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Need for Experimental Programmes on LOCA Issues Using High Burn-Up and MOX Fuels

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Safety studies performed in IPSN and elsewhere pointed out that high burn up might induce significant effects, especially those related with fuel relocation during LOCA situations. Uncertainties exist regarding how much the existing safety margins associated with peak clad temperature, clad oxidation, core coolability, clad residual ductility can be reduced by new fuels like the MOX one, burn up increases, the arrival of various alloys for fuel rod cladding. A better knowledge of specific phenomena associated to fuel effects is required in order to estimate the new margins and to resolve the pending uncertainties related to the LOCA criteria. Therefore, in addition to the programmes currently planned in the Halden reactor, IPSN is preparing the so-called "APRP-Irradié" (High Burn up fuel LOCA) programme. One of the important aspects of this programme is In-Pile experiments involving bundle geometries in the PHEBUS facility located at Cadarache, France.

1. INTRODUCTION

In France and in other countries, a permanent evolution of the light water reactors (LWR) is observed since the seventies. The evolution deals with the reactor designs (900 MWe/3 loops, 1300MWe/4 loops, N4, future EPR). It is also related to the fuel management and burnup increase (3 cycles, 4 cycles, 39[1] GWd/tU, 47, 52, 60 GWd/tU in the next future). This evolution affects the fuel itself (UO_2, MOX, Gd fuel), the cladding (Zircaloy, Zirlo, M5) and the control rods (Ag-In-Cd, B_4C). As a consequence of these modifications, there is a permanent need to reassess the reactor safety studies which implies improving the associated knowledge and upgrading the corresponding calculation tools. Such a need is not specific to the French situation. For the studies associated with the continuous evolution of the reactor operation, the safety authorities requirements are both related to the design basis accidents and the severe accidents. They have to appreciate to which extent their analyses and criteria might be modified by the burnup increase and the type of fuel. In France, under safety considerations, it was requested prior to any generic authorisation of discharge burn-up extension, that the high burn-up fuel behaviour be validated, with the support of appropriate R&D tests results, under accidental conditions, particularly under Loss-of-Coolant-Accident (LOCA) conditions.

The current regulatory safety criteria for LOCA, still in use in most countries, are derived from the ECCS acceptance criteria that were issued by USAEC in December 1973 and published in the Code of Federal Regulations (10.CFR50, part 50.46) as "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors".

The criteria are stated as 5 requirements, concerning the calculated performance of the cooling system under the most severe loss-of-coolant accident conditions. A summary of these conditions is given on the figure 1 against. These first two requirements address: the peak cladding temperature (PCT) which shall not exceed 1204°C and the maximum cladding oxidation rate, defined through an equivalent cladding reacted (ECR), which shall nowhere exceed 17% of the cladding thickness before oxidation but after cladding swelling with or without rupture.

TYPICAL LOCA TRANSIENT

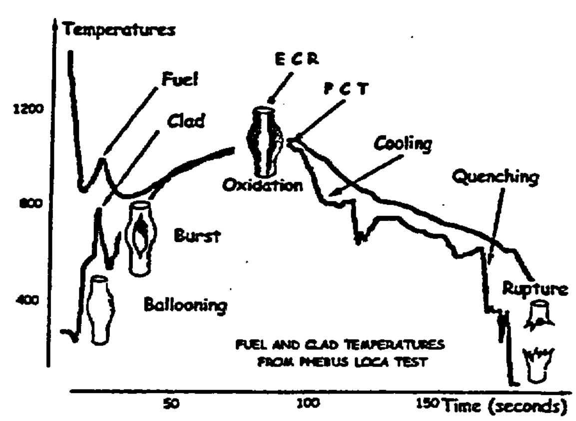

Figure 1

[1] Mean value per assembly

The third request addresses the maximum hydrogen generation, the total amount of which shall not exceed 1% of the hypothetical amount generated by the reaction of all the metal in the cladding surrounding fuel. Finally the last two requirements are related with core cooling. The calculated changes in core geometry shall leave the core amenable to cooling and after any operation of the ECCS, the core temperature shall be maintained at an acceptably low value and decay heat removed for the extended period of time required by long-lived radioactivity.

