Review of the established good practices in development of clad ballooning and embrittlement models for higher burnup PWR fuel

Final Report

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SUMMARY

This report addresses the issue of the possible ballooning of fuel cladding following a LOCA, when it overheats and the primary circuit has reduced greatly in pressure.

Such ballooning could cause the cladding of adjacent fuel rods to meet, and impede access by coolant, leading to fuel overheating and possible fission product release to the primary circuit (and to the secondary containment, given the postulated LOCA.)

This has been recognised for some decades as an issue of concern for water-cooled reactors. There has been through this time a series of major experimental programs, variously subjecting rods and bundles of rods to conditions intended to reproduce those likely to be experienced in the reactor under these circumstances. Associated with these experimental programs has been a major effort to model this fearsomely complex set of phenomena.

These experimental programs, and complementary modelling efforts, are reviewed and discussed.

The fundamental problem is that two distinct broad classes of phenomena occur simultaneously, and occur in an interacting, coupled fashion.

Fuel pins balloon, a process that incorporates all the hidden but considerable complexity of the thermal and mechanical behaviour of fuel rods.

The other phenomenon is the upward flow into the core of a chaotic mixture of liquid water and steam. The flow of this liquid and vapour is naturally influenced by the existence of balloons ahead of it; it is likely to be diverted if ballooning is significant, albeit entrained droplets and vapour will be diverted to differing degrees. Equally, diversion or otherwise of coolant will naturally affect the cooling of cladding, and this changed cooling will change cladding temperatures and consequently the creep and other deformation of the cladding. The disruption of the flow, and the capture and break up and re-entrainment of liquid droplets, by spacer grids, adds another major complexity. Modelling of all of this is plainly challenging.

The present state-of-the-art is identified, and suggestions are made for the way forward.

Starting perhaps 15 years ago the need for a coupled thermal hydraulic and fuel structural mechanics analysis was recognised, and one or two attempts along these lines have been made, and these probably represent the best current approaches to modelling of the ballooning phenomenon.

However, both of these are based on long-standing and historical approaches to both such classes modelling in nuclear engineering, and they do not use the more sophisticated methods that are now routine in other branches of engineering.

We identify an approach that follows the same spirit, of coupling thermal hydraulic and structural mechanics models, but where the models being so coupled much more represent the state-of-the-art in the treatment of their respective phenomena.

1 INTRODUCTION

The main requirements of this work were to:-

-Review the state of knowledge in the field of clad ballooning and the potential to demonstrate that, in the event of a postulated fault transient, the fuel assembly will retain a coolable geometry and the potential for disposal of the fuel using existing refuelling equipment.

-Consider the practicality of further development of analysis tools to support regulatory judgments on this topic.

There is obviously an enormous literature on the subject. Fortunately, there do exist several very large and fairly comprehensive review documents. A further, updating review, with some tens of participants and scheduled to take ~ 24 months or so, is just being organized. We will refer to these extensively, and confine more detailed discussion of individual experimental programs or modelling approaches to more of a selected sampling basis.

The structure of the report is as follows:-

In Section 2 we attempt to set in context clad ballooning during loss of coolant accident. In Section 3 we identify the complex interacting phenomena that need to be taken into account during ballooning, and discuss in broad terms approaches to its prediction, and in particular the respective roles of measurement and modelling. We describe the main literature in Section 4, identifying in particular the several (and very valuable) major reviews, and the 'updating' review in hand.

In Section 5 we introduce a characterised listing of the main experimental programmes, and in Section 6 give a little more detail on a small sample of these. In a similar way we discuss the main fuel modelling codes in Section 7. In Section 8 we consider whether there is any evidence for clad ballooning being an issue of concern anyway, and if it is, what should be done about it. Concluding that it is an issue, in Section 9 we suggest a way forward to address the problems that clad ballooning raises. Conclusions are drawn in Section 10.

2 CLAD BALLOONING DURING LOCA

2.1 The context

The concept of the "design basis accident" (DBA) is fundamental to the assessment of safety of nuclear electricity generation. The "loss of coolant accident", (LOCA), features large amongst these. Initially most attention was focused on large-break LOCA, but more recently greater attention has been paid, in addition, to smaller breaks, as these can present different challenges.

The basic requirement is that minimal radioactive material is released as a result of such an accident. In the context of a loss of coolant accident, this in practice becomes a requirement that significant fuel melting is avoided, albeit the breaching of the cladding of a number of fuel rods is generally acceptable as long as the release of radioactive material remains within acceptable regulatory limits.

Whilst in general termination of the fission chain reaction can be counted upon following LOCA, there remains the need to remove the decay heat of the fuel. Whilst this is initially perhaps only 7% of the previous operating power, and it does decline significantly, albeit over a period measured in hours rather than seconds (see Figure 1), this is still a significant power and power density.† Removal of this thermal power requires that sufficient coolant can be brought to the core, and that the core, during this time, retains a "coolable geometry". Ballooning of the cladding is a mechanism that can lead to significant geometrical changes in the core, and by which coolability could be impaired.

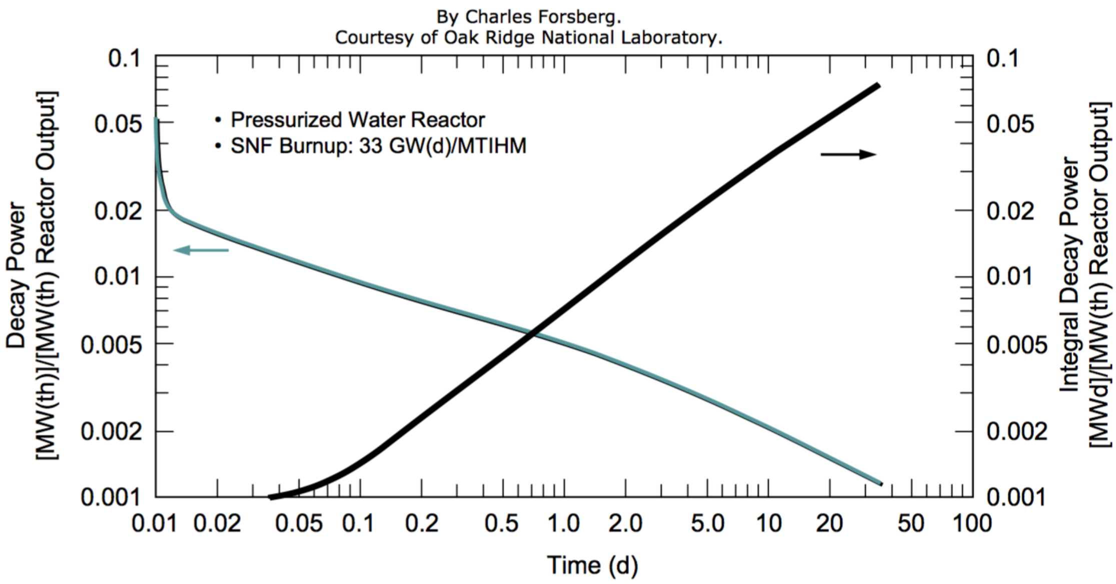


Figure 1
Decay power and integral decay power following shutdown.

†Because it is only ~1/15 of normal operating power it is perhaps easy to underestimate the absolute power and power density associated with decay heating. For a modern large PWR, it corresponds to something like 200,000 one-kilowatt electric heaters, operating in a volume similar to that of a domestic bathroom.

Confidence that the core will not undergo ballooning to the extent that coolability is impaired is thus vital, and some credible means to provide this assurance is necessary.

2.2 LOCA Accidents

There is obviously a large number of different LOCAs that could be identified for any one reactor type, varying as regards, for example, plant state at the time, fuel burn-up, and the size of the leak and the location of the leak. There are of course also many different reactor configurations, even within nominally similar families of reactor type.

We will here just give a very simple generic outline of the kinds of event that it is believed will take place, and will limit the discussion to the generic types of event expected in a large break accident in a PWR.

In Figure 2 is shown a typical four loop PWR, and in Figure 3 is shown a simplified diagram of the reactor cooling system. The classical large break LOCA is a double-ended guillotine break in the cold leg, between the reactor coolant pump, and the reactor pressure vessel; that is, one of the four horizontal sections of piping visible in the figure joining the pump to the reactor vessel. These are large pipes, typically 800 mm or more in diameter.

