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Fusion Reactor Materials

Background

Fusion Research in Europe is one of the largest co-ordinated programmes of the European Commission. SCK•CEN focuses its fusion participation on material characterisation under radiation and coolant interaction: structural material for the vessel modules, as well as material used for the diagnostics and maintenance instrumentation. This last aspect is presented in the contribution "Advanced Instrumentation and Teleoperation" of the present report. The aspects related to waste management and decommissioning strategies are also a new field where SCK•CEN valorises its expertise into the fusion reactor issues.

Objective

To contribute to the knowledge on the behaviour, during and after irradiation, of the fusion-reactor materials and components.

Programme

- Study of the mechanical behaviour of structural materials under neutron irradiation: steels, inconel, molybdenum, chromium, involving tensile and compact tension tests, microstructural evaluation and corrosion measurement;
- Modelisation of the characteristics of irradiated first wall material such as beryllium: swelling, fracture behaviour, reactivity with air and steam.
- Detection of abrupt electrical degradation of insulating ceramics under high temperature and neutron radiation;
- Study of dismantling and waste disposal strategy for fusion reactors;
- Feasibility study for the testing of blanket modules under neutron radiation.

Achievements

Stainless Steel: Low cycle fatigue of irradiated AISI 316L(IG)

AISI 316L stainless steel has been selected as the main structural material for the International Thermonuclear Experimental Reactor (ITER) fusion device. Although this steel was extensively investigated, most results concern irradiation temperatures above 300°C. Here, tensile and fatigue specimens of plate and TIG (Tungsten Inert Gas) weld materials were irradiated in the BR2 reactor at 42°C up to a neutron fluence corresponding to 5.4 dpa. Low-cycle fatigue tests were performed at room temperature according to the prevailing standards. By comparing our results with published data on the same AISI 316L reference plate, we observed that fatigue life was almost not affected by irradiation and testing conditions (Fig. 1). The same conclusion applied to the TIG welds. It should be noticed that, by contrast, the tensile properties are very much affected by irradiation. The main difference between unirradiated and irradiated tests lies in the way the applied total strain is distributed into a specimen in terms of elastic and plastic strains. The unirradiated specimens exhibit a permanent deformation, while irradiated ones are loaded in the elastic regime. Comparing low cycle fatigue life results, one should not only look to the total strain, but also to its elastic and plastic parts.

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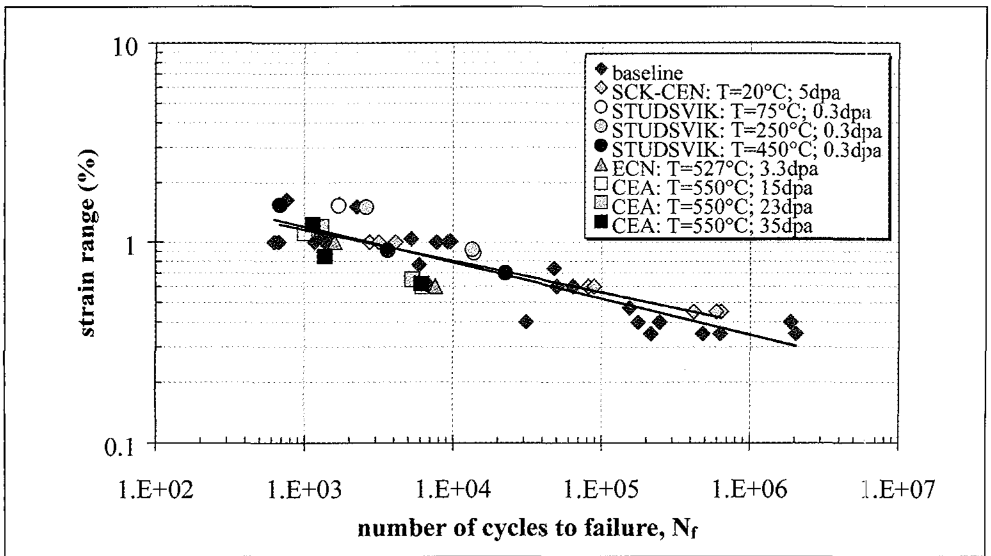


AISI 316L-fatigue life is not affected by irradiation. SCK•CEN data are in good agreement with STUDVIK, ECN and CEA data although experimental and irradiation conditions are very different