2. UNCERTAINTIES AND PENDING ISSUES

In the aftermath of the AEC LOCA criteria release, numerous studies were undertaken world-wide in order to improve the basic knowledge of the physical phenomena intervening in LOCA transients, so as to allow a better prediction with realistic models. Beyond the numerous experimental investigations that were conducted on unirradiated rods or cladding, either in-pile or out-of-pile, there exists a few number of available results of such experiments with irradiated material. Following is a very short review of the current knowledge on clad and fuel rod behaviour gained from experiments on irradiated material, that will introduce the pending questions and critical issues for irradiated fuel behaviour in LOCA.

2.1 UNCERTAINTIES

2.1.1 CLAD BEHAVIOUR

An important progress in knowledge relative to irradiated clad behaviour has been obtained from the results of the French EDF/IPSN [1,2] program (TAGCIR and HYDRAZIR tests), addressing the oxidation kinetics and quench bearing capability of irradiated zircaloy. The main outcome concern:

- the protective effect of corrosion oxide scale;
- the oxidation kinetics of irradiated zircaloy;
- the resistance to quench loads of irradiated zircaloy;
- the effect of high hydrogen content, as a result of internal hydriding during LOCA transient.

Relative to oxidation kinetics and quench behaviour, a comprehensive understanding of all involved phenomena and of their inter-related influences is not yet achieved and leaves still pending questions, most of them being not specific to high BU fuel. One important question is the influence on clad quenching resistance of axial constraints that may result from differential contractions upon quench between guide tubes and a fuel rod blocked in spacer grids as a result of ballooning or metallurgical interaction. Such blockage consequences had been evidenced on past tests at JAERI [3] on unirradiated rods and should therefore be expected to some extent on irradiated rods.

2.1.2 ROD BEHAVIOUR

There exists a few number of available results from experiments with irradiated fuel rods under LOCA conditions. The main outcome were found in results from the PBF-LOC tests[4,5] in the USA, the FR2 tests[6] in Germany, and the FLASH5 test[7] in France. They concern the fuel relocation process and an increased cladding deformation.

FUEL RELOCATION

All the available tests performed with irradiated fuel rods experiencing LOCA conditions have shown an accumulation of fuel debris in the swollen region –called balloon- of the burst cladding which resulted from fuel fragments slumping from upper locations (see figures 2 and 3 *below from FR2 results).*

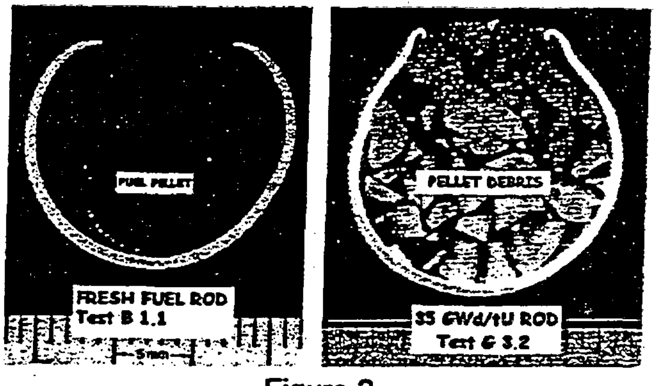

Figure 2

This process, here after called fuel relocation, is initiated at the time of the cladding burst, as demonstrated by the FR2-E3 and E4 tests. It is thought that the driving forces are both gravity and the pressure difference between the rod upper plenum and the channel.

This fuel relocation is not counteracted by the fuel-clad tight bounding, which exists with high burnup fuel.

Indeed, the process was observed in the FLASH-5 test with 50GWd/t fuel in spite of a rather low clad strain (not higher than 16%). It is thought that the cladding temperature increase combined with its ballooning suppress –at less partly- the bounding making possible the fuel relocation.

Finally, fuel relocation process is not specific of high burnup fuel. It was also observed for fuel rod having a burnup as low as 48MWd/t (LOC5-7B test).

AN INCREASED CLADDING DEFORMATION

In PBF-LOC experiments 2 couples of rods (2 unirradiated + 2 irradiated) were simultaneously tested in the same test train. Available data for comparison, although in very limited number due to technical problems, clearly indicate significant differences in the deformation behaviour of irradiated versus unirradiated rods.