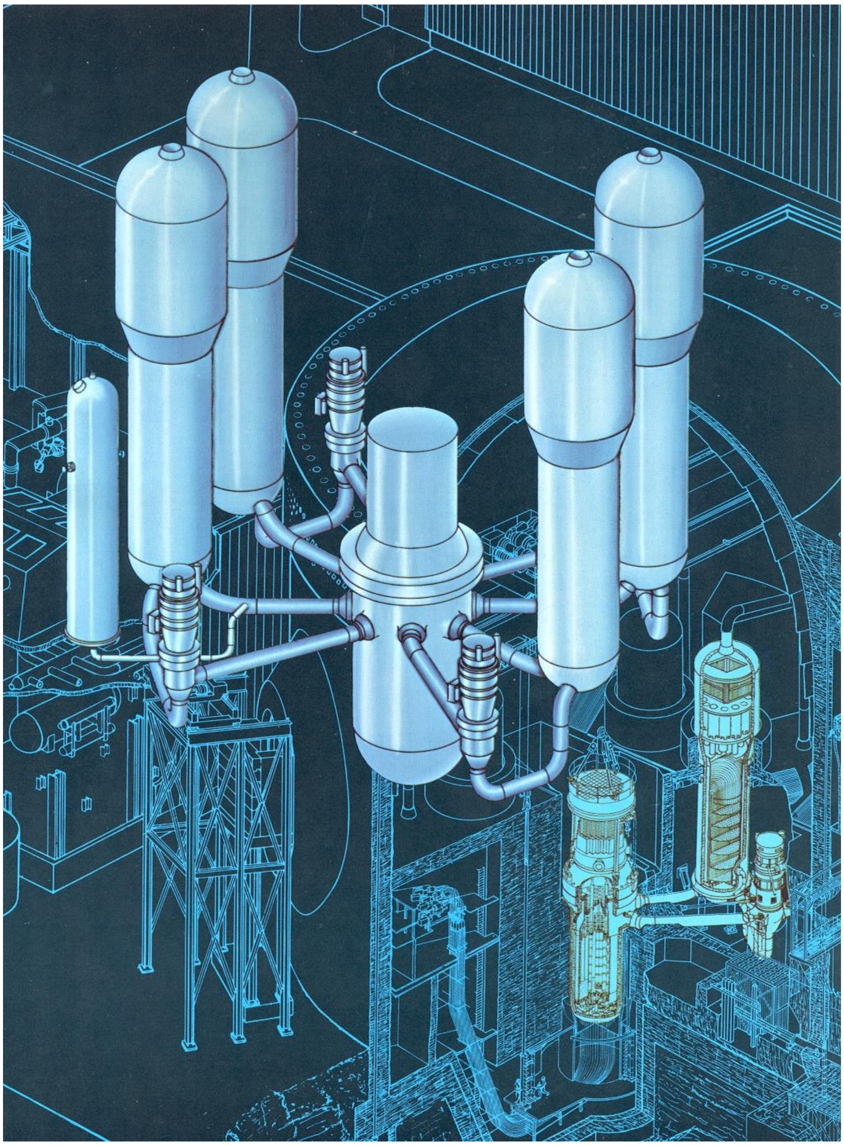


Figure 2
Typical four loop PWR[1]

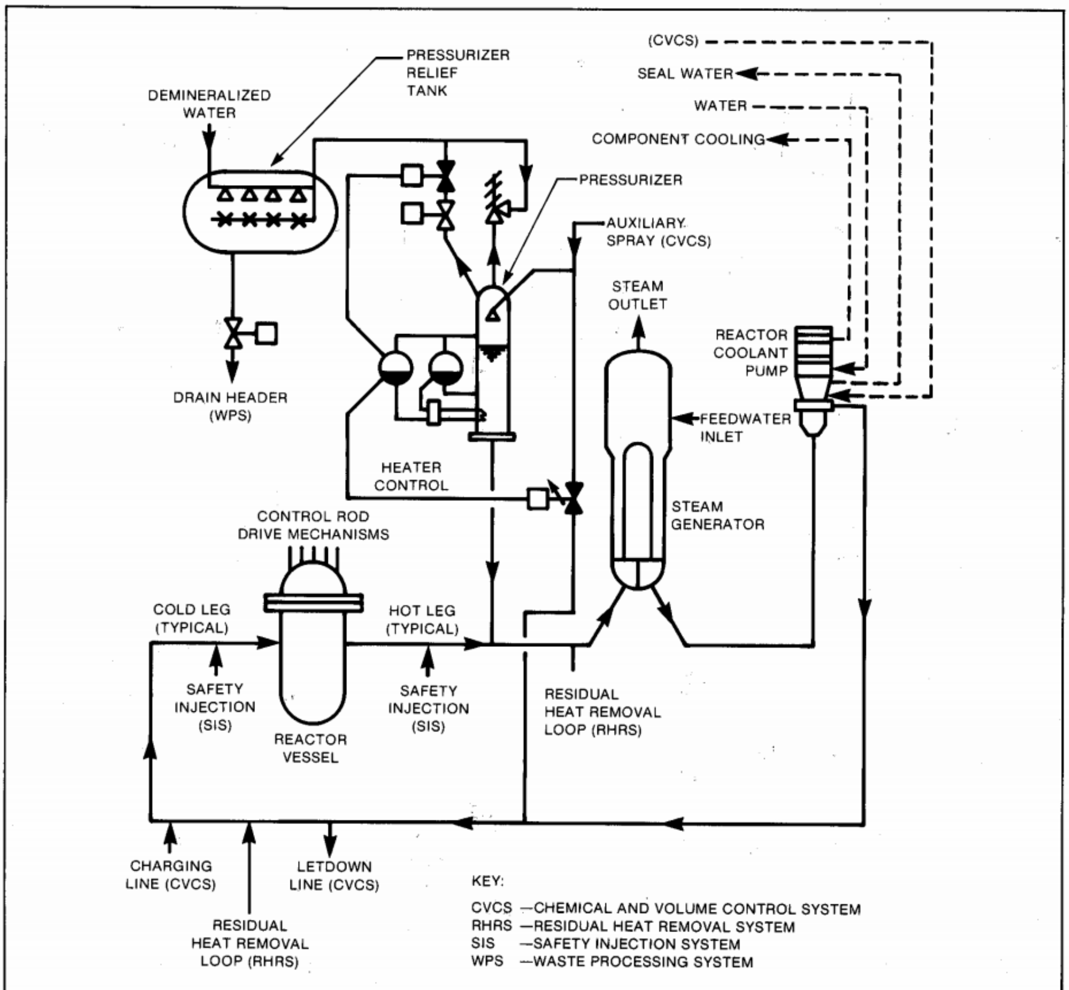


Figure 3-1. Reactor Coolant System, Flow Diagram

Figure 3
Reactor cooling system (from Westinghouse[1])

2.2.1 Large break LOCA

The stages are shown, indicatively, in Figure 4.

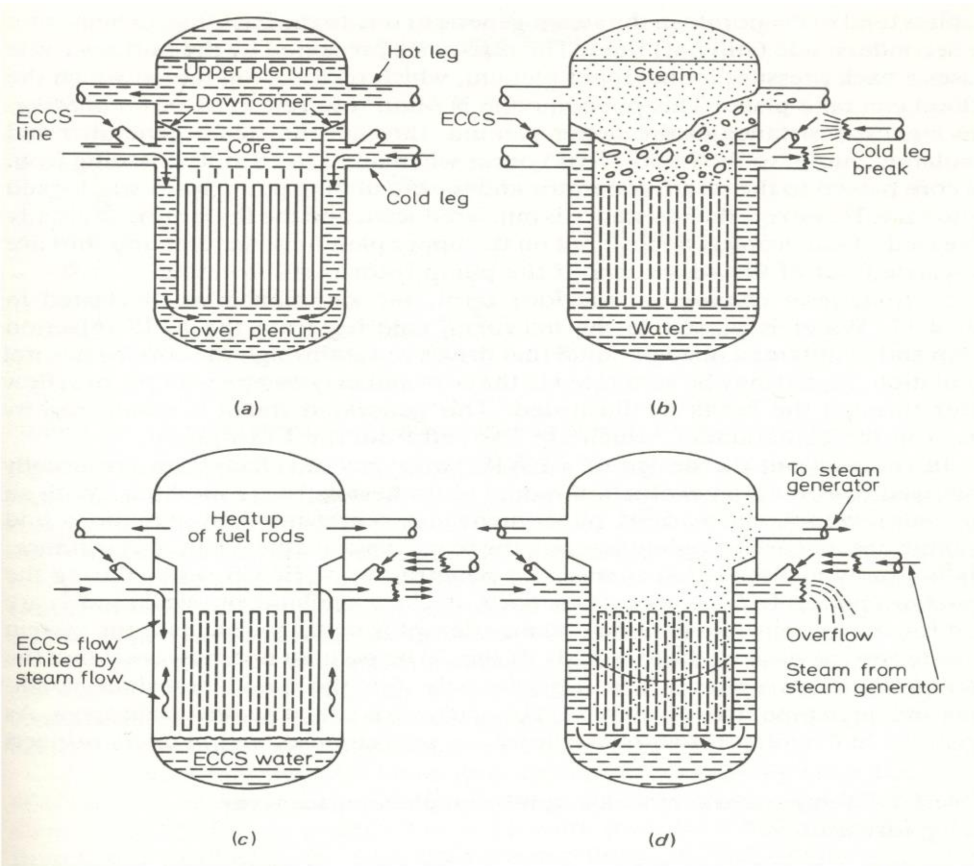


Figure 4
The processes in a notional LBLOCA (G F Hewitt, Personal Communication)

Rapid core voiding as water flashes to steam under the reduced pressure causes neutronic shutdown, while the central fuel probably goes through DNB. Reduced cooling of the fuel causes a flattening of the usual parabolic radial temperature distribution within the fuel, which in full-power, cooled operation would exhibit many hundreds of Kelvin temperature difference between centre and edge. (Cladding and pellet outer edge temperatures will be perhaps ~340C, with pellet centre temperatures ranging up to perhaps as much as 2000C). This leads to the outer parts of the fuel becoming hotter. Various coolant injection systems become activated at various pressures, and whilst much of this injected water may be swept out ('bypass'), liquid eventually begins to accumulate in the lower plenum. In particular, this refill commences when the pressure is low enough to trigger the low-pressure injection system. During this process the core is surrounded by (relatively) stationary steam, and is not far from adiabatic, and the fuel rises in temperature significantly. Once the liquid level climbs to the lower part of the fuel the phase termed 'reflood' begins. It is this reflood phase that is of most significance for present purposes, and we will discuss conditions under reflood further below.

Indicative times for these processes are shown in Table 1. Whilst these are indicative only (and obviously are rather uncertain for a given accident, and there are many possible accidents), the relative durations are perhaps of more significance for present purposes. In particular the reflood phase is long compared to the phases that precede it.