Reduced activation ferritic-martensitic (RAFM) steels

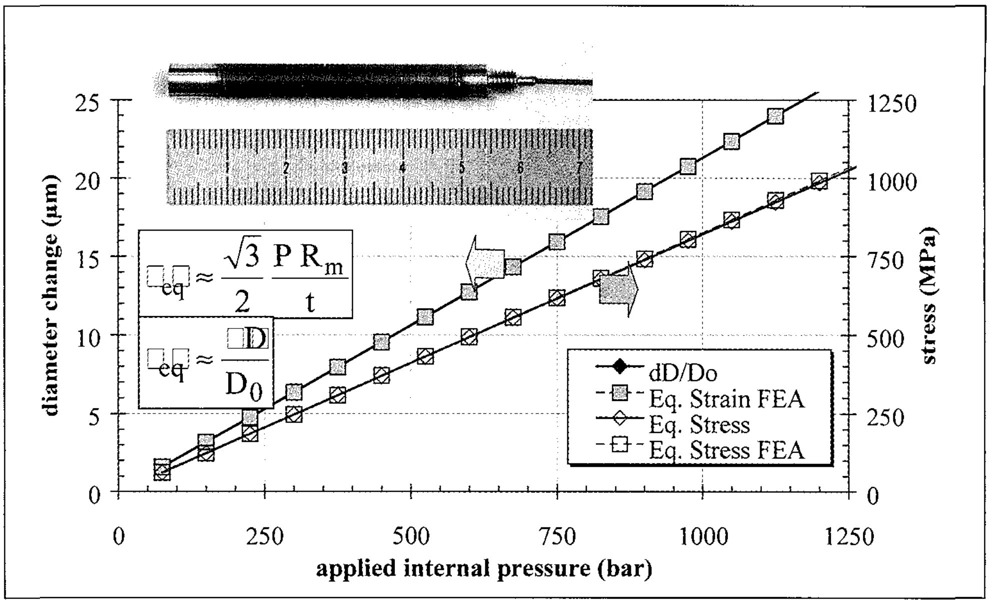
A first study, mostly bibliographical in 1999, started on a new reference industrial RAFM steel called EUROFER'97. Its chemical composition and improved properties reflect the results of previous characterisation tests, including irradiation assessments. Future irradiation tests in collaboration with other European partners have been prepared. The study was also extended to ODS (Oxide Dispersion Strengthened) RAFM steels. Information on the most effective production routes of commercially available alloys has been gathered. A literature review was carried out for instance on the effect of Cr content on the mechanical properties. This study used also the past experience at SCK•CEN on ODS alloys production for fast breeder application.

Corrosion of stainless steels: Detection of crack-initiation and crack-growth

Irradiation assisted stress corrosion cracking (IASCC) is considered as an important issue for cooling channels made out of austenitic stainless steel. In order to assess IASCC under fusion relevant parameters, instruments have to be developed for the on-line monitoring of crack-initiation and crack-propagation. By measuring the spontaneous current or voltage fluctuations (electrochemical noise) between a reference electrode and a test sample (e.g. a pressure tube), one can follow what happens at the surface of the electrode. The sensor used in the present study consists of two Pt wires, placed for instance at two diametrically opposite positions on the pressure tube, connected to each other but insulated from the pressure tube. The pressure inside the tube is gradually increased above the material yield strength, thereby creating micro-cracks on the surface. Spikes appearing on the measured potential indicated the initiation of each crack. Electron microscope examination confirmed the observation after the test. Another technique, called acoustic emission, has also been studied for the on-line detection of cracks. Whenever they appear, mechanical elastic energy is released in the form of a transient high frequency acoustic pulse. We studied how to guide this acoustic signal from the strained sample, inside the autoclave, to the outside by means of a metal bar (acoustic waveguide), where the detection can be performed by commercially available sensors. The measurements were performed in a static autoclave, equipped with a slow strain rate test unit. They showed clear signatures of crack initiation and propagation, long before the actual breaking of the sample.

Inconel: In-reactor radiation induced creep

The ITER shield modules will be attached to the backplate by four radial supports, and connected to it by a pair of electrical straps, all of which fixed by Inconel 718 bolts. They are exposed to a moderate neutron fluence (less than 1 dpa) at elevated temperature (300 to 350 °C). To deal with operational loads, these bolts are pre-stressed during assembly.



