EXPERIMENTAL DATA ON PCI AND PCMI WITHIN THE IFPE DATABASE

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Abstract

Following the conclusions reached at the end of the FUMEX-I code comparison exercise, the International Fuel Performance Experimental Database (IFPE) gave priority to collecting and assembling data sets addressing: thermal performance, fission gas release and pellet-clad mechanical interaction (PCMI). The data available that address the last topic are the subject of the current paper.

The data on mechanical interaction in fuel rods fall into three broad categories:

1. fuel rod diameter changes caused by periods spent at higher than normal power;

2. the result of power ramp testing to define a failure threshold;

3. single effects studies to measure changes in gaseous porosity causing fuel swelling during controlled test conditions.

In the first category, the fuel remained un-failed at the end of the test and the resulting permanent clad strain was due to PCMI caused by thermal expansion of the pellet and gaseous fuel swelling. Some excellent data in this category come from the last two Risø Fission Gas Release projects. The second category, namely, failure by pellet-clad interaction (PCI) and stress corrosion cracking (SCC) involves the simultaneous imposition of stress and the availability of corrosive fission products. A comprehensive list of tests carried out in the Swedish Studsvik reactor is included in the database. The third category is a recent acquisition to the database and comprises data on fuel swelling obtained from ramp tests on AGR fuel and carried out in the Halden BWR. This data set contains a wealth of well-qualified data which are invaluable for the development and validation of fuel swelling models.

Introduction

The aim of the International Fuel Performance Experiments Database (IFPE) is to provide a comprehensive and well-qualified database on Zr clad UO_2 fuel for model development and code validation in the public domain. The data encompass both normal and off-normal operation and include prototypic commercial irradiations as well as experiments performed in material testing reactors.

The database is restricted to thermal reactor fuel performance, principally with standard product zircaloy-clad UO_2 fuel, although the addition of advanced products is included where available, e.g. fuel including MOX, gadolinia and niobia, etc., and clad variants. Emphasis has been placed on including well-qualified data that illustrate specific aspects of fuel performance. Following the conclusions reached at the end of the FUMEX-I code comparison exercise, priority was given to collecting and assembling data sets addressing: thermal performance, fission gas release and pellet-clad mechanical interaction (PCMI). Whilst most of the data sets concern irradiation experiments on fuel rods, data on out-of-pile tests investigating fission gas release and fuel swelling are also included. The list of data sets currently available is given in Table 1. No references are given in this paper as all appropriate documentation was scanned and accompanies the data sets on the CDs supplied by the NEA.

The mechanical interaction between fuel pellets and cladding

In the case of water-cooled reactors, the fuel rods are made up of ceramic fissile pellets of UO_2 or $(Pu,U)O_2$ contained in a zirconium-based alloy tube. At the beginning of the irradiation, there is a distinct circumferential gap between the external face of the fuel pellets and the inner bore of the cladding tube. On raising power, this gap is reduced through thermal expansion of the pellet. This gap becomes less distinct as the ceramic pellets crack under the influence of the radial temperature gradient. The distance between the pellet surface and the cladding inner bore is further reduced as the pellet fragments relocate radially outwards. The free volume is now shared between the residual gap and the space between pellet fragments.

As irradiation proceeds, the cladding diameter reduces by creep driven by the compressive hoop stress induced by the difference between coolant pressure and the internal pressure of the fuel rod. Eventually, the cladding creeps down onto the fuel finally eliminating the fuel-to-clad gap and moving the pellet fragments closer together. As a rough guide, this occurs during the second cycle of irradiation in the burn-up range 10-20 MWd/kg. Prior to gap closure, the cladding is under a compressive hoop stress. Once the gap is closed and the cladding and pellet fragments are in contact, the hoop stress gradually decreases as the pellet fragments resist the reduction in cladding diameter. The hoop stress in the cladding eventually becomes positive when all the internal free space is exhausted and the pellet fragments are in intimate contact.

At any stage during irradiation, an increase in power can cause the pellets to expand through both thermal expansion and fission product swelling to induce a positive hoop stress in the cladding. The immediate reaction of the cladding is to expand outward by elastic deformation and subsequently by plastic strain and creep, thus reducing the interfacial stress. This interaction between the fuel pellet and the cladding is termed pellet-clad mechanical interaction (PCMI). If, however, the clad hoop stress is sufficiently large, and the pellet temperature is high enough to release corrosive fission products, internal cracks may be initiated at the inner bore by stress corrosion cracking (SCC) which grows under the influence of the maintained hoop stress such that the cladding fails. The generic term for this is failure by pellet-clad interaction (PCI).

Operation of commercial reactors with failed cladding is not a viable option as it compromises safety and leads to a highly contaminated primary coolant circuit resulting in difficulties with maintenance and fuel handling on discharge. Reactors are sometimes operated intentionally with a small number of small cladding defects, but wherever possible such failures are avoided by careful choice of materials, manufacturing quality assurance measures and operating constraints. In order to define the limit of safe operation, it is necessary to develop fuel performance codes to calculate PCMI effects and validate the predictions against experimental data on cladding diameter change. To take the predictions further by calculating probability of failure, code predictions must be compared with the results of dedicated ramp tests providing statistics including both failed and un-failed rods.

Such data are available, and it has been the intention of the IFPE Database to include as much useful data as possible for use by code developers. This paper sets out to guide users to the most appropriate data for their requirements. To this end, discussion of the data is provided under the following three headings:

- fuel rod diameter changes induced by PCMI;
- the definition of the PCI failure threshold;
- single effects studies to measure fuel swelling during controlled test conditions.

The first two topics have already been discussed. The third has been alluded to when discussing expansion of the fuel pellet. In effect, along with thermal expansion, fuel swelling is one of the driving forces for PCMI. Whereas inexorable swelling caused by the incorporation of fission products in the lattice is a linear function of burn-up, the concern here is the formation and behaviour of gas bubbles when they are created during high-power operation. As will be seen later, the data set included in the database is a highly detailed and valuable study of this phenomenon.

Fuel rod diameter induced PCMI

The major part of this data set is made up of PIE diameter measurement before and after fuel rods have been subjected to a period of high power. There are three main data sets addressing this behaviour. These will be discussed first, followed by a small number of cases concerning the related effect of clad length changes driven by expansion of the fuel column. These are all in-pile measurements which can illustrate some of the dynamic effects of PCMI. Additional data are available from the series of Studsvik ramp tests and these are discussed further on.

