Reactor Pressure Vessels
The Belgian Enhanced Surveillance Technology

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ABSTRACT

An enhanced reactor pressure vessel surveillance technology is being developed in Belgium. The aim is to determine in-service shifts of static initiation fracture toughness in a more direct and precise manner than possible through the current indexation procedure to the CVN impact test. The technology entails the reconstitution, precracking and slow bend testing of broken CVN remnants, and emphasizes the maximization of information by combining static tensile and CVN load-displacement traces to the PCCV traces. Damage modelling based on dislocation theory and micromechanical modelling provide the interpretation tools allowing to justify plant-specific embrittlement trend curves.
This enhanced surveillance technology is being actively validated.

1 Introduction

The vessel integrity of the nuclear power plant (NPP) is investigated with reactor pressure vessel (RPV) surveillance programmes. Reactor surveillance capsules are positioned in the beltline region towards the inner surface of the nuclear reactor vessel. Every capsule contains Charpy-V bars (CVN), tensile specimens and fracture mechanics samples, prepared from the actual materials of which the vessel was manufactured. The specimens, located closer to the core than the vessel wall, experience accelerated exposure and are representative for the irradiated vessel at a later time. All retrieved surveillance capsules of the Belgian NPP's are tested at SCK•CEN, the Belgian Nuclear Research Centre.

The test matrix within the capsule scopes the future embrittlement status of the reactor pressure vessel. The actual information on the vessel embrittlement, evaluated by nationally accepted regulations and standards, indicates that a significant life extension of the vessels would be possible. As a consequence, a modified withdrawal scheme for the remaining capsules in the Belgian NPP's was adopted. This schedule allows to investigate the embrittlement status beyond the design end-of-life date.

As the regulation to assess RPV embrittlement is based on empirical arguments, it is necessary to try to understand the underlying physical processes responsible for material degradation.
SCK•CEN and Tractebel Energy Engineering have developed an Enhanced Surveillance Strategy encompassing upgraded fracture toughness mechanics testing, microstructural interrogation and advanced modelling.
This methodology is the result of more than 20 years of research and development.

2 Classical surveillance

The bulk of the information used for the RPV surveillance programme comes from the CVN impact tests that reveal, as a function of irradiation fluence, the evolution of the RPV transition curves (energy, lateral expansion, shear fracture appearance (SFA)).
As is well known, two effects due to irradiation occur. The transition curve shifts to higher temperature and in the US regulation applied in Belgium, this shift is measured at the 41 Joule energy level. The second phenomenon on the Charpy plot is a lowering of the upper shelf energy: here regulation requires that this energy value does not decrease under 68 Joule throughout the vessel lifetime.
The safety evaluation of pressure vessels makes, however, use of the fracture toughness K, which is the actual resistance of the material against crack initiation and propagation.
Regulation uses empirically fitted lower bound K_{Ic} initiation and K_{Ia} arrest fracture toughness curves, as defined in the ASME XI code.
Regulation shifts these curves with irradiation according to the CVN 41 Joule shift.
This methodology entails however significant uncertainties susceptible to penalize plant operation flexibility and life management decisions. The enhanced surveillance strategy developed in Belgium allows to alleviate these drawbacks.

3 Enhanced surveillance

The "Enhanced Surveillance" concept is synthetized by the block diagram on Fig. 1.
"Classical" suveillance contains more information than has been exploited so far: we refer to the load-time traces from instrumented CVN tests, which allow to construct load-temperature diagrams and to directly integrate the tensile test data [1, 2].
Reconstitution technology [3] allows to prepare new test specimens from CVN remnants. In particular, welding of end tabs to cubes of 10x10x10 mm is used to make precracked Charpy-V specimens (PCCV). These PCCV samples can be impact tested to obtain K_{Id} or can be tested quasi-statically (by 3 point bending) to obtain K_{Jc}. New ASTM standards are in final preparation for the conduction and interpretation of such tests in the critical transition temperature range.
Microstructural interrogation of irradiated steels is essential to characterize both the fracture process and the in-service damage mechanisms. More direct visualization techniques as Scanning and Transmission Electron Microscopy are complemented by indirect methods such as Positron Annihilation, Internal Friction and Mössbauer Spectroscopy.
Advanced modelling is the analytical cement of the preceding experimental building blocks of the enhanced surveillance. It encompasses two related aspects:

- damage modelling based on dispersed-barrier theory, primarily aiming at quantifying irradiation effects on the flow properties (yield stress, work hardening) and on the microscopic fracture stress of the material;
- micromechanics-based link between steel matrix strengthening and fracture toughness using critical stress or triaxiality-modified critical strain criteria.