Relation between the internal pressure and the resulting stress strain state in the pressurised tube.

However, during operation, the bolts lose part of the pre-load due to stress relaxation, mostly as a result of radiation induced creep. Almost no data are found in the literature on the dimensional stability of Inconel 718 under neutron irradiation. An experimental assessment in BR2 was therefore started, with material irradiated under the typical loading conditions of the future ITER bolts. The tests were performed at 300°C up to 0.5 dpa, the specimens being pre-stressed to around 80 % of their yield strength. A pressurised thin-walled tube geometry was selected as the most adequate sample geometry . The tubes were first laser-welded to end plugs and then filled with argon at liquid nitrogen temperature. The volume of argon is selected in such a way that the desired pressure is reached at the operation temperature of 300°C. The figure below shows the relation between the stress-strain state and the applied internal pressure in the tube. The increase in tube diameter is used to determine the irradiation creep induced stress relaxation.

After completing all pre-irradiation characterisations (precipitation hardened heat treatment, tensile properties, hardness, optical and transmission electron microscopy), the samples were irradiated up to 0.52 dpa. Un-irradiated pressurised tubes were also aged at 300 °C for a period equivalent to the period spent in the reactor. No significant change of dimensions could be detected. Post-irradiation measurements will be performed early 2000.

Molybdenum: Analysis of tensile and fracture toughness on irradiated alloys

Plasma facing components will be submitted to high heat and radiation fluxes and disruptive loading. Material choice is dictated, among other criteria, by tritium permeation, plasma - wall interactions, erosion resistance, thermally induced fatigue, swelling and embrittlement, creep properties, thermal ageing, neutron activation, biological hazards, fabrication capabilities and cost. Although the interest of molybdenum has recently decreased in the fusion community, this material remains interesting for its high melting point, its high mechanical resistance at elevated temperatures, its low thermal expansion coefficient and its good thermal conductivity, which results in an excellent dimensional stability and a good resistance to thermal shocks. The tensile properties for candidate divertor armour Mo-alloys are now well established. However, the assessment of the radiation damage on the tensile, fracture toughness and fatigue properties still requires a large effort. In this context, tensile and toughness properties of two Mo-alloys (TZM and Mo-5%Re) were investigated at SCK•CEN. Specimens have been irradiated in the previous years at 40°C and 450°C, up to approximately 0.2 dpa. The selected alloys show good tensile strength and ductility in the baseline condition, but a drastic loss of ductility after irradiation. The lower irradiation temperature induces a higher reduction of ductility. Fracture toughness tests were also performed on pre-cracked and notched specimens. In baseline condition, the fracture toughness increases with temperature and the fracture mode is cleavage. Although a drastic diminution of ductility is observed on tensile test results due to neutron irradiation, the fracture toughness decreases only slightly for the two temperature conditions. This low irradiation embrittlement observation is very interesting, but an extrapolation of this behaviour to higher fluence values would require additional validation.

Chromium: Characterisation under neutron radiation

Chromium is another potential candidate for plasma facing components. Although the material is known for being quite brittle, tests were performed for particular alloys with potentially improved characteristics. A mechanical characterisation of two different Chromium alloys, called DUCROPUR and DUCROLLOY was performed, looking to the influence of heat treatment and neutron radiation, up to 0.5 dpa. The following tests have been performed and analysed on unirradiated and irradiated samples: tensile tests, impact tests up to 1100 °C (on sub-size and full-size Charpy-V specimens); static notch toughness tests up to 500 °C (on sub-size Charpy-V specimens). The results were compared with literature data and with complementary experiments performed at ESI in Austria.