Phase	s
Bypass	20-30
Refill	30-40
Reflood	40-250
Long term cooling	250+

Table 1
Indicative times for the phases of a large break LOCA

In Figure 5, reproduced from NEA 6846[2], is shown a typical prediction of the evolution of pressures and temperatures during these events.

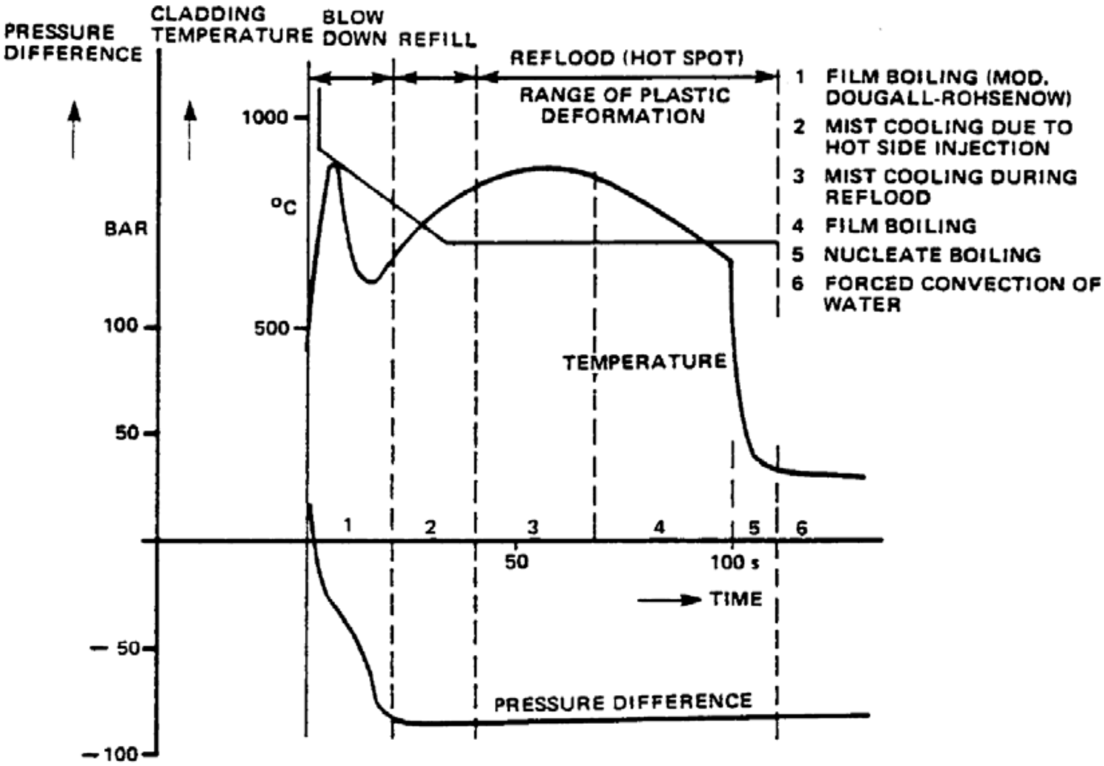


Figure 5
Double ended cold leg break, pressure difference across the cladding and cladding temperature at the hot spot. Reproduced from [2].

2.3 <u>The thermal hydraulic conditions during reflood</u>

At the beginning of the reflood stage we have essentially dry core subchannels, where the cladding is now very hot (far above, for example, saturation temperature). At the base of the subchannels we have a water level rising relatively slowly, with the liquid water struggling to make contact with (struggling to re-wet) this overheated metal.

We show in Figure 6 below a few stills extracted from a remarkable film, where this process of the entry of liquid water into a grossly overheated metal subchannel was reproduced experimentally. An electrically-heated tube had liquid water introduced to its base. The tube was located inside the core of a nuclear reactor, but only so that the nuclear reactor could serve as a source of neutrons that allowed the disposition of liquid and vapour to be filmed via neutron radiography. (Neutrons pass through the metal, but are absorbed sufficiently by the water to allow liquid and vapour to be distinguished. The metal tube walls are the light regions to either side, and the light and dark within the tube are vapour and liquid respectively.)

Vaporisation of some liquid does occur, and at these conditions vapour specific volumes are some hundreds of times those of the liquid, such that a large volumetric flow rate of vapour is produced.

These stills show the gradual movement upwards of liquid water, and show it unable to make wetting-contact with the walls. Liquid is be driven up as slugs and droplets entrained in the high-speed vapour flow.

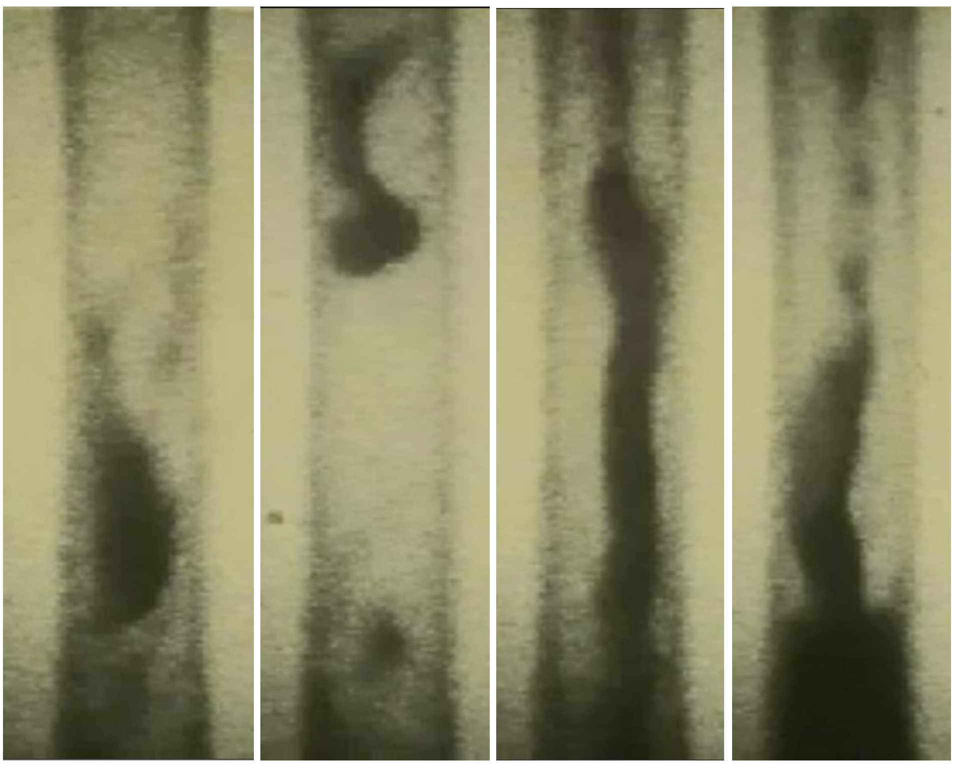


Figure 6
Stills extracted from the simulated reflood experiment of Hewitt

As the liquid water level rises, boiling occurs, and a flow of vapour, with entrained liquid water, rises up around the fuel rods. With vigorous boiling in slender channels, and at relatively low, and possibly even atmospheric pressure with the very large density ratio noted above, the generation and entrainment of slugs of water in the vapour flow is a highly chaotic process.

Well above the quench front, in the region where the highest cladding temperatures will be experienced, the conditions may be characterized as very hot cladding, cooled by a flow of superheated vapour, with entrained saturated water drops. These water drops play an indirect but very important role in the cooling, as they evaporate into the superheated vapour, and through their high enthalpy of evaporation keep the vapour temperature much lower than it otherwise would be. The droplet size distribution has a significant effect on the ability of the droplets to perform this role. This size distribution is a complex consequence of the slug generation and entrainment mentioned earlier, and of droplet evaporation, coalescence and breakup in their travel up the coolant sub-channel, and through the several spacer grids. In addition, whilst they do not wet it, the drops play some role by direct cooling of the surface as they bounce from it off the vapour cushions they generate as they approach.

Thermal-hydraulic models can only attempt to represent all this via empirical correlations, and the fidelity of such representations is naturally limited. Codes such as WCOBRA / TRAC, Relap and TRACE indeed have ‘Reflood models’ incorporated; complex sets of bespoke correlations and approximations, that attempt to model this complexity, from the chaotic generation of liquid slugs, through to the possibly mist-flow cooling. Details are available in the theory manuals of the respective codes.

Spacer grids play an important role, albeit one that is very difficult to quantify. These are only explicitly represented in WCOBRA/TRAC and related codes. They increase the turbulence and swirl of the vapour flow, but probably more important is their interaction with the entrained liquid. Being essentially unheated (unlike the cladding) they are cooler, and more easily wet. It is expected that droplets strike and wet the grids, and then the liquid from these grids in turn gets re-entrained. This has been explicitly represented in WCOBRA/TRAC and can be represented in other codes by adding heat-structure components to the model (but most licensing calculations omit this phenomenon).

The turbulence of the flow, and the droplet number density and size distribution, will all be rather different just downstream of the spacer grid from what they would have been otherwise, followed by something of a reversion to these conditions as the spacer grid is left further behind. (Effects such as this are taken into account in a semi-empirical fashion, indeed, in the models mentioned above.)

