedicated fuel for Accelerator Driven Systems (ADS).

Applied research essentially remains market-driven. The tendencies of future research are the long-term intermediate storage of spent fuel and issues related to the further increase of discharge burnup. The problem of transporting fuel rods from the NPPs to research institutes has become very pressing. SCK•CEN applied to obtain a license for its BG18 transport container, in co-operation with Transnubel (TNB) who will operate the SCK•CEN container. The license is expected to be granted early 2001.

Fundamental Nuclear Fuel Research

Thermochemical fuel research

In 2000, we continued to study the fundamentals of the interaction between alkali metals (in particular, but not exclusively cesium) and UO_2, with the objective to deepen the knowledge on the grain boundary behaviour of cesium in nuclear fuel. In close collaboration with the ITU (Institute for Transuranium elements, Karlsruhe), uranium and cesium were sputter deposited *in situ*, forming thin layers which could subsequently be studied with X-ray Photoelectron Spectroscopy (XPS). It is a known fact that for reactively deposited uranium oxide layers (sputter deposition of uranium in an oxygen-argon atmosphere), uranium is never found in valence states higher than IV. With the addition of cesium to the sputter atmosphere, the co-deposited layers show that uranium can be forced in higher valence states, up to U^{VI}. Even more remarkable is the fact that cesium covered UO_2 surfaces, when exposed to oxygen, will oxidise much faster than uncovered surfaces. It is believed that the capability of cesium to decompose oxygen molecules and form ionic oxygen species lies at the origin of this behaviour.

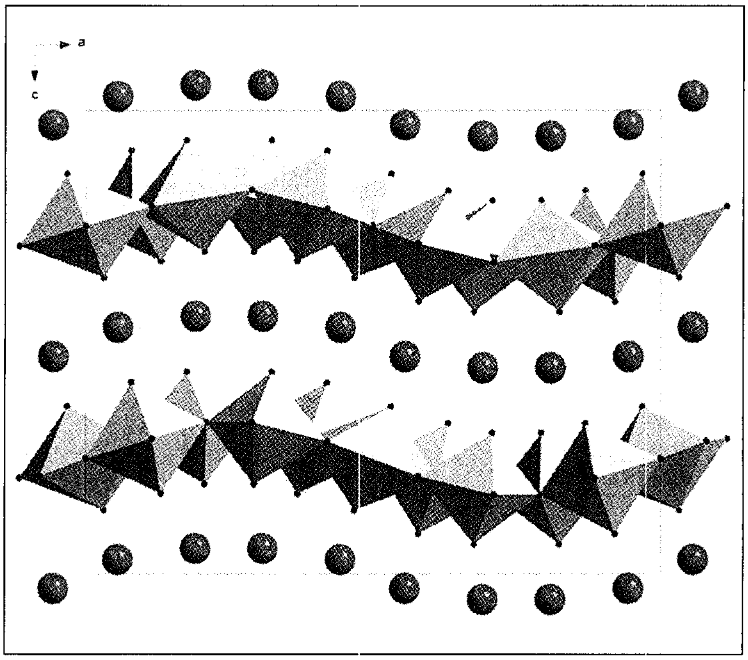
SCK•CEN's thermochemistry laboratory is equipped with an X-ray diffractometer that allows to perform in-situ analyses under controlled atmosphere up to 2000°C. The increased resolution of modern X-ray equipment has a slight drawback since the inherent asymmetry of low-angle diffraction peaks is more pronounced. This has to be correctly accounted for in the structure refinement programmes. We conducted an extensive study on the asymmetry of low angle diffraction peaks. With the tubular furnace that works under controlled atmosphere, the laboratory has successfully continued the study on the crystallographic structure of alkali metal uranates and the uranium valence states in these compounds. The analysis of cesium uranate was complemented with neutron diffraction experiments at the Institut Laue-Langevin in Grenoble, and with spectroscopic investigation of the local environment of the uranium atoms by X-ray Absorption Fine Structure Spectroscopy (XAFS). The combination of neutron diffraction, X-ray diffraction, XPS and XAFS will lead to a better understanding of the local uranium environments.