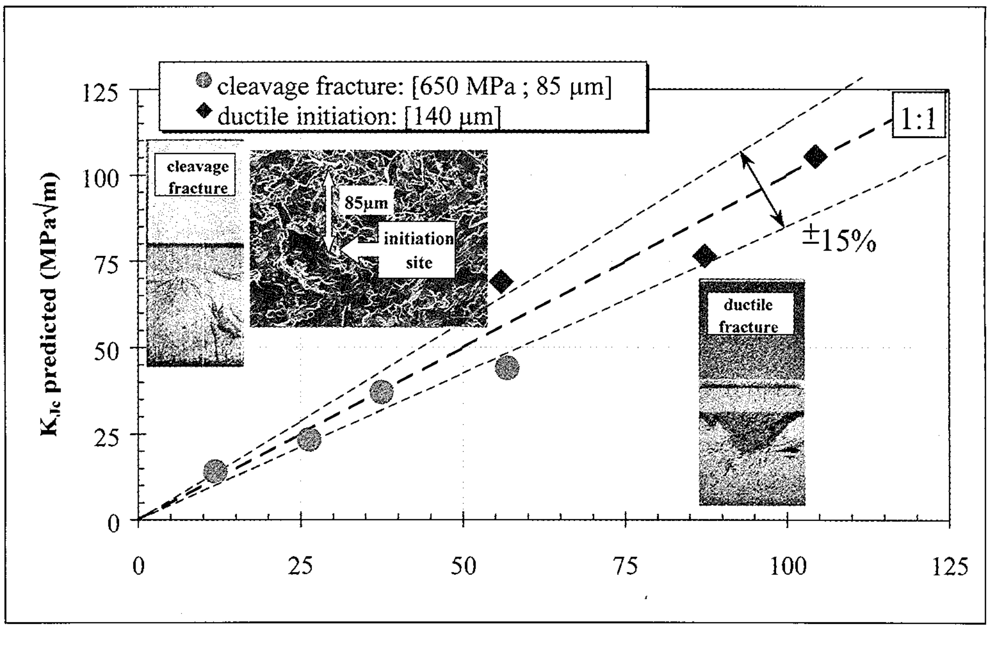
Beryllium: Integrity test and swelling assessment after annealing.

Severe beryllium embrittlement has been reported after high heat loading. It was therefore needed to further investigate the integrity of irradiated beryllium when exposed to high heat fluxes. Three types of irradiated beryllium were available at different total fluences (two irradiated at 230 °C up to 600 appm He, the third at 50 °C up to 3900 appm He). They differ by the BeO content (0.6 %, 1.2 % and 1.9 % respectively). The samples were submitted to a thermal shock at 600 °C and 700 °C (heated up in vacuum within 30 minutes, keeping the temperature for 5, 27 and 100 hours). Fortunately, no sign of disintegration was observed on any sample. A visual examination revealed all beryllium samples showing the same solid geometry. Even drop tests didn't produce any fracture. The corresponding swelling was also assessed, using density measurements before and after annealing. More significant swelling was observed at 700 °C than at 600 °C, and higher BeO content tends to give lower swelling. On the other hand, annealing time has a negligible effect on the swelling.

Beryllium: Fracture behaviour

Few experimental data on the fracture toughness behaviour of beryllium are available, especially for irradiated condition. Although tensile tests are easy to perform and widely used for investigating the radiation effects on materials, there is a need for further study to correlate the tensile properties with the fracture toughness. During previous years, tensile and fracture toughness data were produced for four fusion-grade beryllium alloys. They were irradiated in the BR2 reactor at various temperature and neutron fluence conditions. Tests were performed in a large temperature range from brittle to ductile fracture. Such a large data set offered a good opportunity to investigate the above-mentioned correlation. The focus was put on the S65 VHP alloy, being seen as the reference ITER material now. Compact tension specimens were systematically investigated with scanning electron microscopy (SEM). In the baseline condition, two fracture mechanisms were identified: cleavage when tested below 300°C, ductile above 300 °C. In the irradiated condition, all samples fail in a brittle manner, with a tendency to intergranular fracture for specimens irradiated at 600°C. The basic idea of micro-mechanical modelling is to take into account the fracture micro-mechanisms of fracture in association with the stress and the strain history during the whole loading process. The most popular model used to predict cleavage fracture toughness is the Ritchie-Knott-Rice model (1968). It considers that fracture occurs when the maximum principal stress reaches a critical micro-cleavage stress value over a critical distance ahead of the crack tip. The figure below shows a typical fracture surface and how one locates the fracture initiation site on a SEM micrograph. Using finite element calculations, it is possible to derive the local maximum principal stress at the initiation site.