4 Enhanced surveillance demonstration

The enhanced surveillance technology is being actively validated. Important components in this effort are the BR3 vessel sampling and testing programme and the BR2/Chivas accelerated irradiation and testing programme to investigate radiation damage mechanisms incubation

effects.
Both projects will significantly contribute to the benchmarking of the micromechanics approach in advanced modelling and the fracture mechanics indexation tests based on small specimens. These topics are at the cutting edge of science and technology in the considered field.

The BR3 vessel sampling and testing

The BR3 vessel, manufactured by Babcock and Wilcox in 1958, is of particular interest in terms of the fracture toughness indexation issue. The BR3 plate is similar to the Yankee Rowe lower shell plate: it has been austenitized at 950-980 °C, was operated at a temperature of 260 °C and is nickel-modified. In 1984, this vessel was successfully wet-annealed at 343 °C during one week, after receiving a maximum neutron fluence of 3.3 E19 cm^{-2} (> 1 MeV) in 72000 EFPH. The operation license was renewed. The plant was shutdown in 1987 at 86000 EFPH and at a maximum neutron fluence of 4.0 E19 cm^{-2}.

The embrittlement of BR3 vessel materials (i.e. base and weld metals) displays "Outlier Behaviour" as compared to Regulatory expectations and to "Established" damage modelling. Namely, the BR3 materials' behaviour cannot be reliably predicted by current methods and is expected to be significantly more favourable than according to such predictions. This does primarily reflect the fact that the inadequacies of toughness indexation to Charpy-V data along current engineering methods are simply more pronounced for the BR3 materials than for most others. But the problem is also displayed by various industry steels, quite noticeably for instance by the welds at the Belgian Doel-1,-2 pressure vessels, and by the Yankee Rowe base and weld metals. It is most likely that both BR3 and Yankee Rowe plants would still be in operation to-day, had this toughness indexation issue been resolved at the time of their shutdowns. Also, we have recently shown that the vessel anneal option presently contemplated in the U.S. is probably avoidable for most plants, provided the much cheapier and more straightforward option of "Enhanced Surveillance" be implemented.

In fall 1994, SCK•CEN decided to sample and test the BR3 vessel. This was accomplished in January 1995 in cooperation with PCI Energy Services. The Electron Discharge Machining technique was used to remove 14 samples weighing approximately 10 kg each. The cuts, in the shape of boat samples extending to a depth of 7.4 cm in the 11 cm thick wall, were performed at vessel midplane and at nozzle elevation (see Fig. 2). In latter location, the neutronic exposure is negligible. Moreover, this material can be considered to be an adequate baseline, as a careful evaluation has demonstrated that thermal ageing effects are negligible as well.

It is interesting to compare our preliminary 41 Joule shift results to previous experiments and evaluation in Fig. 3.

The plate designated as YA9 was specially produced and heat-treated to simulate the Yankee Rowe A302-B upper shell plate (Yankee/BR3 surveillance). The plate YA1 was similarly designed to represent the Yankee Rowe nickel-modified lower shell plate and thus the BR3 vessel plate. Furthermore, fine grain versions of both YA9 and YA1 were also investigated, as well as versions corresponding to vessel surface conditions. Finally, the plate PT-A has the same chemistry as the BR3 plate, but has fine grains. All these plates were irradiated at 260 °C in test reactors. The full curve on the figure is a 1991 SCK•CEN evaluation [1], considered to be an upper bound for the K_{Jc} shift, and the 'star' is a similar ORNL/UCSB evaluation. The BR3 plate embrittlement is less than predicted, and most importantly, the lack of Ni influence for all these plates [1] is confirmed.

BR2/CHIVAS: Incubation and kinetics of RPV damage mechanisms

For some PWR plants, long term life management or life extension may entail neutron fluences in the range of 5 to 8 E19 cm^{-2} (> 1 MeV). There are very few surveillance data available at such large exposures. We have found [1] that for Western steels, a poorly characterized damage mechanism can cause a sudden increase of embrittlement, starting at an incubation fluence of 4-5 E19 cm^{-2}. This defect mechanism is rather stable, and is not affected by dose rate. It thus can be studied at the high flux materials testing reactor BR2, where the pressurized water loop Callisto is particularly suited for such task [4].