Modelling of fission-gas release in LWR fuel

Since 1994, we develop a new mechanistic model for fission gas release (FGR) in LWR fuel. The development involves two complementary aspects.

First, we analyse basic underlying mechanisms and focus on the grain boundary behaviour as this determines the onset of release. In particular, we developed a model for the precipitation of fission products in a grain boundary, which embodies a variable reaction rate on the precipitate surface. This enables to account for modifications of the local fuel chemistry, or to distinguish between the behaviour of different migrating species. In addition, we have assessed the influence of the trapping parameters on the precipitation rate according to different models from the open literature. We extended these models to incorporate the variable intrinsic reaction rate. We established the interrelationship among the models while we discussed their limitations and range of validity. The results reveal a critical value above which the influence of the intrinsic reaction rate (between a fission product and an intergranular trap) on the global precipitation rate becomes negligible. This is beneficial given the uncertainty pertaining to this parameter. In addition, it justifies the assumption of an infinite intrinsic reaction coefficient for intergranular bubbles in FGR models, and can explain the similar behaviour of different species for which the reaction rate coefficient is larger than the critical value.

The second aspect is the development of a model for FGR that couples the kinetics of the intra- and intergranular behaviour of gas atoms in both directions. As such, we will be able to assess quantitatively the contribution of the various mechanisms to the overall release process. In 1999, we developed the model for thermal release and coupled it with the FTEMP2 code of the Halden Reactor Project, providing the radial temperature distribution only. This year, we coupled the model with the COMETHE code of BELGONUCLEAIRE to provide the radial distribution of the temperature, the power and the burnup in the pellets. The simulation of the empirical Halden threshold for release proved to be quite satisfactory. Moreover, the predicted grain boundary saturation concentration, $2x10^{15}$atoms/cm^2, corresponds very well with literature. At the same time, the predicted outward shift of the radius within fuel for onset of thermal release with burnup is in accordance with experimental observations. Nevertheless, an overprediction of the threshold for FGR appears at low temperatures, where fission gas filled bubbles at grain boundaries - resulting from diffusion controlled precipitation - are not yet developed. We also did not account for the open porosity resulting from the fabrication process. We therefore included an athermal release process. This component is determined by the athermal open porosity fraction.



Crystal structure of Cs4U5O17 as determined from the neutron and X-ray diffraction experimental data. The figure presents a view along the <010>-direction. Each octaeder contains an uranium atom at the centre and has oxygen atoms at its corners.

Integral experiments

In 2000, we successfully completed the feasibility study of an experimental irradiation programme THOMOX. The project involves an instrumented irradiation of a series of advanced nuclear fuels under PWR conditions (8 different fuel types and a total of 16 fuel rods) and aims at the assessment of both the FGR and the thermal properties of the fuel rods. The fuel types are all ceramic oxides and include $(Th,Pu)O_2$, $(U,Pu)O_2$ and UO_2. The main parameters are the degree of dispersion of fissile atoms and the matrix in which they are embedded. We will irradiate the fuel for 4 years in the PWR CALLISTO loop of BR2. The target burnup is 50 GWd/tM and the life average linear power will be 300 W/cm. To achieve the target burnup within the proposed time, we proposed subsize rod dimensions and remind that the flexibility of the irradiation conditions in BR2 needs to be utilised.

The irradiation programme BACCHANAL on burn-up accumulation in MOX fuel of BR3 design, continued. The experimental and theoretical results of this programme serve as a basis for more advanced irradiation programmes. The follow-up of BACCHANAL includes the assessment of the linear power of the bundle and the comparison of calculated predictions, the on-line follow-up of the cooling water in the PWR loop and the out-of-pile gamma spectrometry of the fuel rods.