The Risø Transient Fission Gas Release Project

In this project, short lengths of irradiated fuel were fitted with in-pile pressure transducers and ramped in the Risø DR3 reactor. The fuel used came from either IFA-161 irradiated in the Halden reactor or from GE segments irradiated in the Millstone BWR. Using this refabrication technique, it was possible to back-fill the test segment with a choice of gas and gas pressure and to measure the time dependence of fission gas release by continuous monitoring of the plenum pressure. The short length of the test segment was an advantage because, depending on where along the original rod the section was taken, burn-up could be a chosen variable, and during the test the fuel experienced a single power. Some segments were tested without refabrication. Here the fuel stack was longer than in the case of the refabricated tests and hence the segments experienced a range of powers during the ramp depending on axial position in the test reactor. These "unopened" segments were used to confirm that refabrication did not affect the outcome of the tests. Extensive hot cell examination compared the fuel dimensions and microstructure before and after the tests.

Some 17 tests were performed and all but one (which failed) have been included in the database and provide valuable clad diametral deformation (ΔD) and fuel swelling as a function of ramp power and hold time. A summary of the test matrix is given in Table 2 with a brief comment on the degree of PCMI observed. It was noticeable that the Halden irradiated fuel, i.e. the "Risø" tests, showed distinct ridging and ridge height growth, unlike the GE tests, where there was little evidence of this. This contrast in behaviour is illustrated in Figure 1 showing the diameter traces before and after tests Riso-a and GE-a. Figure 2 shows the variation in diameter change observed for GE-i which was tested without refabrication. It is clear that the diameter change is a function of position and hence power during the test.

Figure 1(a). Traces of rod diameter before and after testing for Riso-a, showing distinct ridges and ridge growth

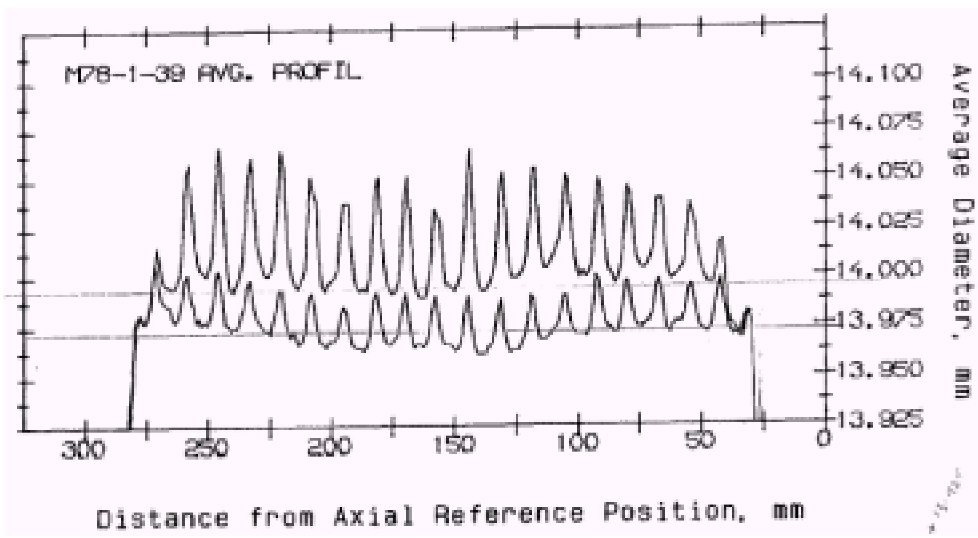


Figure 1(b). Traces of rod diameter before and after testing for GE-a showing indistinct evidence of ridging

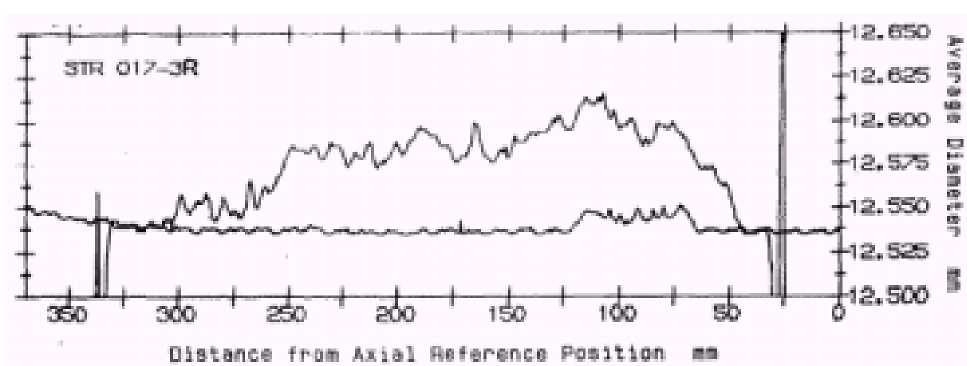
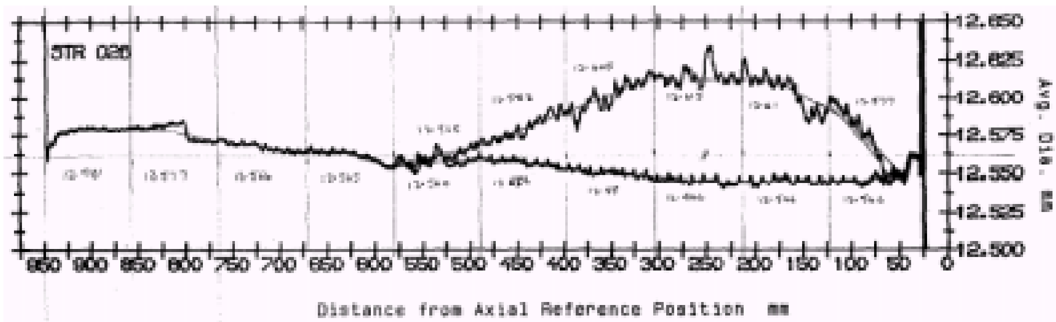


Figure 2. Diameter traces before and after test GE-i showing the dependence of diameter change on position and hence power during the test



The Third Risø Fission Gas Release Project

The third and final Risø project was very similar to the previous one with the exception that many rods were refabricated with pressure transducers and thermocouples, hence the data generated was complemented by knowledge of fuel centreline temperatures during the high-power test. The fuel used in the project was from: IFA-161 irradiated from 13-46 MWd/kgUO_2 in the Halden BWR, GE BWR fuel irradiated from 20-40 MWd/kgUO_2 in Quad Cities 1 and Millstone 1, and ANF PWR fuel irradiated in Biblis A to 38 MWd/kgUO_2. There were seven tests using PWR fuel, three of which were long sections without refabrication. Four tests used GE BWR fuel, one of which was a long section without refabrication and four tests used sections cut from IFA-161 rods. The tests consisted of a period held at a constant high power but, as in the previous project, some tests had short duration over and under power peaks and dips, respectively. These were conducted expressly to investigate axial gas transport to the plenum where the in-pile pressure transducer was situated. An overview of the test matrix is shown in Table 3 whilst an example of diameter traces before and after test AN3 is shown in Figure 3. Note in particular the growth of ridges at the pellet ends and the reduced deformation over the thermocouple. This is on account of the reduced power and lower PCMI induced by the hollow pellets. The large diameter change and its dependence on axial position for test GE7 is shown in Figure 4.