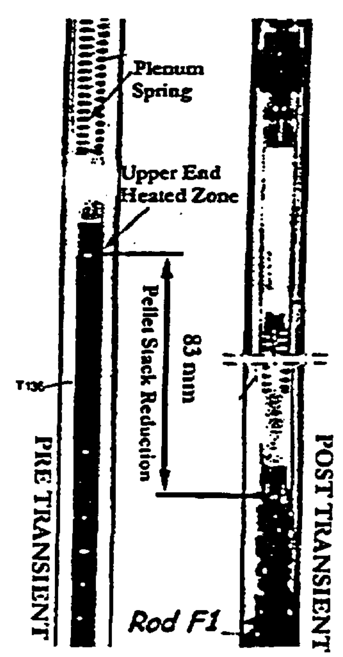

Figure 3

- A higher circumferential rupture strain for irradiated rods (a factor greater than 2 relatively to unirradiated rod strain for maximum values) and more axially extended;
- A wall thinning affecting almost all the circumference of irradiated rods, thus indicating low azimuthal temperature differences as compared to unirradiated rods.

These differences in behaviour have been attributed to the lower temperature differences on the clad of irradiated rods, circumferentially and axially, as a result of the pellet-clad gap reduction due to clad creepdown during rod irradiation.

2.2 THE PENDING ISSUES

A better understanding of the specific phenomena shortly mentionned above leads to raise a list of some complementary questions related with rod behaviour, fuel relocation process and coolability issue during LOCA transients.

2.2.1 ROD BEHAVIOUR

The question mark about rod behaviour is related with the influence of hydrogen pick-up and other irradiation effects on ballooning, burst behaviour and embrittlement during reflooding which were not considered when 10CFR50, part 50.46 was released.

2.2.2 FUEL RELOCATION

Several questions are induced by the relocation process. The first ones concern the process itself. The needed data are the following ones.

- Instant of fuel movement at high burn-up, with possible delay due to fuel-clad bonding.
- Filling ratio of clad balloon at high burn-up, with fragmentation of UO2 rim or MOX clusters
- Impact of the relocated material on steam access inside the balloon and hydrogen uptake rate.

The second set of question marks concerns the consequences of the relocation process.

- Which are the effects on peak clad temperature and final oxidation ratio of the local increase in lineic and surfacic powers and of the local decrease in fuel-clad gap resulting from fuel accumulation?

Note that these last issues are particularly important for end-of-life MOX fuel for which power generation is not reduced, unlike for UO2 fuel.

2.2.3 COOLABILITY

Related questions should be considered additionally, relative to flow blockage behaviour of highly deformed cladding with possibly relocated fuel and the embrittlement potentials associated to fuel fragmentation. The 90% value for flow blockage still coolable, as derived from results of flooding experiments (FEBA, SEFLEX et al) on unirradiated rods arrays is questionable since these experiments did not take account of any fuel relocation and associated effects. The needed information are the following.

- What is the maximum flow blockage ratio that leaves coolable an irradiated rods bundle?
- Does the maximum flow blockage ratio attainable with an irradiated rods array remain below the maximum coolable value indicated above?

There is presently a complete lack of data allowing to answer these questions.

- what flow blockage configuration would be worst coolable with occurrence of fuel relocation?

In other words, is the coplanar flow blockage still the worst coolable case?

3. THE IPSN *APRP IRRADIÉ* PROJECT

For many years, IPSN and several other safety organisations have applied a three-tier method for their reactor safety researches. The first step consists of computer code developments from the existing data bases. The second step involves small-scale, out-of-pile experiments, which provide the additional data bases requested by the code developments and their preliminary assessments. But, as the reactor phenomenology cannot be totally reproduced in such small scale experiments, a third step consisting of integral in-pile experiments using real materials is essential for comprehensive accident analyses. Their results allow the final code assessment in terms of reactor applicability and simulation completeness. This in-pile part of a programme assures that the investments done for code developments and small scale experiments will produce profits in terms of reactor safety. This three-tier method is applied by IPSN for the various research programmes devoted to reactor safety, design basis accidents including RIA and severe accidents programmes.

Regarding the LOCA issue, the current testing programmes dealing with irradiated material only involve out-of-pile experiments : separate effect quench tests on irradiated cladding (TAGCIR tests) in France; tests on irradiated cladding and integral type experiments (ballooning / burst / oxidation / quench) on irradiated rods at ANL (USA) [8] and JAERI (Japan) [9] with the support of an important programme of mechanical tests. In addition, OECD has planned an in-pile programme consisting of some single rod geometry tests with irradiated fuel. The programme should be conducted in the Halden reactor and should provide information about the relocated fuel characteristics.