Fig. 4 gives a schematic view of the in-pile section into a 84 mm BR2 reactor channel. Three such sections are available.

The Chivas 1995 programme concentrates on 8 commercial steels of Belgian, French, German and US vintages. The main irradiations are done at 260 °C: this is the operation temperature of the BR3 and Chooz-A vessels. Therefore, direct comparison will be possible, not only with surveillance results, but also with the tests being performed on trepans extracted from the vessels themselves. Most interestingly, Charpy and tensile specimens cut from the BR3 plate at nozzle elevation have been loaded in BR2, while for Chooz-A, archive samples of the steam generator, representative of the base and weld metals of the reactor vessel, are similarly investigated.

In order to modify the relative importance of the various damage mechanisms and to help separate their contributions, most of these steels have also been irradiated at 290 °C and 150 °C. It is planned to observe their annealing responses, using mechanical tests of miniature CVN samples, positron annihilation, internal friction and Mössbauer spectroscopy, complemented by TEM.

In the longer term, CVN remnants will be reconstituted for PCCV fracture mechanics tests. This will be done primarily in the cases where we will predict that the effect of irradiation on the strain rate sensitivity is sufficiently different to allow the most demanding and full validation of 'Advanced Modelling'.

First results on these irradiated materials will be available in the second half of this year.

5 Video-recording

The video clarifies the questions that exist in classical surveillance, and demonstrates how the enhanced methodology can give guidance to understand complex embrittlement issues.

6 References

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Fig. 1 Simplified Conceptual Block Diagram of *"ENHANCED SURVEILLANCE"*