2.4 Mechanical conditions during reflood

2.4.1 The fuel

Analysis of the chemical, structural and mechanical changes taking place in oxide fuel during irradiation and burn up is obviously a vast subject in its own right. A ~500 page exposition is provided by Boyack et al[3] (but there is a large literature elsewhere), with the most relevant points reproduced in part in [2]. For present purposes perhaps the most important observation is that there will be a significant internal pin pressure from already-released fission gases. Depending on burnup, temperatures, and the duration of the accident under consideration, there could be further release of fission gas during the course of a LOCA. Whilst the internal pin pressure will be less than the PWR primary circuit pressure, with the reduction in primary circuit pressure during a LOCA the pin becomes subject to a significant net internal pressure. It is this that is the driving force for ballooning.

2.4.2 The cladding

Oxidation occurs for all zirconium-based alloys in hot water. Just how much oxidation will have taken place obviously depends upon when in the life of a particular fuel rod the accident in question takes place. The degree of oxidation does not depend significantly upon just which alloy of zirconium is involved provided that the transition time (at which the protective oxide film degrades) is not reached. However, the use of niobium in modern claddings does influence the transition of the material to the much softer beta phase, as does any pickup of hydrogen due to oxidation occurring prior to the fault (although the hydrogen effect is weak). The more modern materials (Zr-Nb-Sn (Zirlo) and Zr-Nb (M5) generally oxidize less than Zircalloy-4.

Oxidation has an influence during the fault, because the temperatures can become sufficient for oxygen to diffuse into the matrix of the zirconium and to produce a stabilized alpha phase region, which is substantially harder and more brittle than any material that has undergone phase transition. It is this process which determines the ability of the cladding to resist the thermal stresses associated with quench.

This process is little affected by any oxidation prior to the fault, but is weakly influenced by hydrogen pickup in the cladding.

Hydriding, broadly the absorption into the metal of some of the hydrogen freed up by the oxidation of Zirconium by water, has similar effects, reducing cladding ductility post-quench. At high temperatures, it has only a limited effect and will have little influence on cladding ductility during clad ballooning.

Radiation (displacement) damage is continuously slowly annealed at normal operation temperatures and plays no direct role in cladding behaviour during LOCA, as it is annealed-out quite early in the event.

All of this underlines the requirement that any analysis of ballooning during LOCA needs to be done using material properties and constitutive relations that adequately represent the condition of the cladding at the time of the LOCA.

2.5 The coupled effects of mechanical and thermal hydraulic behaviour during reflood

Given the capability to predict the thermal and mechanical response of a single pin (eg via a suitable fuel pin modelling code), all that is required are the temporal, axial and azimuthal variations of coolant temperature and heat transfer coefficient on the surface of the cladding. This would be sufficient for the pin modelling code to predict the ballooning response.

It is not of itself directly a "coupling" issue, but the provision of that level of detail, in particular regarding azimuthal variations, is not possible using the kinds of thermal hydraulic treatment normally employed, and which were alluded to above.

There is a further difficulty. As the fuel rods balloon, they change the geometry available to the flowing coolant, and thereby modify the coolant flows in the various sub channels. The cladding surface area available for heat transfer is also modified. Compounding this, since the flow is two-phase, with entrained saturated drops in a superheated continuous vapour phase, it is necessary to know whether or not this flow diversion also is associated with a change in droplet concentrations. In essence, if the vapour gets diverted because a subchannel is partially blocked, do all the droplets go with that diverted vapour, or do some carry straight on, leading to a droplet-rich flow past the blockage, and a droplet-poor flow diverted to an adjacent subchannel. Are the droplets big enough to be inertial?

2.6 High burn-up fuel in LBLOCA

In general, whilst high burn up fuel could have large internal pressures due to large cumulative gas release, in most core management schemes, it is likely to be operating at a relatively low rating, with low internal pellet temperatures. It is also for the same reasons likely to have relatively low decay heat.

Given this, the chance of significant ballooning being exhibited by high burn-up fuel is probably quite low and this is the basis of current safety cases within the UK.

However, after a threshold burnup, the behavior of the fuel pellet in response to a LBLOCA can be significantly different and this could result in changes to azimuthal temperature gradients and hence to the achieved diametral strain.

There is a tendency of fuel at these burnups to bond to the cladding and this could act to limit the stain achievable, but indications from the Halden IFA 650 series of tests are that high diametral stains are achievable. This is a topic of ongoing work, but since it is not important for current core designs, it is out of the scope of the current study.

2.7 Small break LOCA in the absence of HP injection

In the past it was common to have high pressure injection capability, which indeed meant that a small break LOCA presented less of a concern as regards clad ballooning than did a large break one.

There have been suggestions that new plant could be built without such high-pressure injection systems. In those cases, the response to a small break loss of coolant accident could well be very different from the response to a large break one.

In such a small break LOCA voiding is likely to be from the top down, with the upper regions of the fuel gradually being uncovered, and uncovered for long periods, before wetting and quenching finally occurs from below.

Under these circumstances, with the upper part of the pin being the part that is exposed, the local pin rating will tend to be smaller. However, the time for which it is exposed will tend to be rather long. Understanding the trade-off between the these two effects, one of which helps and one of which hinders, is difficult, particularly as the mechanical response of cladding to temperature and pressure is highly non-linear.

The need to consider these circumstances represents a significant broadening of the "classical" concerns and investigatory requirements for ballooning.

3 APPROACHES TO THE PREDICTION OF CLAD BALLOONING

3.1 Introduction

In subsequent sections we will discuss the various measurement programs that have been, and are being, undertaken to observe clad ballooning, and the various computational modelling tools that have been developed.

However, it seems useful first to discuss from a philosophical point of view what needs to be achieved, and what possible routes in principle are open to achieving it.

The starting point is that of a plant operator who needs to demonstrate with a high degree of confidence that frequent faults and design basis accidents would not result in significant fuel melting, or the breach of a significant number of fuel rods. Inter alia, that corresponds to his core retaining a coolable geometry, or equivalently to its not experiencing a degree of coherent ballooning such as to prevent adequate coolant flow.

It is important to bear in mind that the starting point for any ballooning analysis is itself the endpoint of a set of circumstances that are not really amenable to accurate mechanistic prediction; the chaotic and unsteady blowing down and partial voiding and then refilling of the primary circuit.

To compound this, the operator needs to demonstrate this confidence over a wide range of plant conditions (for example, burn up, fuel loading patterns, time into fuel cycle), and for a wide range of possible primary circuit breach sizes and locations. This requirement is driven by the fact that it is difficult to make a robust argument that any one accident provides the most challenging conditions, which would help in limiting the cases needing to be considered.

3.2 The role of measurements and modelling

Plainly, it is not really practicable to conduct full-scale prototypical tests at all, and certainly not for the wide range of conditions that would be needed if the aim were simply to rely upon demonstration that 'cool-ability' would always be maintained. Arguably, it might be possible to do this if one could identify a set of conditions that reliably provide the greatest challenge to the plant and the integrity of the fuel. One would then need to conduct a test of considerable fidelity, under these conditions. However, as noted above, for the present problem the subtleties of the behaviour are such that one cannot with confidence identify any circumstances that would reliably be "the worst case".

Claims relating to behaviour during these events must then be based upon predictions based upon first-principles, mechanistic modelling, augmented and complemented by appropriate measurements. This fundamentally moves the role of the measurements from being "show me how my plant will behave" to that of providing confidence that one's predictive tools have adequate fidelity.

Many physical phenomena and processes are important in determining the course of these events, and we will discuss these in more detail in the following section. As regards the measurements, what are needed is measurements made in a number of tests in which all of the principal physical phenomena expected to be important in real accident play a similar role. Not all tests will (or need) "exercise" all phenomena to the same degree, but a reasonably comprehensive coverage between the tests should be an objective in defining the tests. Similarly, whilst tests are likely to be small-scale relative to the real reactor, some analysis of scaling needs to be conducted to provide confidence that for these phenomena reasonable relevance to reactor conditions is still maintained.

An example may be helpful here. Let us assume that it is concluded that the hydrodynamic response of a flow of vapour with entrained droplets in a region of the core where its flow is obstructed by a balloon is an important phenomenon to be able to predict.

One could envisage an experiment in which such a flow was caused to occur in a rod bundle containing a ballooned region, possibly using air and water, under laboratory conditions.

The phenomena that are important include things like the response of the gas phase, and the response of the entrained droplets. Do droplets behave in an inertial fashion, and essentially try to carry straight on, or are they diverted along with the gas phase? How does this behaviour depend upon droplet size?

In parallel with this, computational predictive tools need to be developed, incorporating as much as possible mechanistic treatments of the physical phenomena expected to be important.

A predictive tool for the clad ballooning problem would be expected, amongst many other phenomena, to be able to predict reasonably reliably vapour flow and droplet diversion. That aspect of its performance would have been able to be compared to the measurements discussed above.

Whilst there is a continuum, and indeed to a degree both purposes may be met to differing degrees within any one experiment, it is worth reiterating the comment above. This approach shifts the main emphasis of an experimental programme from "See how my plant in aggregate will behave", to a program designed to provide validation, on a phenomenon by phenomenon (or 'separate effects') basis, of a comprehensive model that it is hoped will predict aggregate behaviour.

3.3 <u>What phenomena need to be taken into account</u>

We will not address the earlier stages of the accident, with choked flashing flow through uncertain breach sizes, and all the consequent unpredictable events. We will focus on phenomena during the re-fill and re-flood phases. Broadly, conditions may be characterised in the refill phase as low flows of steam, and in the reflood as a vigorous flow of steam with entrained liquid. The former possibly presents a greater challenge to the clad, although this depends on durations, time to heat up and so on. The latter presents a greater challenge to the modeler and experimenter. Much of what follows is by default applicable to both, but we will focus on reflood, as, broadly, if we can find ways to model and predict it, we can probably be confident we can do so for refill.