But these programmes will not solve all the previously mentioned uncertainties because these ones are mainly associated with the combined behaviour of fuel and cladding under representative conditions of the reactor evolution during the LOCA transient. Based on the long fruitful experiences of a three-tier method, the so called *APRP Irradié* programme, providing the in-pile experiments third tier, should provide the missing part of the data bases required for code assessments in terms of reactor applicability and simulation completeness. This programme is prepared in a coherent way with the present international efforts in order to validate, and possibly update, the results obtained from separate effects tests and previous limited in-pile tests.

3.1 THE MAIN EXPERIMENTAL OBJECTIVES

The main objectives of the in-pile experiments will be to investigate the behaviour of fuel and cladding with conditions representative of the reactor during LOCA sequences. The main factors that will be accounted for are:

- the nature of fuel (UO2, MOX, Burn-up),
- the fuel-clad thermomechanical coupling (i.e. fuel relocation)
- thermal azimuthal gradients (main factor affecting cladding strain and blockage ratio)
- thermal-hydraulic aspects (i.e. quenching, coolability of blocked arrays)
-

3.2 TEST DEFINITION RATIONALE

The following analysis provides the rationale for the *APRP-Irradié* programme characteristics. It is shown that the conditions for having representative data for reactor applications are both in-pile tests and bundle geometry.

3.2.1 NEEDS FOR IN-PILE TESTS

The in-pile test need results from three reasons.

Neutron flux provides the unique way to produce the correct heat generation in the fuel fragments, corresponding to the residual power, whatever are the relocations induced by the ballooning and/or the burst of the rod. Both the exact amount of heat generation in the balloon and the heat exchanges with the rod channel depend on the characteristics of the relocated fuel fragments, their size, shapes, and compaction ratio. This heat generation correctness is one of the main conditions for having realistic estimates of the relocation consequences in terms of equivalent clad reacted, peak clad temperature and hydrogen uptake inside the balloon. All these aspects impact the strength of the rod during the quenching phase and the residual ductility of the rod after the LOCA transient.

During the blowdown phase of the LOCA transient, there is much less heat generation in the fuel and the clad coolant heat transfer is drastically reduced. Therefore, the fuel-stored energy is redistributed in the pellet and the cladding. Simultaneously, within a few seconds, this redistribution produces a decrease of the pellet center-line temperature from 1500°C down to, say, 1000°C and an increase of both the pellet rim and clad temperatures from 300°C up to 1000°C. Due to these temperature transients, the central part of the pellet will experience a contraction while the rim and the clad will undergo an expansion. Fuel mechanical stresses and fragmentation could be induced by these adverse effects. It has to be kept in mind that during usual experiments, for which a blowdown phase is not reproduced, clad and fuel temperatures are simultaneously increased or decreased without producing any comparable thermomechanical transient. In-pile tests including a blowdown phase provide the way to get a definitive answer regarding the additional fuel fragmentation prior to the relocation and how much this refragmentation process affects the amount and the characteristics of the relocated fuel.

Finally, during reflooding and quench process studies, in-pile tests allow to maintain the heat generation in the fuel corresponding to the residual power. By this way, more representative conditions of the thermomechanical loads of the rods are provided. Without such a power during the reflooding phase, steam production and cladding oxidation are reduced; the temperature transients experienced by the rods are less severe. Consequently, under estimates of core embrittling during reflooding could be obtained.

3.2.2 NEEDS FOR BUNDLE GEOMETRY

In addition to the requirement associated to heat generation mentioned above, bundle geometry is a second important condition to produce realistic data. Relocation being closely associated with the volume which is made free by the rod burst, it is clear that a correct amount of relocated fuel will be produced only if the sizes of the balloons are representative of the reactor conditions. Such balloon sizes can be obtained –as explained below- only with bundle geometry. This is the reason why these tests are essential and complementary of single rod tests.

During the early stage of the LOCA transient the fuel rods experience the ballooning and burst processes. For such phenomena, bundle geometry is a necessity to get a correct azimuthal temperature field around the fuel rods since this field is crucial to produce a realistic balloon size. An illustration of the impact of the azimuthal temperature field on the stain at burst is given by the results of single rod tests with heated or non-shroud. A uniformly heated shroud reduces the azimuthal temperature variation around the rod. For such tests a higher rod deformation is obtained. Conversely, an unheated shroud tends to increase the azimuthal gradient and, therefore, leads to small rod deformation at strain (see figure 4 below).