The refabrication of irradiated fuel rods is close to implementation in hot-cell. The welding conditions were optimised and the quality control procedures were prepared. In 2001, the equipment will be installed in the hot-cell.

Applied Nuclear Fuel Research

We evaluated the operational behaviour of fuel rods with different type of cladding, and high burnup (80 GWd/t_M) .The evolution of the fuel microstructure and the corrosion/hydriding of the cladding were studied by optical microscopy.

The electron microprobe (EPMA) based methodology to analyse a poly-dispersed binary system (UO_2/PuO_2) has been further applied on non-irradiated MOX. Its excellent quantitative reliability in combination with a spatial resolution of 1µm on the Pu content and distribution has been confirmed.

Reactor Pressure Vessel Steel

Electrabel Convention 1999-2003

Reconstitution Technology

In recent years, the development of new laser systems has increased the interest of laser welding for reconstitution purposes. We performed a preliminary study to evaluate the capabilities of a 45 kW CO_2 laser to reconstitute Charpy-V specimens. Careful selection of the laser welding parameters (speed and power of the beam) for the reconstitution of Charpy specimens reduces the width of the weld and the heat-affected-zone and lowers the maximum temperature in the specimens during welding.

Important results of this study are:

- CO_2 laser beam welding can be successfully applied for reconstituting Charpy–V specimens (10x10mm²). The optimum parameters are a beam power of 22 kW for a welding speed of 4 m/min;
- temperature measurement shows rapid heating and cooling during the laser welding process. The maximum temperature is 430 °C at a distance of 2 mm from the welded joint and less than 300 °C at a distance larger than 3.5 mm, in agreement with the RESQUE results;
- three-point bending tests on laser welded un-notched specimens show good mechanical resistance of the laser welds. The methodology of the un-notched three point bend test was successfully established to estimate the quality of the welds during European Round-Robin with the acronym RESQUE;
- charpy impact tests show good agreement between original and laser-reconstituted specimens.

Development of the potential drop technique for ductile tearing investigations

The Multiple Specimen Test Technique (MST) and Unloading Compliance Test Technique (UCT) are qualified techniques, which are widely used. However, the UCT applied to small specimens such as precracked Charpy specimens (PCCv), underestimates the crack growth prediction and is very sensitive to compliance measurements. Moreover, the UCT does not always give satisfactory results with very ductile materials such as aluminium. Therefore, it was decided to investigate the use of the potential drop (PD) technique within the Electrabel – SCK•CEN Convention 2000.

We developed the direct current potential drop (DCPD) technique and we investigated and quantified the effect of several parameters. We can summarise the important points of the investigation as:

- to avoid excessive specimen heating, we recommend to apply pulsed current instead of constant current;
- to avoid errors due to thermocouple effects, we recommend to measure the potential during and after the pulse. When this procedure is used, the reverse DCPD is not needed to avoid errors due to thermocouple effects;
- temperature is an important parameter as the electrical conductivity highly depends on it. The temperature effect was taken into account using a temperature calibration function;

- within the specified region, the position of the current probe is not very important;
- contrary to ACPD, the technique is not sensitive to an induced potential. It is, however, recommended to use shielded cables;
- the PD method is very sensitive to the position of the potential probe. To improve the repeatability of the measurement we recommend to measure the potential using a clip gauge;
- we successfully developed a clip gauge allowing simultaneously PD and Crack Mouth Opening Displacement (CMOD) measurement;
- the dependence of V_0 (initial potential) with the load should be taken into account to obtain an accurate blunting line.

Development of the crack arrest technique

When it can not be demonstrated that a fast propagating crack could initiate, the investigation of the crack arrest concept is required. However, the current regulation is based on a semi-empirical methodology to determine the lower bound crack arrest curve and this results in large conservatism. Therefore, it would be interesting to obtain a direct measurement.