For ductile fracture, the Rice and Tracey void growth model is used. As for cleavage fracture, two parameters are required: the critical void growth rate and again the characteristic distance. The critical void growth was approximately evaluated from the tensile tests. The characteristic distance, derived from the finite element calculations, is found equal to ~140 µm. Within experimental and numerical uncertainties, both models accurately predict the fracture toughness in cleavage as well as ductile fracture mode



For both cleavage and ductile initiation, both micromechanical models predict well the fracture toughness. SEM micrographs showing the fracture surface of two samples that failed in cleavage and ductile modes are also included.

Beryllium: Interaction with air and steam

A safety study was conducted concerning the use of beryllium in a fusion reactor. For the assessment of accidental scenarios such as a loss of vacuum accident or a loss of cooling accident, kinetic data are needed on the beryllium/air and beryllium/steam reaction respectively. We studied these reactions with combined thermo-gravimetry and differential thermal analysis. Three different materials were studied: unirradiated S-200F, the same material irradiated up to 300 appm He, and S-200E (higher BeO content) irradiated up to 21000 appm He. In air, all beryllium types exhibited parabolic oxidation kinetics at 600°C within the duration of the experiments (six hours). At 800°C and 1000°C, the reaction accelerated and became linear until the specimen was depleted of unreacted beryllium. At 700°C, the dense samples (unirradiated and low fluence irradiated) showed parabolic behaviour while the reaction accelerated for the high fluence irradiated porous samples. The irradiated samples showed a higher oxidation rate in air than the unirradiated specimens, except at 600°C. There was no pronounced influence of porosity on the reaction rate, but this is probably due to the relatively high theoretical density of the irradiated porous Be samples. In steam, kinetics was parabolic for all tested beryllium types at 600°C. At 700°C, kinetics was parabolic for unirradiated Be and irradiated dense Be, and accelerating/linear for irradiated porous material. At 800°C, all samples showed accelerating/linear behaviour. There was no influence of porosity on the reaction rate of beryllium in steam, except at 700°C, where the reaction rate for the irradiated porous samples is an order of magnitude higher than for the irradiated dense material. This can be attributed to the different kinetic regimes for the two beryllium types at 700°C. No pronounced influence of irradiation on the reactivity of beryllium in steam was observed.

Ceramics: A neutron benchmark to detect Radiation Induced Electrical Degradation (RIED) Effect

Ceramic insulators will be used in the fusion reactor vessel wall as part of the heating, current drives and diagnostics systems. These insulators will be subject to neutron fluxes and high temperatures. It is thought that the electrical characteristics of most insulators could then undergo a two-step degradation process. After a slow decrease in insulating resistance, a sharp breakdown (RIED) is observed when a given dpa damage is reached. Such a sudden degradation must be avoided during reactor operation. At present state, only fragmentary results are available, in terms of neutron fluence, energy and flux, as well as temperature and sample material. The objective of the work is to perform a neutron benchmark experiment to obtain more reliable values of the RIED threshold, under representative fusion conditions. A rig has been specially designed to work under vacuum, and the BR2 reactor will be operated at reduced power (5%). Irradiation conditions will be as follow: a neutron flux of 10^{+12} n/cm^2.s (E>0.1MeV), a gamma heating of 100 Gy/s, temperatures of 350 °C, 400°C and 450°C. The rig is mainly composed of modules, centred in a pressure tube to be introduced in the reactor and connected to a control panel for vacuum circuit, heating regulation, measurements and data acquisition. Electrical resistance measurement is particularly delicate, due to the very high resistivity of the ceramic samples and the high voltages involved. The rig is designed to be reloadable, using new modules and new samples, in case of future similar tests. The experimental device is presently under construction. Irradiation and data analysis is foreseen in the mid of 2000.

Waste strategy: Dismantling options

Present-day design of nuclear installations includes a careful planning of the decommissioning steps. This is now requested by national regulation bodies. It covers not only technical capabilities, but also financial analysis. In order to cover the nuclear liabilities, a cost estimate for the decontamination/dismantling phase is calculated, and suitable provisions are made during the lifetime of the facility. This should apply also to a fusion reactor. It is therefore useful to compare the impact of different decommissioning strategies on the management of activated and contaminated fusion materials. A detailed analysis of the decontamination/dismantling strategies and technical alternatives applied to a fission reactor was carried out. First, we identified the different techniques, the important tasks and procedures involved in a typical decommissioning operation in general. It was greatly inspired by the experience gained during the ongoing dismantling of the BR3 reactor in Mol. Specific features relevant to the fusion reactors and the major differences with fission reactors were pointed out, allowing to further focus the attention on these topics for further evaluation. The presence of tritium and the high-energy neutron activation, including concrete activation, were some of the most important specific topics identified, but the management of the important radioactive waste flow represents also a challenging aspect. Fusion decommissioning planning, specific materials and waste routes, waste characterisation and handling will gain from the experience gathered on present fission plant decommissioning projects.