3.3.1 Creep and rupture

Cladding plainly can creep, and may well rupture. Good models of the creep and rupture behaviour of the cladding are required, where proper account is taken in these models of other processes that modify this, such as oxidation or radiation embrittlement and so on.

3.3.2 Pellet eccentricity

There is in general a clearance between the inside surface of the cladding and the external surface of the pellet, and there necessarily is such clearance as soon as the cladding begins to creep. In general this radial gap will be azimuthally non-uniform, but there is no way of knowing in just what azimuthal location the gap will be smallest, or even non-existent, and opposite which will be presumably a region where the gap is a maximum.

Since the size of this gap can have a noticeable effect on heat transfer and clad temperature, and since creep rate is a very sensitive function of cladding temperature, there is a tendency for creep to be localised in the azimuthal location where the pellet-clad gap is smallest. Such localised strain could lead to cladding failure at average perimeter strains that are much lower than would the case if the azimuthal distribution of clad temperature and strain was uniform.

3.3.3 Anisotropic cladding behaviour and the 'hot side straight'

The 'hot side straight' effect is a consequence of the anisotropy of the cladding material. Hoop strain is accommodated in part by an axial flow (=axial shortening) towards the straining location. If the hoop strain is azimuthally non-uniform (because of an azimuthal temperature variation) this side (azimuthal location) is shortened, tending to straighten it, and move it back towards the pellet stack. This of course causes further heating, with positive feedback. This effect is greatest in the range ~725 – 775C (for Zircalloy), but declines above this as phase change occurs[4].

3.3.4 Other sources of azimuthal non-uniformity

There are various other sources of azimuthal non-uniformity that can have similar effects. Even in an otherwise perfectly uniform and infinite lattice, there is a degree of azimuthal non-uniformity in conditions around a pin arising simply from the lattice geometry (think of the cruciform shape of a pin-centred subchannel). However, in a "real" assembly there are various additional sources of non-uniformity. Some pins will be more influenced by un-fuelled control rod guide tubes and so on, and others will be adjacent to a subassembly edge (or indeed corner). Almost all pins will be influenced at least to a degree by macroscopic radial (across the core) variations of flux and rating, which in turn will generate azimuthal variations for an individual pin.

3.3.5 Stable and unstable deformation

Whilst unstable creep can occur, localized at a particular azimuthal location, there are factors that could lead to negative feedback, with more stable creep and the development of azimuthally more uniform strain. We are here primarily interested in conditions where the cladding is being heated at its inner surface by the pellet, and is being cooled at its outside by a flow of steam. Azimuthally localised straining will increase the pellet-cladding gap, and this is likely to intensify of heating experienced by the inner surface of the cladding at this location. The high stress exponent for dislocation creep (around five) also leads to a tendency for the stain rate to accelerate and concentrate at the hot spot; leading to short balloons of limited diametric strain.

Competing against these beneficial effects is the effect of external cooling of the cladding, which is complex.

If the ballooning is causing increases in vapour flow speed, rather than gross reductions in vapour mass flow rate associated with near-blockage conditions, external cooling is likely if anything to be improved. If the effect of temperature changes is stronger than the effect of stress changes, then there is the potential to arrest or slow the ballooning at the affected location.

The net result is that cladding that has already ballooned at one axial location may tend to a reduced rate of subsequent strain there, with strain instead preferentially occurring at other axial locations. This could have a stabilising effect on the deformation, leading to larger average strains before failure, and accordingly higher blockages

3.3.6 Turbulent flow

Heat transfer during the ballooning process is essentially one of single-phase convection from the surface of the pin to the vapour, followed by heat transfer from the vapour to any entrained droplets. Prediction of turbulent single component flow is heavily dependent upon empirical relationships. In practice it is routine that relationships are used to predict flows in circumstances that are quite different from those involved in the developing of the relationship. Whilst that is quite widely true, under the circumstances at issue here, with the characteristics of the turbulence probably modified considerably by the presence of a fairly dense droplet field, it probably contributes a rather greater uncertainty than normal.

3.3.7 Flow regimes, and droplet behaviour

Even limiting ourselves to channels being re-flooded from the bottom up, there is considerable uncertainty over just which flow regimes become established, and when and where. (The stills from the neutronic film give some indication of the kinds of flow that might obtain relatively close to the re-wetting front.) It is probably reasonable to assume that the flow progresses to become essentially a mist-like flow, with some distribution of sizes of droplet entrained in a rapid flow of superheated vapour. There is then a need to consider the (Weber number dependent) breakup of droplets due to shear stresses and fuel wall or spacer grid collisions, and droplet coalescence through droplet-droplet collisions. Simultaneously with this, there is a steady process of droplet evaporation as the droplets are heated by the superheated vapour. Just how large is the effective heat transfer coefficient for this, and how does it depend upon (say) the Reynolds number, or the droplet size (let alone shape, which is anything but spherical for larger droplets, is not mechanistically predictable.

As the bulk vapour flow is caused to change direction, either by a spacer grid, or perhaps by encountering a region of flow blockage, there is uncertainty over the degree to which entrained droplets "go with the vapour", or, because of their greater density (momentum), have a tendency to carry on in the direction of original travel.

Droplet size has a big effect upon droplet behaviour. We have noted above that droplet behaviour can be inertial, or they can go with the flow; which they do is largely dependent on their size. Evaporative heat transfer from the droplets to the superheated vapour is also strongly dependent upon droplet size. Large droplets, per unit mass, probably contribute little fresh vapour, and consequently little cooling of the extant superheated vapour, simply because they have relatively little surface area per unit mass. Heat transfer and evaporation increases as droplets size decreases, but there is probably a point below which droplets become so small that they move so readily with the flow of vapour that the flow of vapour relative to the droplets becomes too small for rapid evaporation to occur. Just what droplet size distributions exist, and what are the sizes at which these effects have just which relative strengths, is in truth not very well known. Nonetheless, they are all important for reflood and ballooning analysis.

3.3.8 Fuel rod to fuel rod heterogeneity

Whilst they may be nominally identical, there are likely to be significant rod-to-rod differences. Some of these will be systematic, and some stochastic. This issue was discussed a little above, where the differences in burn up and cooling around any individual fuel rod caused by proximity to unfuelled locations, or to subassembly edges, was mentioned. There will be stochastic contributions also, arising from manufacturing tolerances in pellet and tube, and in fill gas pressures.

3.3.9 Coupling and feedback

As cladding balloons the cross-sectional area available for flow in the associated subchannel of course is reduced. If all fuel rods ballooned identically and simultaneously there is still of course feedback to the flow, as vapour speeds will be increased, and probably as a consequence droplet size distributions change. Indeed, cooling could well become more effective, with higher vapour heat transfer coefficients, but the effect on droplet size and associated cooling is less clear-cut.

However, the circumstances are rather more complicated than that.

If a rod balloons and its near-neighbours do not, or if they balloon but to a lesser degree, there is likely to be a redistribution of flow from the more blocked channel to those nearby offering an easier passage. For similar reasons to those mentioned above, cooling of the subject rod and of those adjacent ones will be affected, and with the net result being that the differences in balloon behaviour might be accentuated, or might be damped out. Since there are competing, contrary, effects, it is not really possible to assert one or the other in general.

What one can assert with confidence is that feedback between rods could occur, and this needs to be investigated by measurement and modelling, if only to dismiss it, but more probably for it to be included in models as mechanistically as possible.

3.3.10 NUREG 0630

At least to a degree, an alternative approach is to follow the rules presented in NUREG 0630 as long as one's Regulator is prepared to accept them. In essence, following a series of experiments, the NRC produced correlations that could be used to estimate cladding rupture temperature, a (bounding value for) cladding burst strain, and a bounding value for subassembly flow blockage.

In effect the empirical NUREG-0630 model ignores the cladding creep, but rather proposes a relation between the mechanical load on the un-deformed cladding and the rupture temperature of the cladding.

There is a fundamental philosophical problem with this approach, which we will note before we move on to more numerical and pragmatic ones. All of the quantities involved are coupled; hoop stresses depend upon internal pressure, which in turn depends upon hoop strain, and so on and so on. We cannot really compute one in isolation, and then use the correlation to determine the second.

Even if we accept this, there are still various other problems about the use of this approach. The correlations were obtained empirically from experimental data, which in turn was largely obtained for relatively rapid heat up, under conditions of low heat transfer. Concerns outside the United States were more focused upon ballooning in the refill and reflood phases of the LOCA where there is rather more heat transfer, and correspondingly slower rates of temperature rise. That further reduced the confidence with which the NUREG correlations could be applied.

There is a further problem with the fundamental soundness of a stress-failure criterion. Indeed, ballooning tests at the Berkeley laboratories at constant hoop stress demonstrated that the strain was essentially independent of stress in the LOCA conditions. In the limit, the NUREG criteria would allow one in principle to control the hoop stress to be below the failure stress and totally inhibit failure at any strain. That work suggested that strain-based failure criteria were the most credible, and they were incorporated into codes like CANSWELL.

3.3.11 Closing remarks

The above processes all contribute to making achieving it difficult, but in essence there are two things that we need to know:-

-We need to be able to predict the temperature of the cladding.

-We need to be able to predict the mechanical response of the cladding on any particular pin to a given temperature history.

To add to the complication, these two phenomena are of course linked.

The alternative, of a simple essentially empirical approach such as NUREG0630, has problems in principle, and doubts over the applicability of the conditions under which much of its data was gathered.

4 THE LITERATURE

There is an enormous literature addressing the above areas.