The crack arrest test technique is standardised in ASTM E1221-96. Although there is an interest to measure this material property, very few laboratories are currently performing such test. The reasons being that it requires large specimens and the test results should satisfy five stringent criteria before being considered as a valid measurement. Therefore, this technique has nearly never been used with irradiated specimens.

The reasons for developing the crack arrest technique at SCK•CEN is:

- to gain expertise in this domain;
- to verify and establish a physically based correlation with small specimens, such as Charpy specimen.

We developed successfully the technique and performed tests on large specimens of 1 and 2 inch thickness.

3-D Finite Element Modelling of Compact tension specimen for Loss of Constraint Analysis

To support the new standard ASTM E1921-97 for fracture toughness testing in the transition regime, we performed 3-D finite element calculations on side-grooved and non side-grooved PCCv and CT specimens. The goals of this study are:

- to assess the ASTM procedure for fracture toughness determination;
- to compare PCCv and Compact Tension (CT) specimens in terms of constraint, which is an important parameter controlling cleavage fracture.

The main deliverables of this investigation are:

- the formulations proposed in ASTM E1921-97 to calculate the J-integral for a PCCv and 1T (1 inch thick) CT are sufficiently accurate for application, although improvement is possible. The maximum error on the reference temperature is 3 °C;
- CT and PCCv specimens display a plane strain behaviour at low load level;
- no constraint difference between PCCv and 1T-CT is observed up to 40 MPa√m;
- CT and PCCv specimens are possibly a more constrained geometry than the SSY (Small Scale Yielding) plane strain condition;
- the constraint for a PCCv loaded above 40 MPa√m decreases when compared with a 1T-CT specimen;
- the model predicts a reference temperature about 10 °C lower for a PCCv specimen as compared to a 1T CT specimen. This is an important result that is supported by a large experimental database.

Development and Validation of Fracture Toughness Tests on Miniaturised Specimens

The objective of this activity is the validation of direct fracture toughness measurements using specimens of small-size geometry, which can be obtained from machining broken Charpy-V specimens. The two specimen types are:

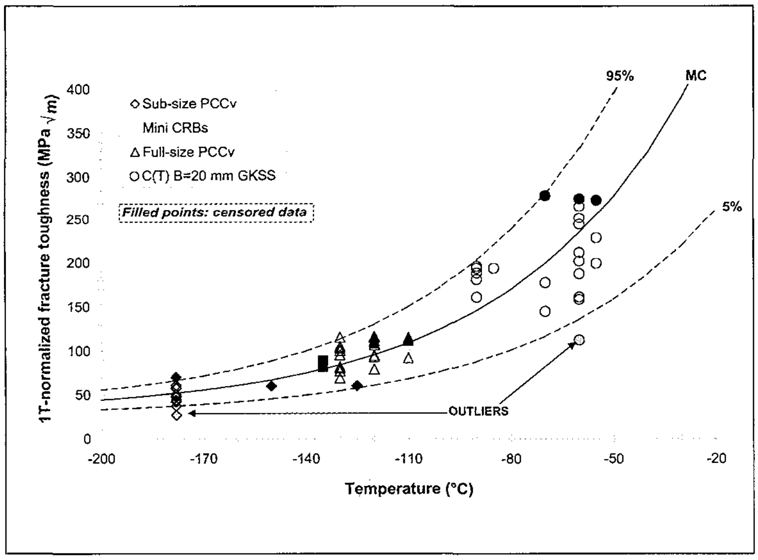
- the miniaturised precracked Charpy-V specimen (MPCCv);
- the miniaturised Cracked Round Bar (MCRB).

We obtained fracture toughness results in the ductile-to-brittle transition regime for two well-characterised reactor pressure vessel steels, and compared using the Master Curve analysis to reference data measured using bigger samples of the more conventional CT geometry.

To cope with specimen size limits established by test standards in order to avoid loss of constraint, we chose two strategies:

- testing at very low temperatures, thus approaching lower shelf conditions;
- correcting fracture toughness for loss of constraint, where loss of constraint was assessed by in depth finite element calculation.