Figure 3. Axial diameter traces before and after test AN3. Note the growth of ridges and reduced deformation over the thermocouple.

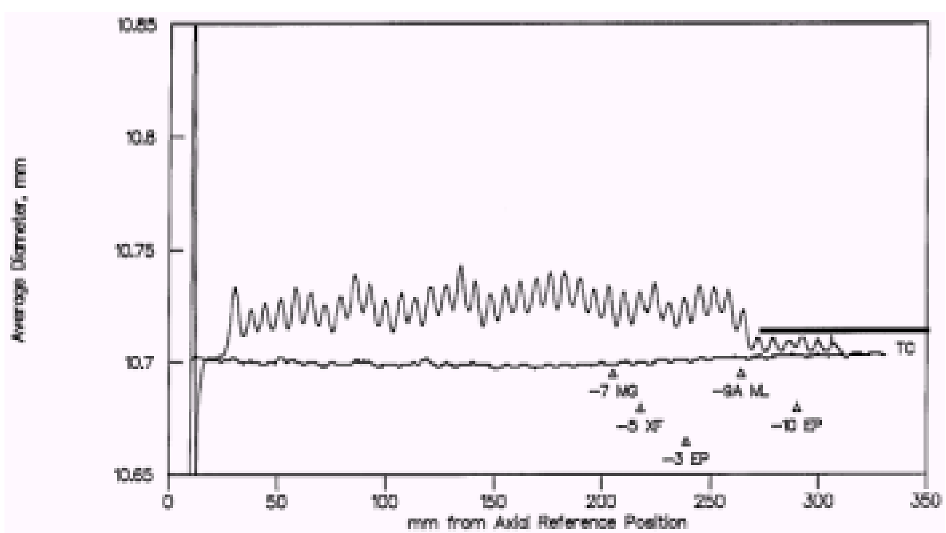
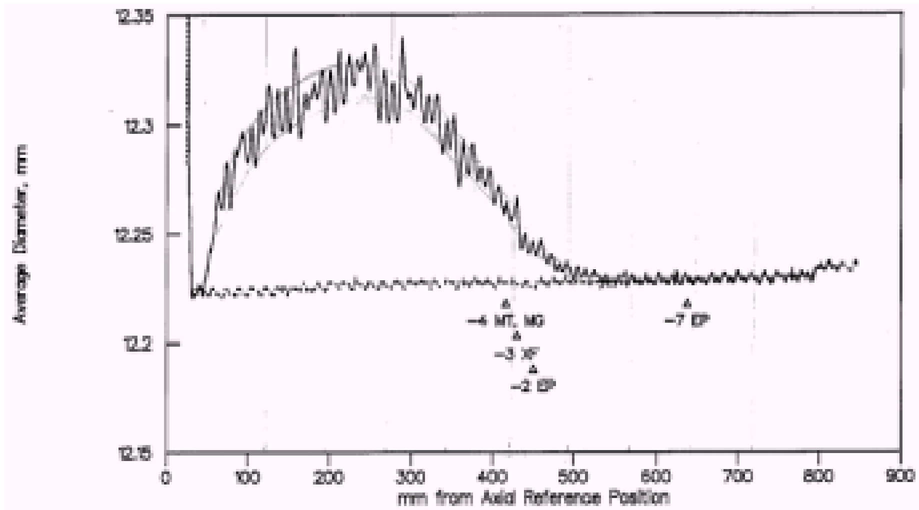


**Figure 4. The axial variation in diameter before and after the test GE7.
The variation in ΔD is on account of the axial power profile over this long rod.**



Both figures show the positions at which sections were cut for PIE by transverse metallography (MT), longitudinal metallography (ML), transverse micro gamma scanning (MG), transverse X-ray fluorescence (XRF) and transverse electron probe micro analysis (EPMA). Thus the diameter changes are accompanied by detailed measurements of grain size, gas bubble and fission product distributions.

CEA/EDF/FRAMATOME PWR rods ramped in OSIRIS

This data set contains details of three standard PWR rods and one segmented rod irradiated in EDF commercial reactors. One rod and one segmented rod were refabricated and ramp tested in the CEA OSIRIS reactor to investigate their PCI resistance. The data set contains details of the pre-characterisation of the fuel pellets, the cladding tube and the assembled fuel rod.

Segment J12-5 was from the fifth span of the segmented rod J12 irradiated for two cycles in Gravlines 5 to 23.8 MWd/kgU. This was refabricated with new end plugs without disturbing either the fuel column or the internal fill gas and ramp tested in OSIRIS. After conditioning at 21 kW/m it was quickly ramped and held at 39.5 kW/m and discharged without failure.

