Plasticity / ductile rupture interaction in irradiated austenitic stainless steels

Jean-Michel SCHERER
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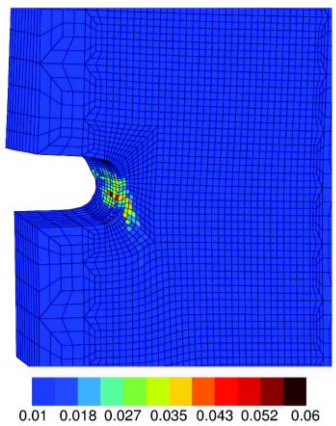
Industrial Partner: CEA
Supervisors: Samuel FOREST, Jacques BESSON, Benoît TANGUY, Jérémy HURE



Void volume fraction f in the region near a notch tip at a/a0 = 0.95 predicted by a porous single crystal model with the initial porosity f0 = 0.01

- Single crystal plasticity
- Strain gradient plasticity
- Void growth and coalescence
- SEM in-situ tensile testing
- Irradiated stainless steels

Abstract:

Due to their proximity to the reactor core, structural materials in nuclear power plants undergo high level irradiations which could may hundreds of displaced atoms per incident projectile (dpa). The proof of ability to exceed lifetimes over 40 years needs to be established by performing predictive simulations of the weakening mechanisms induced by such large levels of radiant exposure.

It is well known that these steels show a very good ductility and resistance to the propagation of the ductile rupture. After irradiation and under loading, austenitic stainless steels microstructures evolve resulting in the modification of mechanical properties. We may cite the irradidation defects (Franck loops and precipitates) which are responsible for the hardening, an intergranular segregation which in-fine induces a weakening of grain boundaries, and germination and growth of intergranular cavities and/or Helium and Hydrogen bubbles (swelling) which may affect plasticity and the mechanical properties.

The aim of this study is to pursue and extend the work of two previous thesis in order to study and model the interaction of the localisation of the plastic flow, peculiar to irradiated materials, and the growth and coalescence of cavities, which are the key mecanisms of ductile rupture in these steels.