We found sub-size results in agreement with reference data, although in case of tough materials tests have to be performed at very low temperatures. From this point of view, the MCRB specimen offers the advantage that an analytical correction for the loss of constraint is already available.



Fracture toughness values obtained from miniaturised and standard specimens on the DIN 20MnMoNi55 steel. Scatter and size effects are well represented by the master curve.

Qualification of Force Values Measured by Instrumented Impact Strikers

Forces measured during instrumented impact tests are used for different purposes, including the assessment of irradiation effects using the enhanced surveillance strategy. Thus, confidence in the calibration procedure used to convert tup strain-gage output into forces applied on the test specimen must be optimised.

In 1999, we established that the most promising approach to the qualification of instrumented impact forces was the comparison between maximum loads obtained in quasi-static and dynamic (impact) tests, using an Aluminium alloy 6061-T651 which is quasi strain-rate insensitive.

We completed the characterisation of the strain-rate sensitivity of this alloy by performing tensile as well as three-point-bend tests at different strain rates. Furthermore, we extended the range of maximum forces investigated from 3 to 22 kN (force range typically encountered in actual Charpy tests on RPVS), by using modified Charpy specimens with widths (in the direction of striker movement) ranging from 3.5 to 15 mm.

Most of the strikers available in the Mechanical Test Lab of RMO for the three instrumented pendula (Tinius-Olsen, Toni-MFL and Wolpert) have been verified by comparing quasi-static to dynamic (strain-rate corrected) maximum forces: this allows to rank both ASTM and DIN tups, and to establish the most suitable for performing impact tests on non-irradiated or irradiated samples according to ASTM or ISO standards. Finally, we obtained optimised calibration curves for every instrumented striker by minimising the differences between strain-rate dynamic maximum forces obtained experimentally and reference values yielded by quasi-static tests.

Radiation Effects on RPV Steels

We initiated this work in 1994 to better understand the thermal ageing/irradiation embrittlement of the Doel I-II weld. We completed the analysis of the supplementary test programme, including tensile tests with miniature tensile specimens charpy impact tests and fracture toughness tests with precracked Charpy specimens. For various conditions of thermal ageing, irradiation levels and post-irradiation heat treatments, we analysed the test results using the load diagram approach. While the experimental data were globally rationalised using the available tools (load diagram, crack arrest, strengthening model), unresolved questions remain, in particular in relation to the copper effect (high versus low copper weld) and role of the thermal component in the embrittlement of such welds.

In the framework of the enhanced surveillance strategy, the load diagram plays a central role. A user-friendly program has been written in Excel in order to apply in a more efficient way the load diagram approach to RPV Charpy and tensile test data. We selected a number of RPV steels in order to qualify the programme.

It is known that the observed embrittlement of RPV steels results from various embrittlement mechanisms that are affected by the material specification (chemistry, heat treatment), as well as the environmental conditions of exposure (irradiation temperature, neutron fluence and flux). The damage model developed some years ago for rationalising RPV steel strengthening was implemented in a user-friendly Excel programme. It is currently applied to an extensive databank to evaluate its performance.

Commercial research

Surveillance capsules from the Belgian NPPs

In 2000, the Doel I/5 surveillance capsule was retrieved from the reactor. We foresee transport to SCK•CEN early 2001 and the analysis according to an enhanced analysis scheme should be performed by June 2001.

SAMBA collaboration with CTMSP, Brazil

The heat-treated rings of base and weld material from the Brazilian INAP-reactor where delivered at SCK•CEN in the second half of the year. Under supervision of AVN and RMR, the surveillance specimens of the INAP-reactor were taken from these rings by an external company. Early 2001, the testing of the reference data for the INAP surveillance program will start, while the other specimens will be encapsulated in 12 surveillance capsules.