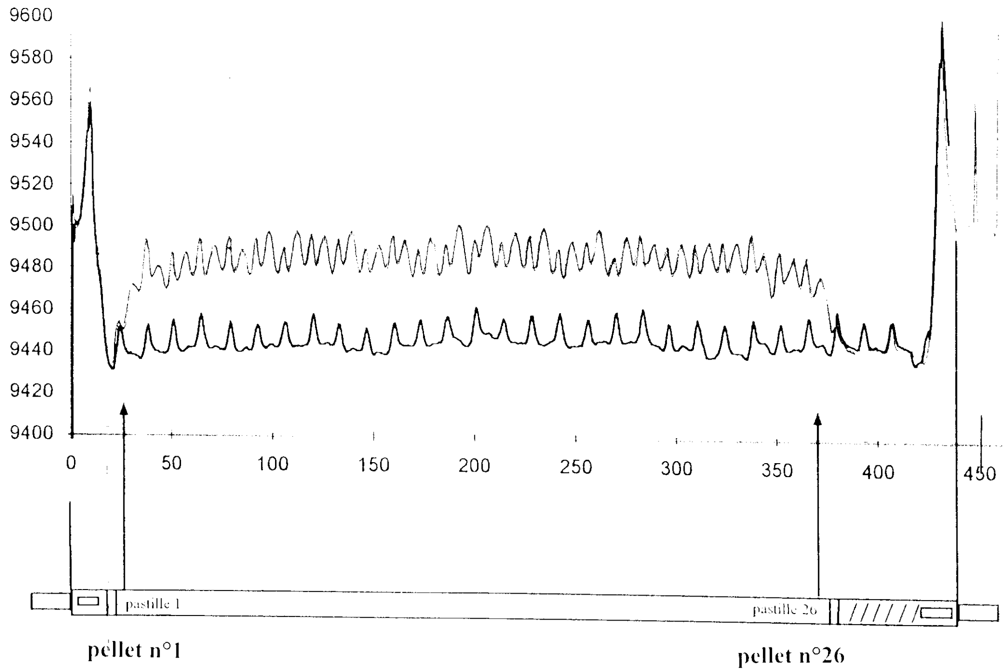
K11 was a full-length rod irradiated to 25.4 MWd/kgU for two cycles in Gravelines 3. A section of this rod was cut from span 5 between grids and ramp tested as K11-5 in OSIRIS. After conditioning at 24 kW/m it was quickly ramped and held at 43.7 kW/m and discharged without failure.

The ramp induced measurable diameter change and ridge height growth in both tests. Figure 5 shows the diameters before and after the ramp for K11-5. Of particular interest in this figure is the growth of secondary ridges located at the mid-length of the pellets. This was also observed in J12-5.

Halden in-pile cladding length changes

Many irradiations carried out in the Halden reactor contain rods fitted with clad elongation detectors. These enable a measure of clad length to be made in-pile providing data as a function of power, time and burn-up. As such, the data are extremely useful for evaluating the interaction between

Figure 5. Diameter traces before and after ramp testing section K11-5 in the CEA OSIRIS reactor

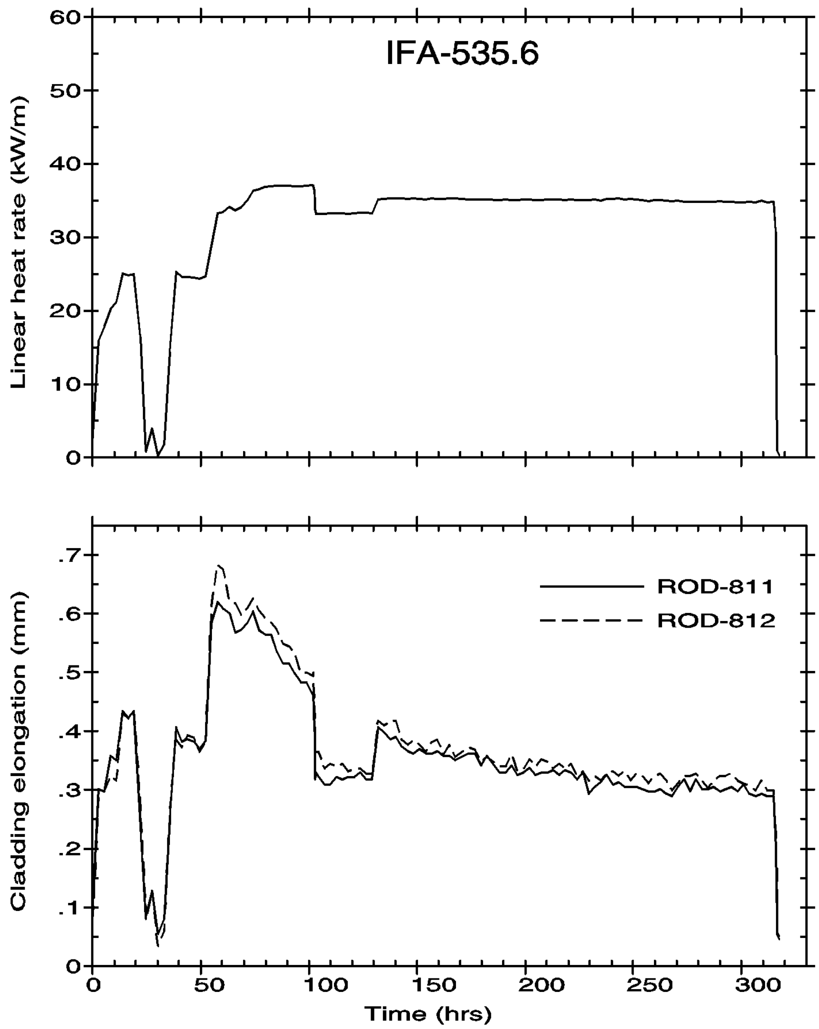


pellet and cladding under various irradiation conditions. The database contains such data for experiments IFA-432 rods 2, 3 and 6, IFA-535.5 and .6 and IFA-597.3 rod 7. IFA-432 contained rods of different grain sizes and was irradiated in the HBWR to provide information on the thermo-mechanical behaviour of fuel rods for code development. PCMI was observed at the start of life for the small gap rod 3, but it was not observed to any degree in the large gap rods 2 and 6 until there was appreciable fission gas release and swelling. IFA-535.5 and .6 were rods irradiated to 44 MWd/kgUO_2 at modest powers in IFA-409 before re-instrumentation and ramping in IFA-535; two rods were fast ramped in loading five and two other rods were subjected to a slow ramp in the sixth loading. The third loading of IFA-597 contained two rods of which rod 7 was fitted with a clad elongation detector. The fuel was cut from rods previously irradiated at low power to a high burn-up of 59 MWd/kgUO_2 in the Ringhals 1 BWR. On re-irradiation in Halden, the rod power was increased to 30 kW/m and decreased slowly to ~22 kW/m at a burn-up of 61 MWd/kgUO_2. In all cases, an increase in power is accompanied by an increase in cladding length which subsequently relaxes. A typical behaviour is illustrated in Figure 6 for the slow-ramped rods in IFA-535.6.

The definition of the PCI failure thresholds

The Studsvik laboratories in Sweden have specialised in performing ramp tests to establish the propensity to failure of different design fuel rods. They have carried out many sponsored programmes of which the database contains the results from the BWR projects INTER-RAMP, SUPER-RAMP, DEMO-RAMP 1 and 2 and TRANS-RAMP 1. Concerning PWR fuel, the database contains results from OVER-RAMP, SUPER-RAMP, TRANS-RAMP 2 and 4; in all, some 54 BWR rods and 80 PWR rods. This section concentrates on the data available to define the failure threshold but, in addition, the PIE carried out in the various programmes also produced good data on diameter change and ridge height growth.