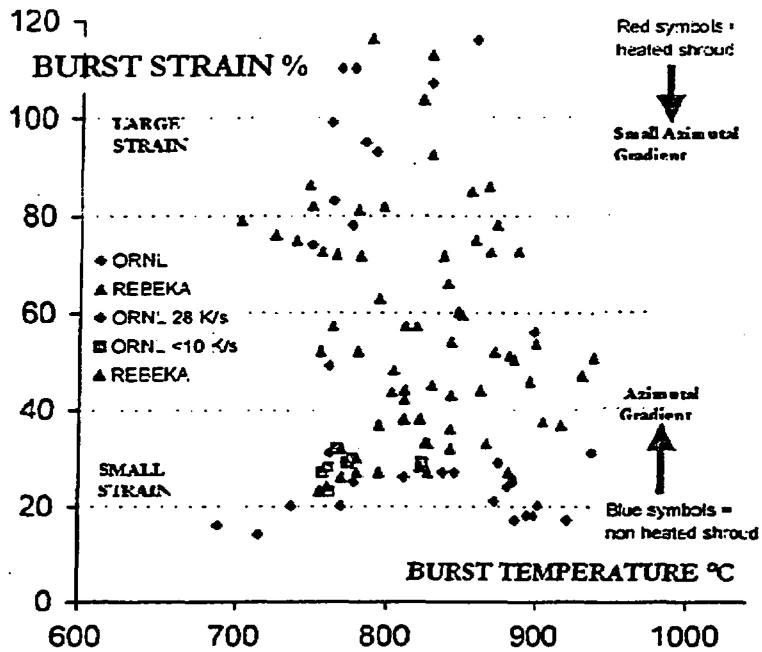

Figure 4

Additional reason for a bundle are the radial interactions between adjacent fuel rods that need to be taken into account because they modify the size and shape of the balloons. Such kind of balloon interactions are clearly illustrated with the side picture (figure 5) from PHEBUS LOCA test 215. Having in mind that the amount of relocated fuel is associated with the size and the shape of the balloon, the picture demonstrates that realistic data will require bundle geometry. This bundle geometry requirement to ensure representative mechanical interactions with neighbour rods was stated in early 80ies – several years ago- in consideration of ORNL MRBT B5/B3 experiments [10].

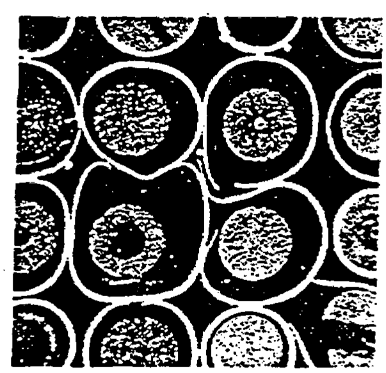

Figure 5

During core reflooding, a bundle is an obvious requirement for reproducing, on one hand, the correct flow blockage induced by the ballooning of the rods and their radial interactions and, on the other hand, the excess of heat generation at the blockage location due to the fuel fragments relocated in the balloons.

Finally, such bundle geometry is also necessary to represent the axial and radial stresses induced by the grids and the adjacent rods, which might restrain the rod contraction during quenching.

3.3 EXPERIMENTAL CONFIGURATIONS

Since it is hardly conceivable to carry out one type of experiments that will address all pending questions with any chance to provide some usable results, it appears more appropriate to perform two kinds of in-pile experiments, namely separate effects tests and integral tests.

3.3.1 SEPARATE EFFECTS TESTS

The objectives of these tests are to address phenomenological aspects, in order to confirm or correct and extent the previous results relative to:

- rod deformation,
- fuel relocation,
- the resulting resistance to thermal shock loads, with or without effect of clad axial constraining.

These tests should be realised with one irradiated rod within a ring of 8 fresh fuel rods, which will provide a representative thermal environment in order to ensure representative strains and subsequent phenomena. In addition, these in-pile separate effects tests should include a blowdown phase. As mentioned before, this phase will provide representative conditions for the temperature transient inside the fuel to study the consequences in terms of thermomechanical pre-fragmentation during blowdown.

3.3.2 3.3.2 INTEGRAL TESTS.

This kind of tests will address the aspects of:

- impact of blowdown phase
- flow blockage
- quenching behaviour and coolability.

These tests should allow to check the absence of unexpected phenomena or unexpected coupling between foreseen processes, and finally provide data for the validation of reactor computational tools.