In September 2000, SCK•CEN organised a workshop on "Combustível para reatores de potência no Brasil" nearby Rio de Janeiro. The emphasis of the workshop lay on fuel licensing. The Belgian organisations that were contracted by SCK•CEN to contribute to the successful workshop were AVN, BELGONUCLEAIRE and Tractebel Energy Engineering.

SCK•CEN sponsored research

The REVE (REacteur Virtuel d'Etude) Project is an international effort led by EDF (Electricité de France) aimed at developing computational tools, based on well grounded techniques such as Molecular Dynamics, Monte Carlo methods, Dislocation Dynamics, capable of predicting the behaviour of materials under irradiation. These tools, each of which deals with a well-defined space and time scale, are currently being routinely used as separate packages in several laboratories around the world. The underlying idea of the REVE project is twofold. The first objective is to build, through close collaboration between European centres (EDF, SCK•CEN, CEA, CIEMAT and various French, Belgian and Spanish universities), a suite of codes capable of deducing, from a detailed simulation of the microstructure changes produced by irradiation in materials, the corresponding changes in macroscopic properties. The second objective, strictly related to the first one, is to enhance collaboration and data exchange between European, American (UCSB, LLNL, ORNL,) and Japanese (CRIEPI,) laboratories, currently tackling the same challenge, in order to speed up the development of such suite of codes. The material chosen for the first phase of this project is RPV steel. Within this project, with a view to preparing the next generation of codes, extended also to other, more complex materials, a EU-Commission-funded Thematic Network for the Improvement of TEchniques for Multiscale modelling (ITEM), which involves 40 European laboratories, has been proposed. Within REVE, SCK•CEN carries out two actions:

1. The development of the simulation codes requires a continuous validation based on ad hoc experimental data: at SCK•CEN the relevant neutron irradiation experiments and most of the subsequent materials characterisation to produce such a data base are going to be conducted. The REVE experimental programme envisages the irradiation of 10 different materials in 5 different flux/fluence conditions in the CALLISTO loop of BR2, and their post-irradiation characterisation through testing of very different specimens, from atom-probe needles to tensile samples.

2. In addition, taking advantage of the expertise accumulated at EDF and elsewhere with the use of multi scale modelling tools, a Materials Simulation Group is in the process of being set up at SCK•CEN, externally supported and promoted by prof. Marc Hou (ULB). This group will initially focus its activities on the problem of the simulation of experimentally observed Cu precipitates stabilised by vacancy clusters in ferritic alloys by means of Monte Carlo techniques, so as to be able to study their interaction with dislocations (key issue in order to determine the yield strength increase produced by irradiation in these materials), this time by means of Molecular Dynamics methods. The longer-term objective is also to implement and develop at SCK•CEN computational tools capable of acting as a link between the simulation and the experimental observation (TEM image simulation, PAS calculation, ...).

IAEA Round-Robin on VVER-440 weld

SCK•CEN finalised its tasks that consisted of:

- the irradiation, annealing and re-irradiation of a critical VVER-440 weld. The irradiations were carried out in the CALLISTO loop and had code name CHIVAS-7 and 8;
- the testing of the numerous specimens, the analysis and reporting of the results;
- the results show that almost full recovery of tensile, fracture toughness and impact properties can be obtained after annealing. The temperature shifts are conservative towards Russian regulation and the re-embrittlement behaviour is intermediate to the lateral (non-conservative in this case) and conservative shift model.

EU-sponsored research projects

RESQUE

We finalised the RESQUE project and the final report, that consists of a number of recommendations for the reconstitution of irradiated material, was submitted to the EU. We introduced an "Accompanying Measure" dossier to the EU to organise a workshop on the findings of RESQUE (and REFEREE). The request was granted and the seminar will be held in 2001.

FRAME

The objective of the cost shared action FRAME is to investigate the eventual dependence of the master curve, used in fracture toughness analysis, on material composition. A number of irradiated model alloys with a chemical composition that has been varied systematically will be tested and analysed. The kick-off meeting of the project was held.