The loss of coolant accident, as the classical design basis accident, has of itself of course generated a large body of previous work.

Within that, the issue of clad ballooning has been studied more or less continuously for perhaps 30 years, with a large number of experimental programs having been conducted. In parallel with this, for ballooning, but for more general purposes, there has been a major programme in the development of computational tools for the general modelling of fuel rod (pellet and cladding) behaviour.

It is plainly neither helpful, nor possible, for us to produce anything approaching a comprehensive review of even the fairly focused "ballooning" related work here.

However, we are fortunate that there do exist several quite comprehensive and authoritative reviews of the position.

We will refer to them in more detail as we go through, but it is worth highlighting a small number of documents at this stage.

4.1 Nuclear fuel behaviour in loss-of-coolant accident (LOCA) conditions[2]

The NEA Working Group on Fuel Safety (WGFS) is tasked with advancing the current understanding of fuel safety issues by assessing the technical basis for current safety criteria and their applicability to high burn-up and to new fuel designs and materials. As part of this, in 2009 the WGFS produced a comprehensive 'state of the art' report (itself an updating and expansion of a similar 1986 document).

This is a ~370 page document, with 20+ authors, addressing the accidents at issue, the phenomena expected to be important, the experimental programs intended to provide evidence, and the status of associated modelling efforts. It cites about 450 references. It provides a comprehensive overview, up to the time of its release, ~2009.

4.2 Update to NEA6846

There are now plans to update this 2009 document, and Tim Haste, of IRSN, has recently been tasked with coordinating the writing of its successor. From the UK, John Lillington, of AMEC-Foster Wheeler, and Simon Walker (Imperial College) have been asked to contribute to this re-writing. The first working meeting of the activity will take place early in 2017 at Imperial College. As with the earlier document, this will be a major effort, by probably 20+ workers, spread over ~12 months.

4.3 Grandjean: 'Review of programs ..fuel behaviour under LOCA ..' [5-7]

Slightly earlier than the NEA report, Grandjean of IRSN produced a three-part review covering similar ground. Part I[5] addressed 'Clad swelling and rupture assembly flow blockage', Part II[6] the 'Impact of clad swelling upon assembly cooling', and Part III[7] 'Cladding oxidation; Resistance to Quench and Post-Quench Loads'.

5 IDENTIFICATION CHARACTERISATION OF EXPERIMENTAL PROGRAMMES

5.1 Introduction

Starting in the late 70's, and still underway now, there has been a vast effort to characterise experimentally the behaviour of LWR fuel under LOCA / ballooning conditions, with some test series lasting over many years. It is not practicable to even to mention them, let alone to review them all. However, as noted above there are some very large recent reports that do a good job of summarising these, and we will provide references as appropriate. The 2006 NEA report, for example, has a 56-page section on test programmes, including some 100 references.

In the sections that follow we will identify the main categories of tests, and highlight what seemed to us to be some of the most important amongst these.

We present a summary listing of the main tests. There are various ways these could be categorised and subdivided.

As an 'outermost' categorization we take tests (a) in which a pin, or pins was caused to balloon, and (b) tests in which cooling of a fixed geometry representative of a ballooned bundle was studied. The first category we subdivide into electrically heated, and in-pile nuclear-heated tests. As appropriate, we employ a final subdivision into single rod and multi-rod cases. Not all possible branches of this tree are populated, of course. Also, some test do not fit into this categorisation, or span multiple categories. Nonetheless, we hope it is a helpful guide.

Dynamic ballooning				Fixed geometry
Electrically heated		In-pile nuclear heated		Electrically heated
Single rod	Multi-rod	Single rod	Multi-rod	Multi-rod

Table 2
An attempt to depict the categorisation applied to the various LOCA / ballooning-related test programmes

We have excluded from the above categorization 'simple' material behaviour tests. A necessary part of any predictive capability for fuel behaviour during LOCA must of course be knowledge of the behaviour of the cladding under relevant conditions. A fuel modelling code requires accurate physical properties data, to compute the (azimuthal variation of) cladding strain, (possible) eventual cladding rupture, and subchannel blockage.

There have been many 'separate effects' tests measuring quantities such as oxidation rate, diffusion rates, phase transitions, and ductility creep of various cladding materials. The information from these of course forms the cornerstone of any structural mechanics predictive tool. It is important that as either new materials are introduced (Zirlo, M5), or existing materials are taken to more demanding conditions, that the parameter space covered by tests such as these is adequate to cover these new conditions.

There are some tests that, whilst not integrated tests, are not strictly separate effect tests either. These include for example tests where the deformation of cladding material is caused to take place in steam, where the oxidation that steam causes can have a significant effect on the mechanical response.

Compared to the integrated tests that will be discussed below, the separate effect tests are relatively simple and cheap; unglamorous, but necessary.

As noted above, a fairly comprehensive summary, covering the period up to 2006, is provided by NEA6846[2]. The tests in this category are generally discussed in Section 5.1 of that document, with subsections devoted to creep (5.1.1) and ductility (5.1.4). Oxidation, diffusion constants and phase change are covered in 5.1.2, 5.1.4 and 5.1.5 . The complexity of even 'straightforward' material property determination is clear: creep and rupture, for example, is variously influenced by irradiation and oxidation, and temperature dependent phase change.

We now present the categorized listing in Sections 5.2 to 5.4 . A summary tabulation is provided in Table 3.

5.2 Dynamic ballooning, electrically heated

5.2.1 REBEKA (ISP 14)

- 5x5 bundle (max. flow blockage 25-84%)
- 7x7 bundle (max. flow blockage 52-66%)
- Electrically heated, German PWR design (3.9m heated length)
- 8 spacer grids and enclosed in a non-heated shroud
- Reflooding conditions common to German PWR of KWU (Kraftwerk Union) design
- Temperature rise 7K/s, coolant flow stream ~2m/s
- System pressure: 4bar
- Performed in Kernforschungszentrum Karlsruhe (KfK)
- References: [5, 8-10]

5.2.2 MRBT at ORNL

- 1977-1981
- Multi-Rod Burst Tests at Oak Ridge National Laboratory
- Steam atmosphere with very low downstream flow
- Electrical simulators with internal heating (0.915m heated length)
- 3 spacer grids
- 4x4 rod bundle
- With and Without heated shroud (direct resistance heated)
- Temperature rise ~9.5 - 30K/s
- Max. Temperature before burst: 850-880˚C
- 6x6 rod bundle
- 8x8 rod bundle
- Unheated shroud
- Temperature rise ~9.8K/s
- Max. Temperature before burst: 775˚C
- Reference: [2, 5]

5.2.3 JAERI

- late 70s to early 80s
- Japanese Atomic Energy Research Institute
- 7x7 simulators of 15x15 PWR rods
- Steam atmosphere with very low flow (conditions of large deformation)
- Internal pressure 20-70bar
- 2 spacers with both heated and unheated shroud
- Unheated shroud
- Temperature rise 5.9 – 9˚C/s
- Internal pressure 20-91kg/cm
- Burst Temperature 750-920˚C
- Guard ring on unpressurized heaters
- Initial pressure 50bar
- Temperature rise 7K/s
- Shroud backed by guard heaters
- Initial pressure 50bar
- Two Heating rates: 1 and 7K/s
- Reference: [5, 11-13].

5.2.4 KWU Erlangen

- Multi-rod tests in Erlangen
- Investigation whether single rod results are influenced by geometrical and thermal conditions in a multi-rod arrangement
- 3x4 array of rods
- Pressurized with Helium (65bar)
- Temperature rise 10-20K/s to about 800˚C
- Reference: [14, 15]

5.2.5 Westinghouse Electric Corporation

- Single and multi-rod testing
- 4x4 array (15x15 size, Zircaloy 4 tubing)
- Internal pressures: 14 and 155bar
- Maximum blockage ~50%
- Reference: [14, 16]

5.2.6 Springfields Nuclear Laboratories

- 4x4 array contained in a shroud
- Heated to ~700˚C
- Internal Pressure: 7.9MPa
- Reference: [14, 17]

5.2.7 PARAMETR-M and TEFSAI-19

- 1999-2001
- Russia
- 19 and 37 rod VVER type assemblies
- Electro-heated and passive rods
- Initial Helium pressure: 2-4MPa
- Initial Temperature: 450˚C
- Heating rate: 0.2-2.5K/s
- Maximum Temperature: 900-1150˚C
- Reference: NEA6846 report [14, 18]

5.2.8 KIT

- QUENCH-LOCA series, began 2010, ongoing
- 21 electrically-heated rods
- Multiple advanced claddings
- Ballooning, H uptake, detailed PIE
- Reference: [19]

5.3 Dynamic ballooning, in-pile nuclear heated

5.3.1 MT tests

- NRU reactor at Chalk river, Canada
- PNL
- US NRC and UKAEA
- Full-length PWR fuel rods 17x17type (fuel length 3.66m)
- Tests MT-1 to 4
- Fresh UO2 fuel conditioned by raising to full power three times
- Temperature rise: ~8K/s
- 6x6 array with 4 corner rods removed
- References: [14, 20-24]

5.3.2 PHEBUS-LOCA

- 1980-1982
- 5x5 fuel rod bundle
- 17x17 PWR type rods (active length 800mm)
- Real fuel containing fresh UO2
- Unheated shroud
- Initial Pressure: 4MPa
- Reference: [14, 25-27]

5.3.3 Halden reactor

- High burn-up tests at Halden Boiling Heavy-Water reactor

- 1980s: IFA-511.X, IFA-54.X:
  - Nuclear heating and electro-heated
  - 7 rods in circular configuration (IFA-511.X)
  - 5 rod bundle in form of a cross, 3x3 square array (IFA-54.X)