Figure 6. In-pile clad elongation as observed during a slow ramp and hold in the Halden experiment IFA-535.6. Note the relaxation during the period of constant high power.



The Studsvik INTER-RAMP BWR Programme

The objectives of this project were to establish the fail-safe operating limits of 20 standard-type, unpressurised BWR fuel rods on over-power ramping at the burn-up levels of 10 and 20 MWd/kgU. The over-power ramping was to be performed at a fast ramp rate of about 4 kW/m/min with the preceding base irradiation performed to represent the conditions in a typical commercial BWR power reactor. The fuel was manufactured by ASEA-ATOM and irradiated in boiling capsules (BOCA) in the Studsvik R2 reactor. The disposition of the rods in the boiling capsules resulted in "high" and "low" power groups of rods. The study investigated the influence of three main design parameters: clad heat treatment (re-crystallised anneal "RX" vs cold work plus stress relief anneal "SR"), pellet/clad diametral gap size and fuel density. Ramping was to power levels ranging between 41 to 65.4 kW/m resulting in 11 failures. A failure threshold around 42-43 kW/m was found for the low power rods and a threshold of around 47-48 kW/m was found for high power rods. PIE was performed to measure diameter changes and ridge height growth.

The Studsvik SUPER-RAMP BWR Sub-programme

The BWR sub-programme consisted of three groups of rods with variations in design and material parameters. The rods were base-irradiated in BWRs Würgassen or Monticello at average heat ratings in the range 11-23 kW/m to peak burn-ups in the range 28-38 MWd/kgU and were subsequently ramp-tested in the R2 reactor at Studsvik. The major results can be summarised as follows:

- *Standard-type KWU fuel rods, group BK7*:

  - When ramped as a single fast ramp from a conditioning power of 25 kW/m, a failure threshold for PCI/SCC was found at 32.5-36 kW/m and power change of 7.5-11 kW/m. Performing the ramps in two steps with conditioning after the first step for 12-24 hours at 32.5 kW/m increased the threshold to at least 37.5 kW/m with a permissible power change of >12.5 kW/m.

  - Single ramps from 18 kW/m provided a failure threshold of 30-33 kW/m with a power change of 12-15 kW/m.

  - Ramp testing up to 40.5 kW/m produced only small dimensional changes, little FGR and slight fuel structure changes. Thin layers containing uranium and fission products were found on the inside surface of the clad only at pellet-to-pellet interfaces or at the location of pellet cracks.

- *Standard GE fuel rods groups BG8 and BG9*. These rods were tested with various ramp rates and final power levels. It was possible to define the conditions below for which no failures were experienced:

  - Single-step ramping from 21.5 kW/m to 41.5 kW/m for the lowest ramp rates tested, 0.033 W/cm/min.

  - Single-step ramping from 21.5 kW/m to 32 kW/m at a safe ramp rate of 0.044 W/cm/min, and from 32 kW/m to 38 kW/m at 0.033 W/cm/min.

  - Up to 44 kW/m there were considerable dimensional changes, little FGR and moderate fuel structure changes.

The Studsvik DEMO-RAMP 1 BWR Programme

Five rods were manufactured by BNFL and irradiated in Ringhals 1 at powers 16-29 kW/m to burn-ups of 14-17 MWd/kgU. The rods encompassed three PCI remedy candidates:

- annular short length pellets;

- Nb_2O_5-doped UO_2 of large grain size;

- high helium pressure.

The project demonstrated an improved performance for all potential remedies as no rods failed during R2 ramping to powers in the range 46-61 kW/m.

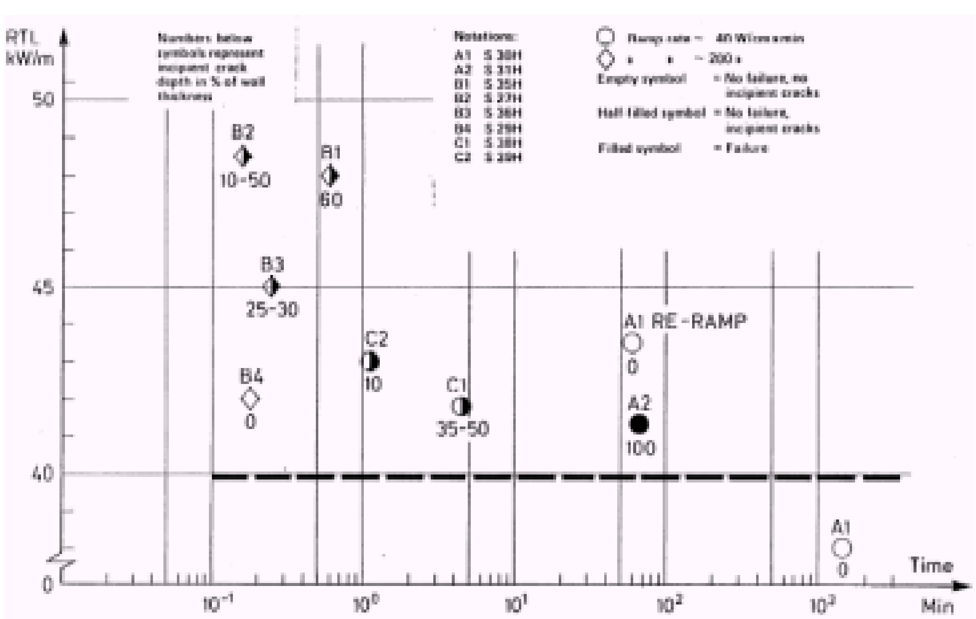
The Studsvik DEMO-RAMP 2 BWR Programme

The principal objective of the DEMO-RAMP II project was to investigate the early stages of clad failure by PCI during fast power up-rating of standard type BWR 8 × 8 test fuel rods. Eight KWU manufactured rods were base-irradiated through three consecutive fuel cycles at heat ratings in the range of 16-30 kW/m to burn-ups in the range of 25-29 MWd/kgU in the commercial Würgassen boiling water reactor in FR Germany and ramped in the R2 reactor.