These tests should be realised with 9 high burn up rods with a ring of 12 or 16 fresh fuel rods which will provide a representative thermal environment in order to ensure representative strains and subsequent phenomena. A blowdown phase will be simulated depending on its importance as observed in the previous studies. Finally, additional axial stress during quenching due to rod blockage in the assembly should be simulated during these tests.

3.3.3 EXPERIMENTAL FACILITY

Such a programme is envisaged by the IPSN in the PHEBUS facility where some twenty LOCA tests were run between 76 and 83 [11,14], see figure 6. By this way IPSN would take advantage of the know-how accumulated when the previous LOCA programme with fresh fuel was run.

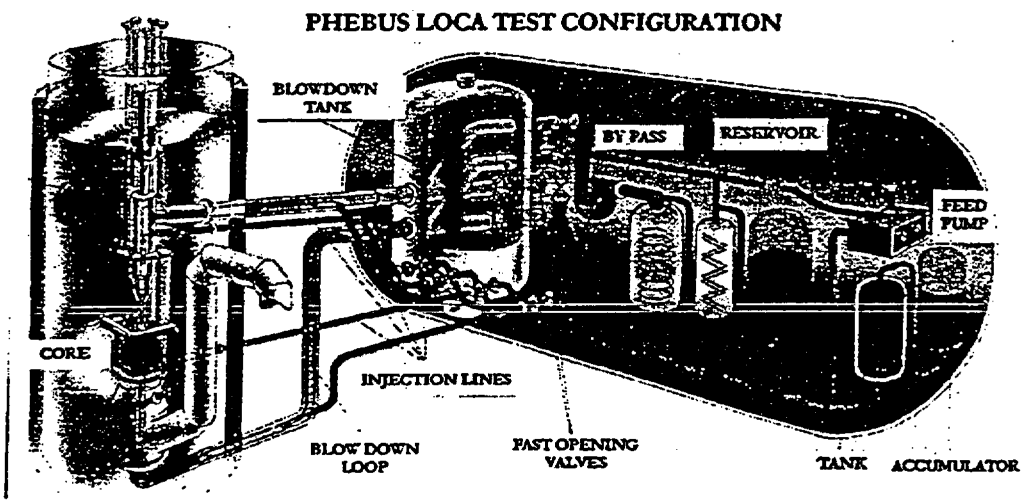

Figure 6

Furtnermore, a new LOCA programme in the Phébus facility would take advantage of the R&D efforts made for the subsequent programmes in terms of high activity material measurements

Tomography technique [15,16] is one of the examples, which can be given how such efforts provide practical applications for the new LOCA programme. This technique provides the 3D location and the nature of the material fragments everywhere in a bundle. The exact geometry of the bundle at the end of the test can be reconstructed and explored from the inside. Fuel relocation studies and code validation will be made easier through this technique.

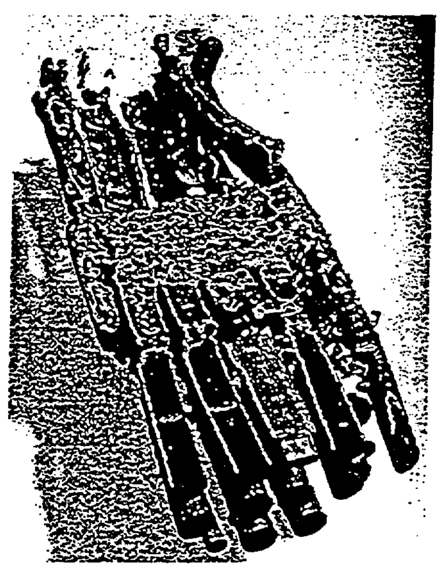

Figure 7

Presently, fragment size less than 500 microns can be located (see figure 7). Further improvement of the existing technique will increase the resolution providing several points inside the clad with an oxide/metal discrimination.

4. CONCLUSIONS

Studies performed in IPSN and elsewhere pointed out that high burnup may induce specific effects under LOCA conditions, especially those related with fuel relocation. Uncertainties exist regarding how much these effects might affect the late evolution of the accident transient and the associated safety issues. IPSN estimates that a better knowledge of specific phenomena is required in order to resolve the pending uncertainties related to LOCA criteria. IPSN is preparing the so-called APRP-Irradié (High Burnup fuel LOCA) programme. One of the important aspects of this programme is in-Pile experiments involving bundle geometries in the PHEBUS facility located at Cadarache, France. A feasibility study for such an experimental programme is underway and should provide soon a finalised project including cost and schedule aspects.