- 2006: IFA-650.X:
  - Pre-irradiated fuel rods with high burn-up
  - Rods filled with gas mixture of 5% Helium and 95% Argon
  - Internal pressure: 40bar
  - System pressure in the loop: ~70bar
  - Heated length: ~518mm
- Reference: [14, 28, 29]

5.3.4 PBF-LOC Tests

- Early 1980's
- Power Burst Facility, INL
- Four single pins per test
- 15x15 PWR type rods (active length 910mm)
- Fresh & irradiated UO2
- Experiments covered alpha, alpha-beta transition, and beta phases
- Reference: [30, 31]

5.3.5 FR2 Single Rod Tests

- Early 1980's
- Fresh & irradiated UO2
- In the FR2 reactor at KfK
- 25 to 125 b internal pressurisation
- Some parallel electrically-heated rods alongside for comparison
- Reference: [30, 31]

5.4 Fixed geometry, 'coolablity'

5.4.1 FEBA and SEFLEX

- KFK Karlsruhe

- **F**looding **E**xperiments with **B**locked **A**rrays
- 1x5 row of rods and 5x5 rod arrays
- Fuel rod simulators, 3.9m long, held by 7 spacer grids, blockage simulated with hollow stainless steel sleeves
- Axial power cosine profile approximated by 7 power steps
- Heated shroud (heated by radiation from heater rods for two hours to reach initial conditions)
- Blockage ratios 62 and 90%
- 600-800˙, reference conditions: 4bar and reflood rate of 3.8cm/s
- Reference: [6, 32]

Fuel Rod Simulator Effects in Flooding Experiments

- Evaluation for sensitivity of FEBA reflood tests
- Reference: [6, 33]

5.4.2 THETIS

- United Kingdom Atomic Energy Authority (UKAEA) at Winfrith Atomic Energy Establishment
- 7x7 rod array with 4x4 group of rods containing blockage region, heated length of 3.58m, ballooning simulated by superimposing pre-shaped Inconel sleeve
- Square shroud not directly heated but raised to equilibrium temperature
- Fuel rod simulators
- Severe blockage of 80 or 90%
- System pressure: ~2-4bar
- Reflood rate: ~1-6cm/s
- Inlet Temperature: ~50-100˙C
- Power: ~100-200kW
- Reference test: 2.1bar, 2cm/s, 88˙C, 99kW
- For reflood rate of 2.9cm/s maximum balloon temperature: 755˙C
- Reference: [6, 34]

5.4.3 ACHILLES

- Late 80s at the Winfrith Atomic Energy Establishment AEEW
- PWR dimensions
- 69 fuel rod simulators (electric) in square array, length 3.66m
- partial blockage in a 4x4 group, blockages created by hollow pre-shaped Inconel sleeves
- Blockage ratio: 80%
- Axial power profile approximated to truncated cosine in 11 steps
- Reference: [6, 35]

5.4.4 CEGB

- **C**entral **E**lectricity **G**enerating **B**oard in Berkeley Nuclear Laboratories
- 44 rod bundle, Electrically heated, heated length 1m
- 61% or 90% blockage in the central 4x4 group of rods
- Blockage simulated by different Inconel sleeved configurations
- Cylindrical shroud
- Initial temperature: 600-800˚C
- Cold reflood rate: 10-50mm/s
- Reference: [6, 36]

5.4.5 FLECHT SEASET

- 1977
- **F**ull **L**ength **E**mergency **C**ooling **H**eat **T**ransfer – **S**eparate **E**ffects **A**nd **S**ystem **E**ffects **T**ests
- Blockage simulated by non-concentric sleeves
- Cooperation USNRC, EPRI and Westinghouse
- 21 fuel rod array
- Reflood rate: 1.27-15.2cm/s
- Pressure: 1.4-2.8bar
- Inlet fluid temperature: 22-78˚C
- Initial peak linear power: 0.89-2.57kW/m
- References: [6, 37]

- 163-rod array
- Reflood rate: 1.52-15.2cm/s
- Pressure: 1.4-4.2bar
- Inlet fluid temperature: 53-122˚C
- Initial linear power: 1.3-3.3kW/m
- Initial cladding temperature: 260-871˚C
- References: [6, 38]

5.4.6 PERFROI

- 2016
- French (IRSN)
- COAL experiments (**CO**olability of a fuel **A**ssembly during **L**oca)
- (7x7, 49 rods, 16 deformed rods, height 3m, electrically heated rods, fuel relocation simulated by local power increase)
- BENSON facility at AREVA Erlangen
- References: [39, 40]

5.4.7 KAERI

- 2015
- Intact bundle: 6x6 array
- Ballooned and fuel relocated bundle: 5x5 array with 9 deformed rods
- Experimental conditions: Intact/Ballooned/Fuel relocated: 2, 4, 6bar at reflood rates of 2, 4, 6cm/s, Initial maximum temperature: 600~700˚C, Coolant fluid temperature: 30~80˚C, Average linear power: 0.5~1.5kW/m
- 5x5 rod bundles, 2x2rod bundles

- References: [41, 42]
-

5.4.8 PAKS

- 2003
- An inadvertent test, in an inadequately-cooled fuel pond
- Full assemblies (E110 cladding)
- Low assembly power (6 – 8 kW each)
- VERY long heat up and ballooning times; ~5 hours, not really reactor-relevant
- Very long balloons formed, and clad failures, some fragmentation
- References: [41, 42]

Dynamic ballooning: Electrically heated

Name	Location	Date	Reference
Springfields	Springfields	1970's	[14, 17]
Westinghouse	Westinghouse	1970s late	[14, 16]
JAERI	JAERI	1970s & on	[5, 11-13]
MRBT	ORNL	1977 - 1981	[2, 5]
KWU Erlangen	Erlangen Germany	1980's	[14, 15]
REBEKA	KfK Germany	1980's (early)	[5, 8-10]
TEFSAI-19	Russia	1999	[14, 18]

Dynamic ballooning: In pile nuclear heated

Name	Location	Date	
MT	NRU, Chalk River Canada	1980	[14, 20-24]
PHEBUS-LOCA	Caderache	1980	[14, 25-27]
PBF-LOC	INL Power Burst Facility	1980's	[30, 31]
FR2	KfK FR2 Reactor	1980's	[30, 31]
Halden	Halden BHWR Norway	1980, 2006	[14, 28, 29]
Paks	Hungary, fuel pond	2003	[43, 44]

Fixed geometry: Electrically heated

Name	Location	Date	
FEBA and SEFLEX	KfK Germany	1984+	[6, 32, 33]
FLECHT SEASET	USNRC, EPRI, WEC	1977	[6, 37]
CEGB	CEGB	1980's	[6, 36]
THETIS	UKAEA Winfrith	1983	[6, 34]
ACHILLES	UKAEA Winfrith	1989	[6, 35]
KAERI	KAERI	2015	[41, 42]
PERFROI	IRSN at AREVA Erlangen	2016	[39, 40]

Table 3
Categorised summary tabulation of some of the main test programmes

6 SAMPLE EXPERIMENTAL PROGRAMMES

Following the same broad categorization as above, we will mention a sampling of experimental programmes.

6.1 Dynamic ballooning, electrically heated

6.1.1 REBEKA Single Rod and Multi-Rod[10]

These were electrically heated out of pile tests on single rods, and on 5x5 and 7x7 bundles of ~full size rods, complete with spacer grids, with steam cooling. A good summary of the objectives of the series, the experimental arrangements, and of the results, is provided by NEA6846[2], page 269, and in Grandjean Part I, Sections 3.1.2 (single rods) and 3.2.1 (bundles).

We will in general not show results, as we do not purport to be producing a summary, but one or two observations and figures are useful here as an indication of the kinds of measurements and observations that are typical of such tests. We will not discuss or characterise them (this is done in eg Grandjean Part I, and of course in the original reports), but show in Figure 7 the axial distribution of circumferential strain, and the blockage ratio, for each of multiple rods in the bundle studied.

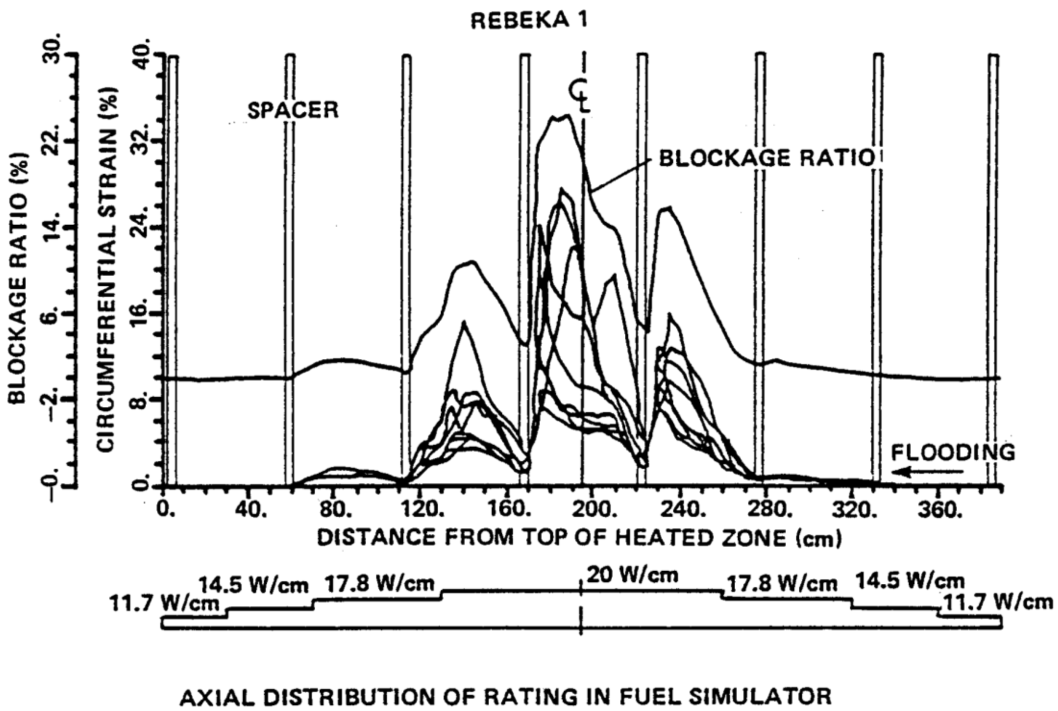


Figure 7
Typical axial variations of final cladding strain for each of the nine inner rods in a REBEKA test, and the aggregate blockage fraction.