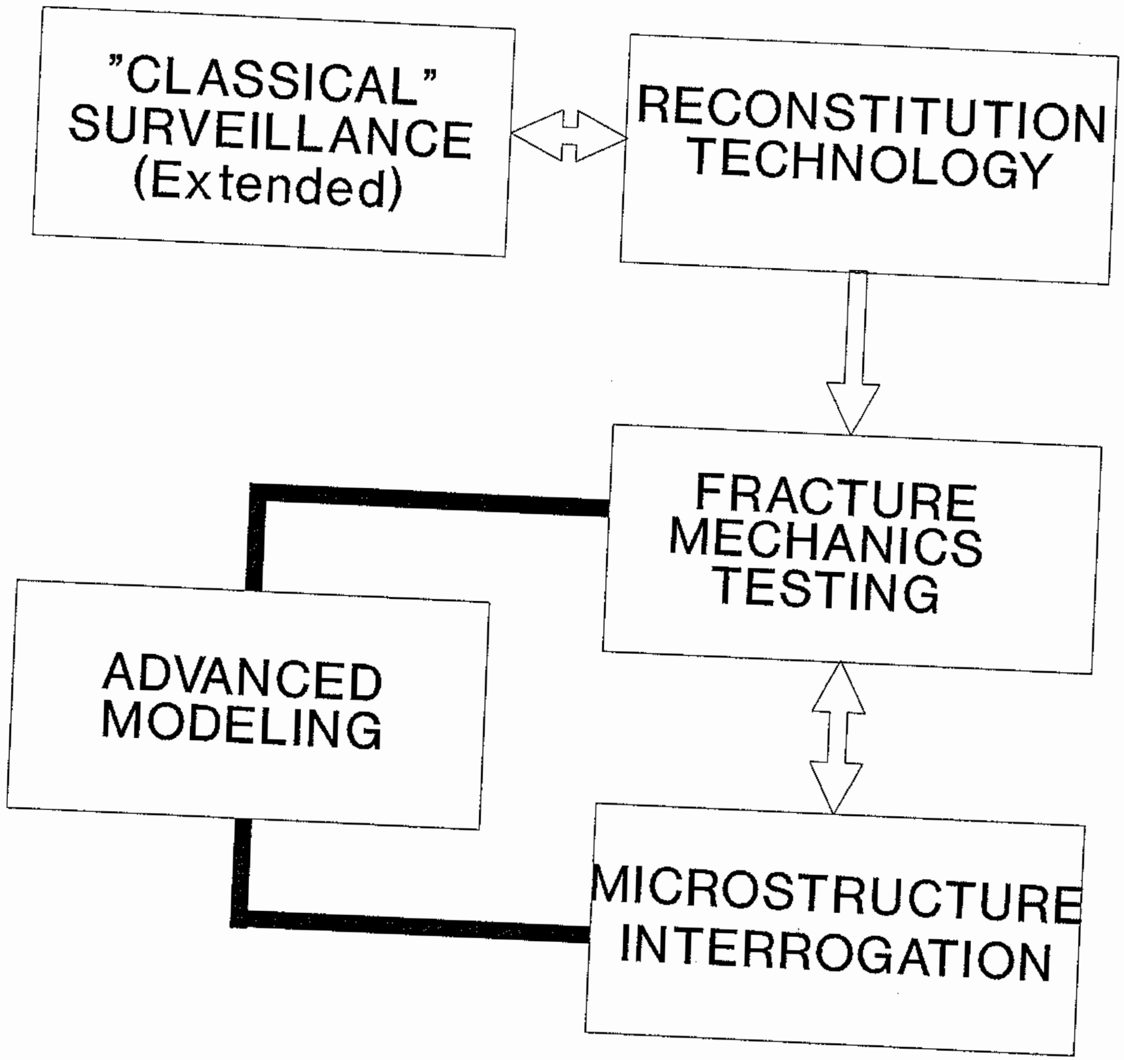
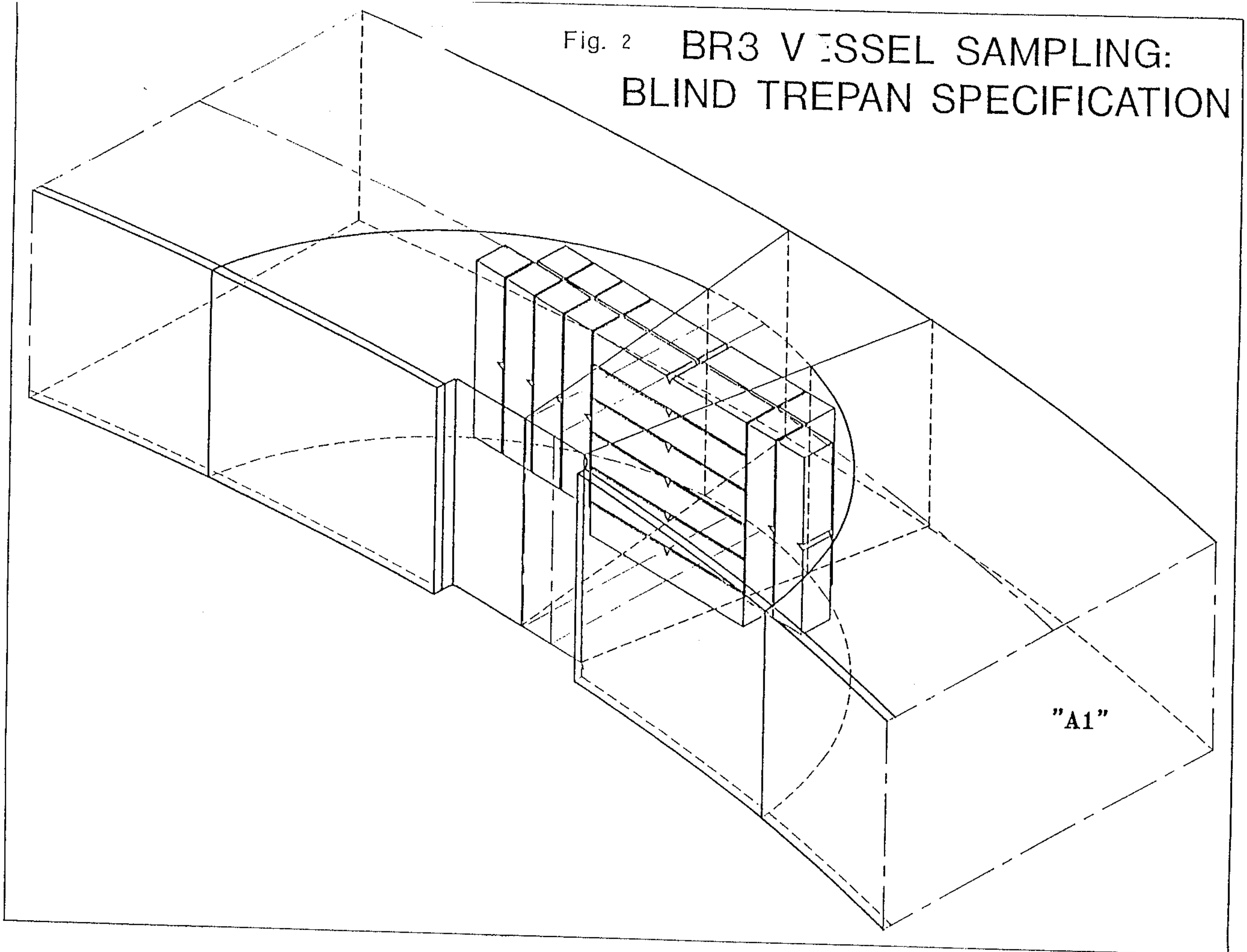