The project produced results that can be summarised as follows:

- The eight rods ramp-tested produced one failed rod, five rods containing incipient cracks and two un-failed rods.
- A fuel failure threshold was established. The location of the failure threshold was in good agreement with the result of the INTER-RAMP project study.
- Due to the success in catching incipient cracks by the interrupted ramp tests, data were produced on the incidence of incipient crack formation as a function of ramp terminal levels and hold times in short time power transients.
- The failure pre-stage was characterised by the identification of incipient stress corrosion cracks at the cladding inside surface and quantified in terms of, e.g. incipient crack depth, restructuring of the fuel pellets and fission gas release, these data being dependent on both power level and time at power.
- It was found that the time for initiation of stress corrosion cracks in the cladding was very short, and that the initiation of cracks was accompanied with very little fission gas release. The best way of illustrating the results is by means of the plot of ramp terminal level versus hold time as shown in Figure 7.

Figure 7. Disposition of defected cladding as a function of ramp terminal level and hold time for rods ramped in the DEMO-RAMP 2 project



The Studsvik TRANS-RAMP 1 BWR Programme

Five KWU BWR design test fuel rods were ramp-tested under a very fast power increase after base irradiation in the Würgassen power reactor at heat ratings in the range of 20-30 kW/m to burn-ups in the range of 18-21 MWd/kgU.

After conditioning at 30 kW/m for 24 hours, the rods were ramped at a rate of approximately 1 000 kW/m/min to power levels in the range of 47.5-56 kW/m. Two of the rods were held at the ramp terminal level long enough to give an indication of failure by the rod elongation sensor, after 58 and 74 seconds, respectively, and later a release of fission product activity to the coolant.

Three of the ramp tests were purposely terminated after a very short time (18.5, 35.5 and 55.5 seconds). Incipient cracks with cladding wall penetration up to 20-50% of the wall thickness were found at pellet interfacial positions in the rods tested as interrupted ramp tests. No incipient cracks were observed at axial regions corresponding to local powers below 40 kW/m. It was concluded that incipient cracks are formed in the cladding if the failure threshold power level is exceeded. The development of the cracks is dependent on time and power.

The Studsvik OVER-RAMP PWR Programme

The programme power-ramped 39 individual test fuel rods of two different origins and designs. Twenty-four (24) of the rods were of KWU/CE design and were provided by KWU. They were delivered to the project following base irradiation in Obrigheim to burn-ups in the range 12-31 MWd/kg. Fifteen (15) of the rods were of Westinghouse design and were delivered after irradiation in the BR-3 reactor at Mol, to burn-ups in the range 16-24 MWd/kgU.

The KWU rods were ramped to final power levels in the range 37.8-53.0 kW/m producing failures in seven rods. The Westinghouse rods were ramped to final power levels in the range 37.5-44.5 kW/m producing failures in seven rods. The KWU rods were divided into six sub-sets whilst the Westinghouse rods were in four sub-sets. Taken individually, failures and no failures were obtained in all sub-sets apart from one KWU sub-set where no failures occurred. For all but this sub-set, failure thresholds could be devised, but the small number of rods and hence poor statistics meant that any derived value had a large uncertainty.

The SUPER-RAMP PWR Sub-programme

This consisted of six groups of rods with variations in design and material parameters. The KWU/CE rods were base-irradiated in the Obrigheim power reactor whilst the Westinghouse supplied rods were irradiated in BR-3. KW/CE rods were identified in four sub-groups: PK1, PK2, PK4 and PK6, with burn-up levels 11-12, 21-23, 21-22 and 22-25 MWd/kg, respectively. The Westinghouse rods formed two sub-groups (PW3 and PW5) having achieved 35-38 and 39-41 MWd/kgU, respectively.

The result of the ramp testing can be summarised as follows:

- *Standard-type fuel rods, group PK1 and PK2*. All rods sustained ramping to power levels in the range 41-49 kW/m and power changes 16-24 kW/m without failure, in spite of large deformations, fuel restructuring and fission gas release particularly for the PK2 rods.

- *Standard rods containing gadolinia, group PK4*. The rods all sustained power ramping to levels in the range 39-50.5 kW/m and power changes 14-25 kW/m without failure, in spite of large deformations, fuel restructuring and fission gas release.

- *Test fuel rods containing large grain size fuel, group PK6*. A failure threshold of 44 kW/m and power change of 18.5 kW/m was established for these rods. The fuel restructuring was modest and the fission gas release was low compared to other PK rods. Significant fuel bonding between fuel and cladding was found.

- *Standard rods, group PW3*. A failure threshold of ~37.5 kW/m was found and a power change of 12.5 kW/m.

- *Test fuel rods containing annular pellets, group PW5*. The rods all failed at power levels 38-43 kW/m with a power change of 13-18 kW/m. Hence there was no improvement in failure resistance over standard solid pellet rods.

- *General observations*. The inside clad of all ramped rods in groups PK1, PK2 and PK4 was mostly covered with a thin layer of deposits containing uranium and fission products. On PK6 rods, significant bonding between fuel and clad was found in large patches.

The Studsvik TRANS-RAMP 2 PWR Programme

The project's test programme consisted of ramping six fuel rods manufactured by Westinghouse and irradiated in the Zorita (Jose Cabrera) nuclear power plant in Spain. These rods had been base-irradiated at heat ratings in the range of 20-22 kW/m.

After conditioning at 20 kW/m for six hours the rods were ramped at a rate of about 100 W/cm/min to power levels in the range of 43-60 kW/m. Of the six fuel rods, three failed after 48-80 seconds (from the start of the ramp) while the remaining three were un-failed after 26-60 seconds. These results showed the relationship between rod power at failure and the time.

The rods underwent a thorough examination programme, comprising characterisation prior to the base irradiation, non-destructive examination between the base irradiation and the ramp irradiations, on-line measurements during the ramp irradiation and both non-destructive and destructive examinations after the ramp irradiations. Clad inside inspections give a correlation between the formation of incipient (non-penetrating) cladding cracks and the rod linear power indicating a damage threshold of about 40 kW/m, possibly lower.

The Studsvik TRANS-RAMP 4 PWR Programme

Seven test fuel rods, re-fabricated by the CEA-FABRICE process from full-size PWR fuel rods of standard FRAGEMA design irradiated to a burn-up of about 28 MWd/kgU in the French reactor plant Gravelines 3, were made available to the project.

Four of the test fuel rods were used for exploratory ramp tests to obtain information on the failure boundary curve and on the ramp test data needed to produce incipient cracks in the cladding of the fuel rods.