Waste strategy: Analysis of human intrusion scenarios into a fusion repository site

Safety and performance analyses of a deep repository for the disposal of radioactive waste resulting from the operation of a fusion reactor had already been carried out by various European partners. A complementary study was performed here focusing on the analysis of future human intrusions into such a repository. We considered the case of the most active waste type that should result from the operation of a fusion plant for which low-activation martensitic steel is used as the main structural material. The repository was assumed to be located in the Boom Clay layer at the Mol site. The scenario analysis indicates that in the case of deep disposal, three variants of a borehole-drilling scenario can be considered: core inspection, residence and unsealed borehole. The analysis of the most drastic human intrusion scenario, i.e. the core inspection scenario, was highlighted. The maximum resulting dose is calculated for a beryllium coating. In the case of a routine inspection the dose to a geological worker is always under 0.5 Sv. This value, under which serious deterministic health effects are unlikely, can be considered as a dose limit for acute exposure. In the case of a close inspection, a maximum dose of 2 Sv is calculated if the intrusion occurs already 200 years after the disposal. The dose drops to under 0.5 Sv after 1,300 years. It should be noticed that the close inspection scenario can be considered as strongly conservative and somewhat contradictory, because it assumes on the one hand that advanced drilling techniques are available and on the other hand that the involved geological worker are not aware of the presence of radioactive materials in the core. Finally, a comparison with fission waste shows that in the case of disposal of spent fuel from a fission reactor, it should take one million years before the same scenario gives doses under this 0.5 Sv level.

Testing breeding blanket module components under radiation

Water Cooled Lithium-Lead (WCLL) is a candidate concept for the tritium generating blanket of future fusion reactors. In this concept, tritium is generated in a 17Li-Pb liquid metal bed, cooled by pressurised water circulating in Double-Wall Tubes (DWT). The permeation of tritium to water is limited by Tritium Permeation Barriers (TPB), which are a coating applied at the outer surface of the DWT. The DWT and the TPB constitute critical elements of the WCLL concept. Therefore, two experiments have been proposed to test the TPB performance and the DWT behaviour in realistic conditions. The experiment would help confirming the general behaviour of the components and in particular the tritium permeation rate into the adjacent structure. A decision on the possible realisation of these experiments is still to be taken. The preparation work was also used to design a multipurpose irradiation rig for testing fusion material (MISTRAL rig). It has as objective to allow the irradiation of metal samples (e.g. tensile and Charpy specimens) in the temperature range of 200 – 350 °C. The experiment is to be installed in the central hole of a BR2 fuel element, allowing fast dpa accumulation rate to be achieved. The rig proposal uses a single wall water capsule, with an electrical heater for temperature control.

Partners

CIEMAT	Centro Investigaciones Energéticas, Medioambientales y Tecnologicas (Madrid, Spain)
FZK	Forschungszentrum Karlsruhe (Karlsruhe, Germany)
ULB	Université Libre de Bruxelles (Physique Statistique et Plasmas) (Brussels, Belgium)
VITO	Vlaamse Instelling voor Technologisch Onderzoek (Mol, Belgium))
ESI	Erich Schmid Institute of Material Science (Leoben, Austria)

Sponsor

EC	European Commission through its European Fusion Development Agreement (EFDA) and its Underlying Technology Programme (Brussels, Belgium)

Presentations

R. Van Nieuwenhove, "Electrochemical Noise Measurements under PWR conditions", Third IAEA WACOL meeting (High Temperature On-line Monitoring of Water Chemistry and Corrosion), Helsinki (Finland), June 6-9, 1999.

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R. Chaouadi, M. Scibetta and A. Leenaerts, "Micromechanical Modelling of Fracture Initiation of Beryllium", Fourth IEA International Workshop on Beryllium Technology for Fusion, Karlsruhe (Germany) September 15-17, 1999.

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