Fig. 2 BR3 VESSEL SAMPLING: BLIND TREPAN SPECIFICATION



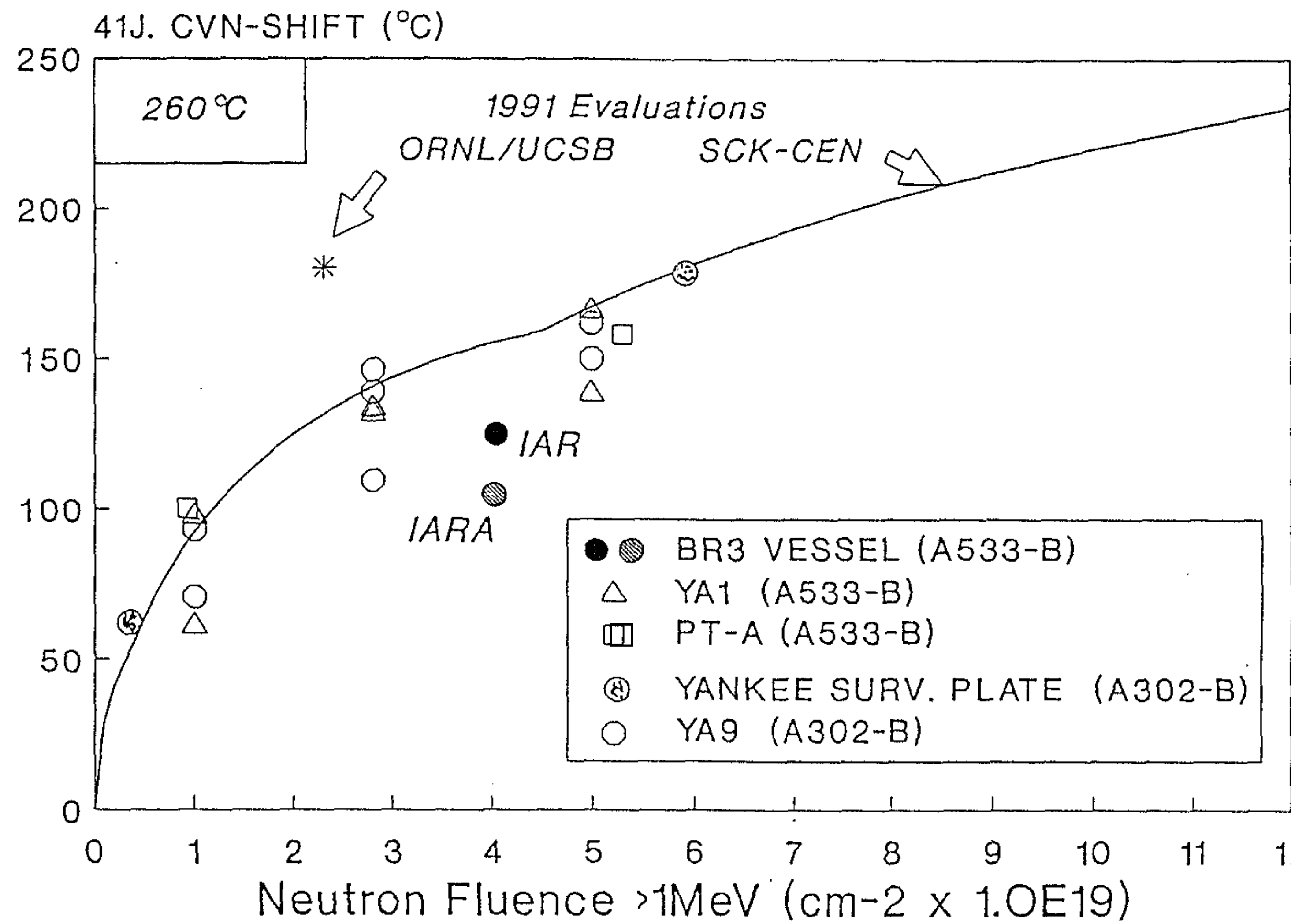


FIGURE 3 The BR3 plate embrittlement is less than predicted and lack of Ni influence is confirmed