The remaining three test fuel rods underwent first a power transient in the R2 loop No. 1, then an irradiation at PWR conditions in a BOCA rig in the R2 to give a burn-up increment of about 4 MWd/kgU, and finally, a second power ramp, to about the same power level as during the first power transient, but with hold to failure.

Based on the ramp results and the results of the non-destructive examinations of the three rods performed prior to, between and after the three irradiation phases the following conclusions were drawn:

- The first transient caused the formation of incipient cladding cracks in only one of the rods. This also rod had the largest amount of pellet-to-pellet dish filling.

- The BOCA irradiation caused a propagation of the cracks in the rod containing incipient cracks.

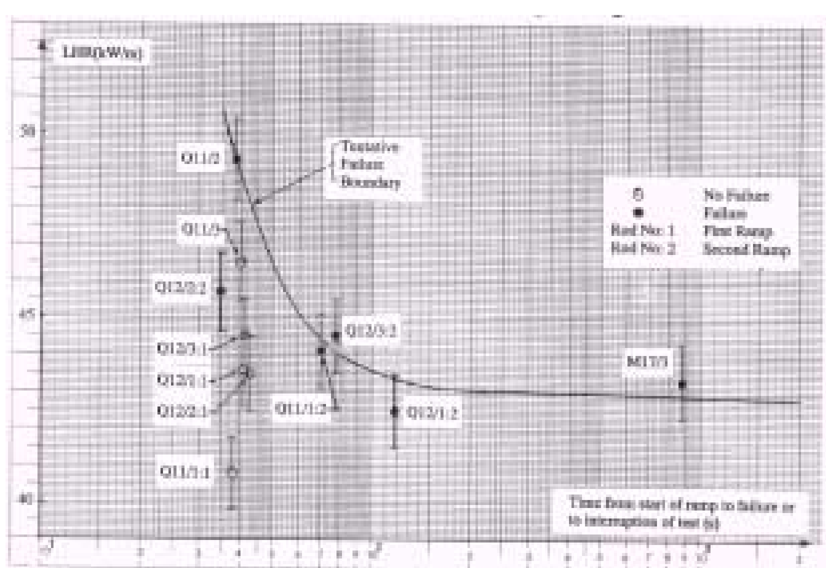
- The second power ramp caused a further propagation of the cracks in the rod containing incipient cracks and caused the rod to fail with a time to failure shorter than would be expected for a rod going through a first transient. This implied that this rod was exposed to cumulative damage which resulted in enhanced failure during the second power ramp.

- For the two other rods it was not possible to draw any firm conclusion about the influence of the first transient on the rod behaviour during the second ramp test due to the lack of an established failure boundary curve for rods ramped only once. Comparing results of previously performed ramp test projects, it seemed probable that the first transient did not influence rod behaviour during the second ramp test.

The test matrix is shown in the table below, whilst the results are synthesised to produce a failure boundary in Figure 8.

Rod	Ramp 1	BOCA irradiation	Ramp 2
M12/3	Yes – failed	–	–
Q11/1	Yes – no failure	–	Yes – failed
Q11/2	Yes – failed	–	–
Q11/3	Yes – no failure	–	–
Q12/1	Yes – no failure	Yes	Yes – failed
Q12/2	Yes – no failure	Yes	Yes – failed
Q12/3	Yes – no failure	Yes	Yes – failed

Figure 8. Failure threshold derived from the results of the TRANS-RAMP 4 PWR programme



Single effects studies on fuel swelling

This data set comprises measurements of inter- and intra-granular porosity and associated swelling from an extensive study of UO_2 fuel power ramped in the Halden reactor. The ramp tests were performed to study the mechanisms of PCI in advanced gas-cooled reactor (AGR) fuel, with the initial clad deformation measurements supplemented by the use of transmission electron microscopy (TEM) and scanning electron microscopy (SEM). Although the cladding in this case was stainless steel, the data on fuel swelling are generic and equally applicable to LWR fuel modelling.

Fuel specimens from 11 ramped rods and two control/reference rods were examined using TEM and SEM. For each specimen, swelling measurements were made at four or five radial locations in the fuel. At least six full grain boundaries were used for the inter-granular study at each location and three complete trans-granular fractures employed for the intra-granular bubbles. In the latter case, the trans-granular regions were examined under very high magnifications to reveal pores as small as 20-25 nm diameter. The SEM study comprises nearly three thousand micrographs.

The microscope study was augmented by use of the ENIGMA fuel modelling code to obtain estimates of the local temperatures and conditions from which the SEM/TEM samples were obtained.

In addition, un-ramped samples of the same fuel were annealed to temperatures of 1 600-1 900°C in a combination of different temperature ramp rates, maximum temperature attained and hold times. Measurements of fission gas release were made during the anneal with porosity distribution and swelling measurements made after the anneal.

The data contained in this data set are extensive and provide invaluable information to fuel modellers on the mechanisms involved in gas porosity formation, both intra- and inter-granular. The use of different ramp rates and the “park” period after ramping allows important conclusions to be drawn regarding bubble nucleation, the effects of vacancy starvation and the influence of irradiation induced re-solution. As such, the measurements provide the necessary data on which to develop and validate a model describing the dynamics of fuel swelling during the course of an over-power transient. For further details of this data set, reference should be made to the appropriate paper presented at this workshop.

Concluding remarks

Access to the database is through the NEA, who provides the data and documentation on CDs. While every care has been taken in preparing complete error-free data sets, there is sometimes a need to issue revisions. It is important therefore that all users be registered with the NEA so any revisions are sent to them as a matter of course. There is no charge made for this service, but recipients are urged to carefully review data sets that are used and to provide feedback on their experience in using the data. Improvements to the database rely on this interaction offered by users.