TACIS

RMR is through Belgatom involved in two TACIS projects on VVER-1000 type of reactors.

In TACIS SRR 2/95, SCK•CEN helped to define the test matrix, reviewed the mechanical testing results and performed the review of the fluence measurements and analysis. This information will be used as background for the correlation and modelling activities foreseen in 2001.

In TACIS R2.06/96 we want to improve the surveillance analysis methodology of the VVER-1000 reactors. SCK•CEN co-designed the instrumented surveillance capsule that was irradiated in Balakovo unit 1 (Russia), did the selection of irradiated specimens from existing surveillance programmes and, evaluated and reviewed the results of the testing and analysis according to the master curve, originally performed by Russian laboratoria. In 2001, corrective actions and/or alternative surveillance strategies will be considered and a number of recommendations will be made.

Support to Infrastructure

Hot Cell equipment exploitation

The hot-cell equipment exploitation team at RMR is a multi-disciplinary flexible group that is central to the department and to SCK•CEN. It has several important tasks:

- managing and maintenance of the cells in operation;
- decontamination and removal of out-of-service equipment and preparation of the implantation of new research equipment;
- decontamination and dismantling of irradiated experiments and transport of the experiment to the PIE installations;
- active support for the research projects and teams;
- waste treatment and packaging for storage at the waste plant.

In 2000, we concentrated the main activities on:

Decontamination of:

- a hot-cell for polishing of fuel samples;
- burst test equipment for fuel cladding research;
- a mechanical hot-cell for fuel sampling on plates or rods.

Decommissioning of:

- a mechanical test hot-cell for cladding materials;
- chemical-reprocessing equipment for uranium targets.

New equipment:

- the build-up of a IASCC-equipment in the corrosion hot-cell;
- study for the installation of the re-instrumentation equipment in the chemical reprocessing hot-cell.

Dismantling:

We partially dismantled a sodium loop for study of local blockage in pre-irradiated fuel bundles. We separated the fuel bundle from the loop and the sodium between the fuel rods was melted out and recovered in a storage drum. The fuel bundle is ready for reprocessing.

Conditioning of fuel rods from fuel research programs:

The non-intact fuel rods were retrieved from their temporary storage under water and conditioned in a welded canister, provided with a Zn-tubing basket. The leak tight canister will be loaded into a Castor BR3 container, used for a long term dry-storage and transported to the waste plant.

Partners

-	Electrabel (Brussels, Belgium)
-	Laborelec (Linkebeek, Belgium)
BN	BELGONUCLEAIRE (Dessel, Belgium)
EDF	Electricité de France (Paris and Fontainebleau, France)
EPRI	Electric Power Research Institute (Palo Alto, USA)
HRP	Halden Reactor Project (Halden, Norway)
JAERI	Japan Atomic Energy Research Institute (Ibaraki and Tokaymura, Japan)
KULeuven	Katholieke Universiteit Leuven (Leuven, Belgium)
ONDRAF/NIRAS	Belgian Agency for Radioactive Waste and Enriched Fissile Materials (Brussels, Belgium)
RUCA	Rijksuniversitair Centrum Antwerpen (Antwerp, Belgium)
TEE	Tractebel Energy Engineering (Brussels, Belgium)
TUI	Transuran Institut (Karlsruhe, Germany)
VTT	Technical Research Centre of Finland (Helsinki, Finland)
VUB	Vrije Universiteit Brussel (Brussels, Belgium)

Sponsors

TEE	Tractebel Energy Engineering (Brussels, Belgium)

Customers

-	Electrabel (Brussels, Belgium)
BN	BELGONUCLEAIRE (Dessel, Belgium)
CTMSP	Centro Tecnológico da Marinha em São Paulo (São Paulo, Brazil)
EU	European Commission (Brussels, Belgium)
PSI	Paul Scherrer Institute (Villigen, Switzerland)
TEE	Tractebel Energy Engineering (Brussels, Belgium)

Scientific Output

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