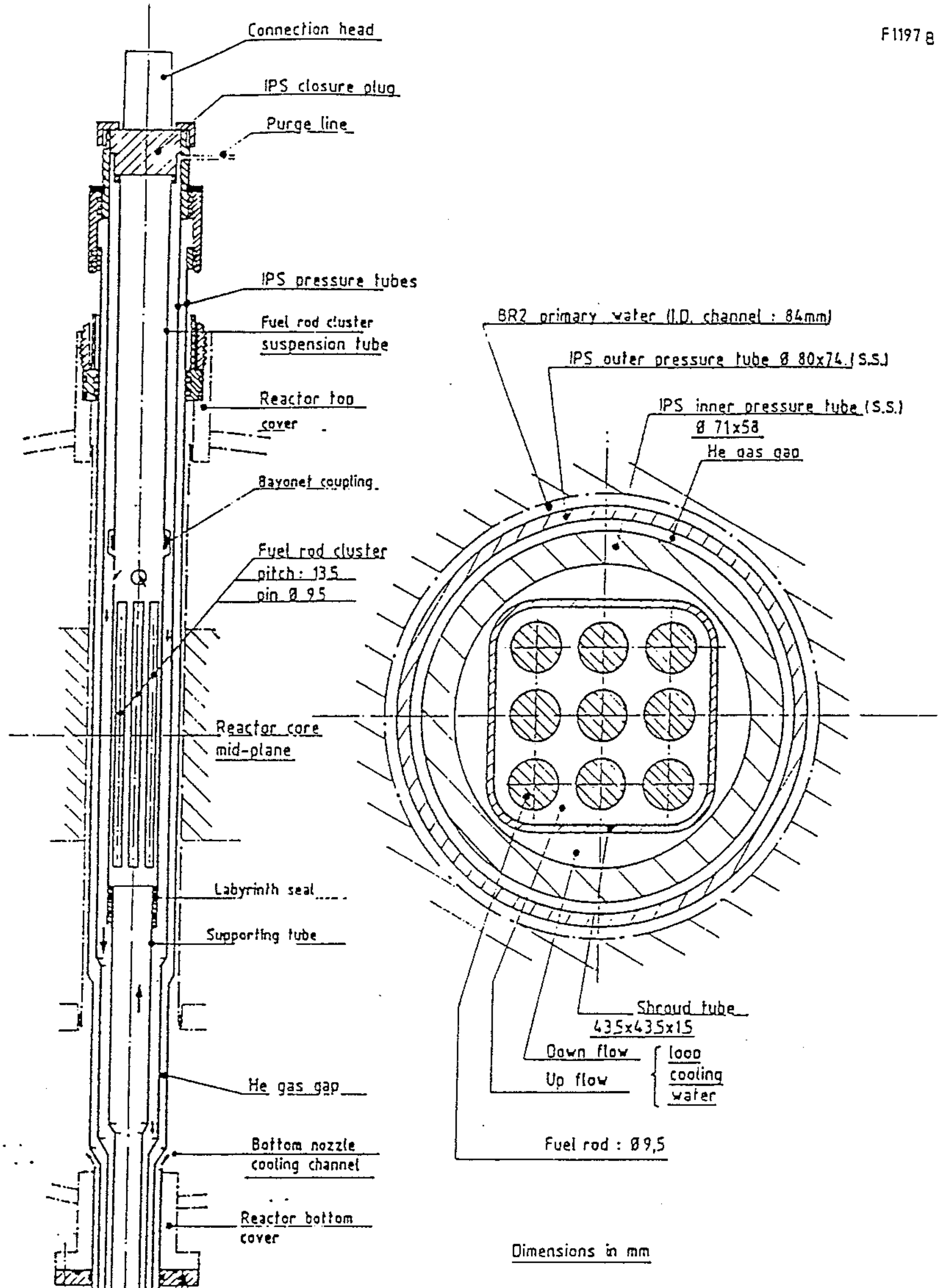


Fig. 4 CALLISTO
Schematic view of the in-pile section in a 84mm BR2 reactor channel.