The creation of the database has met with universal approval and consequently there has been no difficulty experienced in obtaining data for inclusion. However, new data are always welcome. To gain access to the IFPE Database or to offer new data, contact should be made through:

Dr. Enrico SARTORI — Tel: +33 1 45 24 10 72
OECD/NEA Data Bank — Fax: +33 1 45 24 11 10
Le Seine-Saint Germain — Eml: sartori@nea.fr
12 boulevard des Iles
F-92130 ISSY-LES-MOULINEAUX
FRANCE

More information can be found at the following sites:

http://www.nea.fr/html/science/projects.html#fuel

http://www.nea.fr/html/dbprog/

Table 1. IFPE Database, list of cases as of December 2003

Halden irradiated IFA-432	5 rods
Halden irradiated IFA-429	7 rods
Halden irradiated IFA-562.1	12 rods
Halden irradiated IFA-533.2	1 rod
Halden irradiated IFA-535.5 &.6	4 rods
The Third Risø Fission Gas Release Project	16 rods
The Risø Transient Fission Gas Release Project	15 rods
The SOFIT WWER Fuel Irradiation Programme	12 rods
The High Burn-up Effects Programme	81 rods
WWER rods from Kola-3	32 rods
Rods from the TRIBULATION programme	19 rods
Studsvik INTER-RAMP BWR Project	20 rods
Studsvik OVER-RAMP PWR Project	39 rods
Studsvik SUPER-RAMP PWR Sub-programme	28 rods
Studsvik SUPER-RAMP BWR Sub-programme	16 rods
Studsvik DEMO-RAMP I – BWR	5 rods
Studsvik DEMO-RAMP II – BWR	8 rods
Studsvik TRANS-RAMP 1 – BWR	5 rods
Studsvik TRANS-RAMP 11 – PWR	6 rods
Studsvik TRANS-RAMP IV – PWR	7 rods
CEA/EDF/FRAMATOME Contact 1 & 2	3 rods
AEAT-IMC NFB 8 and 34	22 samples
CEA/EDF/FRAMATOME PWR and OSIRIS ramped fuel rods	4 rods
CENG defect fuel experiments	8 rods
CANDU elements irradiated in NRU	36 rods
Siemens PWR rods irradiated in GINNA	17 rodlets
CEA failed PWR rods irradiated in SILOE: EDITH-MOX 01	1 rod
CNEA six power ramp irradiations with (PHWR) MOX fuels	5 rods
BN GAIN $(U,Gd)O_2$ fuel	4 rods
INR Pitesti – RO-89 and RO-51 CANDU fuel type	2 rods
HRP IFA-597.3 rods 7, 8 and 9 (cladding degradation, FCT, FGR at Bu » 60 $MWd/kgUO_2$	3 rods
HRP IFA-534.14 rods 18 and 19 (EOL FGR and pressure, grains size of 22 and 8.5 micrometers and Bu » 52 $MWd/kgUO_2$	2 rods
DOE sponsored BR3 High Burn-up Fuel Rod Hot Cell Programme	5 rods
IAEA/OECD/IFE FUMEX – 1	6 rods
IMC (UK) Swelling data from CAGR UO_2 fuel ramped in the Halden HBWR	13 rods
NRU MT4 & MT6A LOCA simulation tests	33 rods
Total	**502 cases**

Table 2. Data from the Risø Transient Fission Gas Release Project

Test	Burn-up $MWd/kgUO_2$	Maximum power kW/m	Fill gas bar	FGR %	Comment re: test	Comment re: PCMI
Riso-a	37.27	39.8	1 Xe	(14)	Short section	Clad failure Noticeable ΔD with large ridge height growth
Riso-b	34.87	40.0	1 Xe	15.3	Short section	Small ΔD but large ridge height growth
Riso-e	36.95	41.6	5 He	24.9	Short section Long hold time	Noticeable ΔD with large ridge height growth
Riso-h	26.81	40.4	1 Xe	16.7	Short section	Small ΔD but large ridge height growth
Riso-i	42.31	40.1	1 Xe	13.3	Short section	LargeΔD and large ridge height growth
Riso-k	26.79	40.7	1 Xe	24.6	Short section Large gap	Little ΔD, some ridge growth
Riso-l	26.63	39.8	5 He	15.7	Short section Large gap	Little ΔD, some ridge growth
GE-a	28.54	42.4	5 He	18.8	Short section	Large ΔD indistinct ridge growth
GE-b	29.38	38.7	1 Xe	19.7	Short section	Variable small ΔD
GE-g	26.02	42.3	16 He	18.2	Short section	Large ΔD, indistinct ridge growth
GE-h	26.02	42.7	17 He	5.1	Long section Long hold	Good variation of ΔD as a function of power
GE-i	26.02	43.6	17 He	7.3	Long section Long hold	Good variation of ΔD as a function of power
GE-l	26.02	42.0	5 He	19.3	Power dips	Significant ΔD
GE-k	19.31	42.5	5 He	30.0	Power peaks	Small ΔD
GE-m	14.27	41.7	5 He	11.1	Power dips	No measurable ΔD
GE-n	Not included because of failure during test					
GE-o	26.86	44.5	5 He	33.8	Short section	Large constant ΔD

Table 3. Data from the Third Risø Fission Gas Release Project

Test	Burn-up MWd/kgUO$_2$	Maximum power kW/m	Fill gas bar	Instrumentation	Purpose of test	Comment re: PCMI
AN1	36.3	39.8	15 He	PF	Effect of refabrication	Significant ΔD and ridge growth
AN2	35.5	39.0	25 He	U	Effect of power history and refabrication	Large ΔD, good variation of ΔD as a function of power
AN3	36.3	40.7	10 He	PF, TF	Effect of fill gas, see AN4	Significant ΔD and ridge growth; noticeable reduction in ΔD over T/C
AN4	36.3	40.7	1 Xe	PF, TF	Effect of fill gas, see AN3	Significant ΔD and ridge growth; noticeable reduction in ΔD over T/C
AN8	35.5	29.8	25 He	U	Effect of power	Very large ΔD, good variation of ΔD as a function of power
AN10	36.3	34.4	5 He	PF, TF	Effect of power	Small ΔD and ridge growth; noticeable reduction in ΔD over T/C
AN11	36.3	16.9	25 He	U	Low power FGR	No measurements made
GE2	36.9	40.5	5 He	PF, TF	Effect of burn-up and power history	
GE4	20.5	43.3	5 He	PF, TF	Effect of low burn-up and grain size	No change in diameter nor ridging
GE6	37.2	37.9	5 He	PF, TF	Power history	Small ΔD and ridge growth
GE7	35.6	35.5	3 He	U	Power history and fuel type	Very large ΔD and ridge height growth, good variation of ΔD as a function of power
II1	38.4	40.3	1 Xe	TF	Link to 2nd project	Symmetric variation of ΔD and ridge height growth axially along test section
II2	22.4	42.8	5 He	PF, TF	Link to 2nd project	Modest ΔD along length with very large ΔD at location of failure
II3	12.8	44.7	5 He	PF, TF	Effect of burn-up	Negligible change in ΔD
II5	42.4	40.1	5 He	PF, TF		Symmetric variation of ΔD and ridge height growth axially along test section

Note: PF refers to in-pile pressure transducer, TF refers to the presence of a fuel centreline thermocouple (T/C). U is a test performed on a rod without refabrication.