Azimuthal thinning and rupture are shown in Figure 8. *Inter alia*, this shows the 'hot side straight' effect discussed earlier. The importance of the azimuthal temperature differences is clear from Figure 9, with a ~50 degree difference roughly halving the average burst strain.

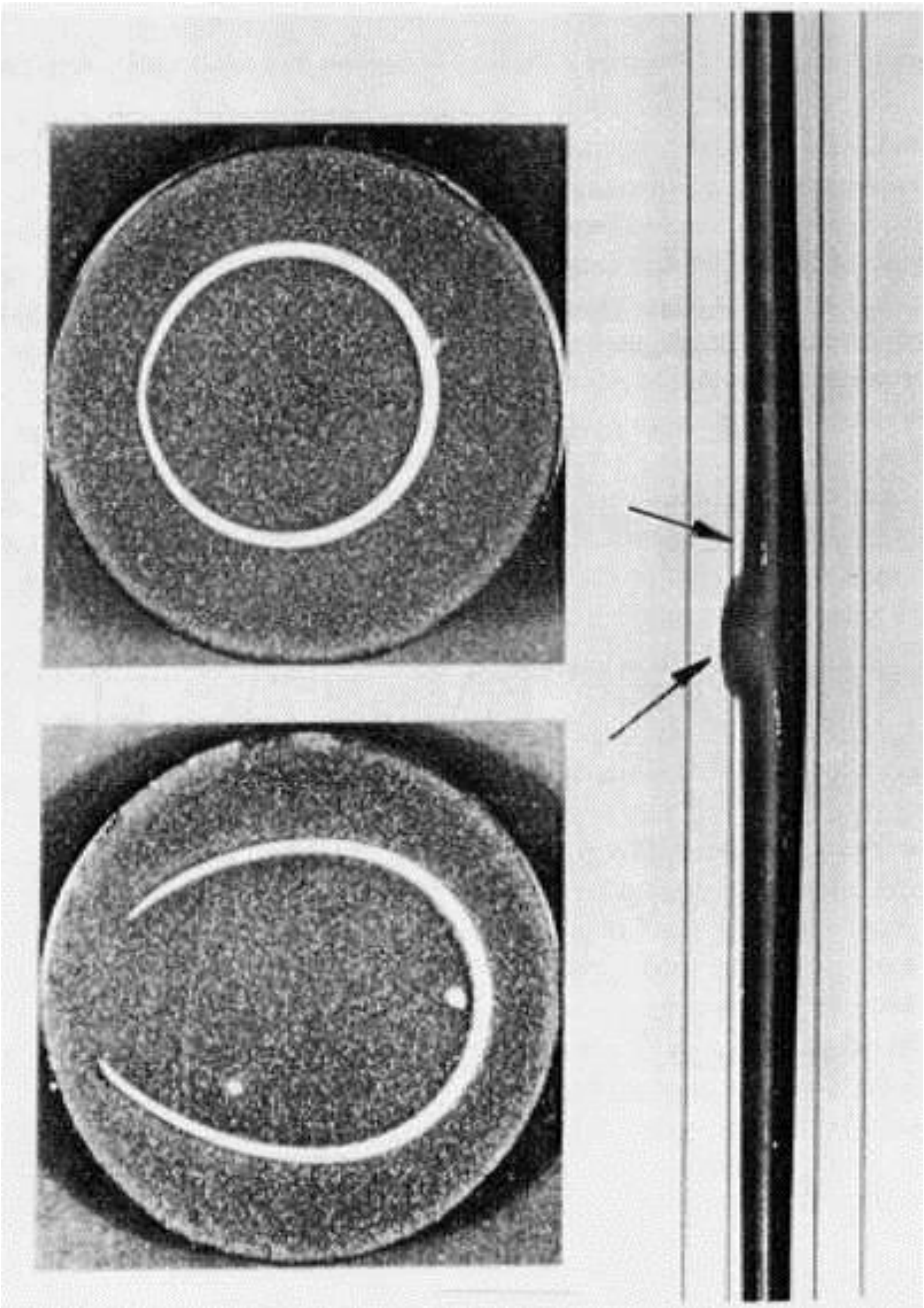


Figure 8
Views of typical post-test zircalloy tube from REBEKA.

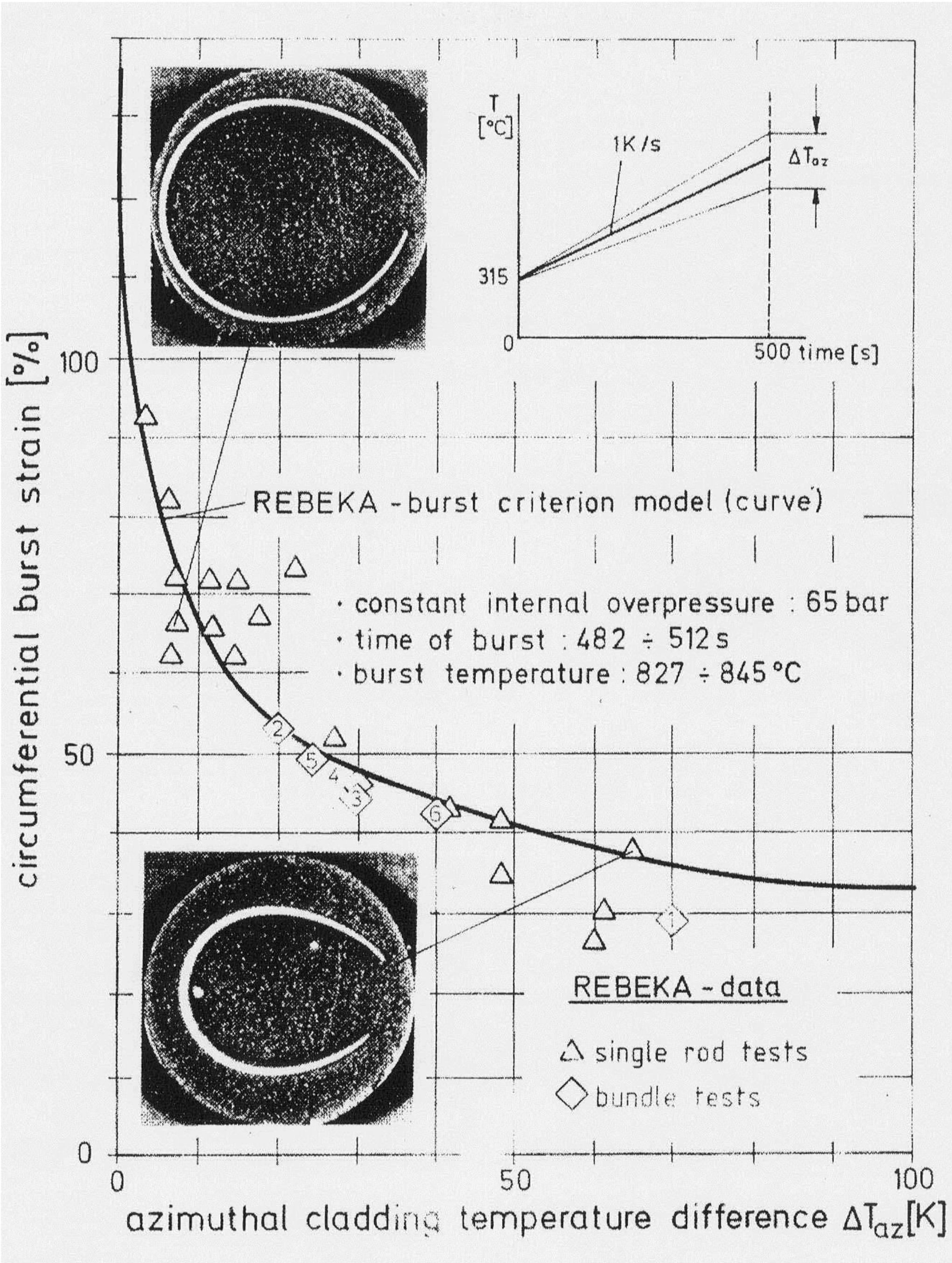


Figure 9
A typical observed variation of circumferentially averaged burst strain with azimuthal temperature difference, from the REBEKA tests.

In Figure 10 is reproduced Figure 5 of Grandjean Part I[5], showing the observed burst strain as a function of temperature, for various heating rates (actually for a single rod case). The complexity of the behaviour is very apparent. Ductility increases with temperature in the alpha phase, and then drops sharply as the beta phase begins to appear, before rising once more as the beta begins to dominate. Overlaid on this is a very marked dependence on time / heating rate, with lower rates associated with greater ductility.

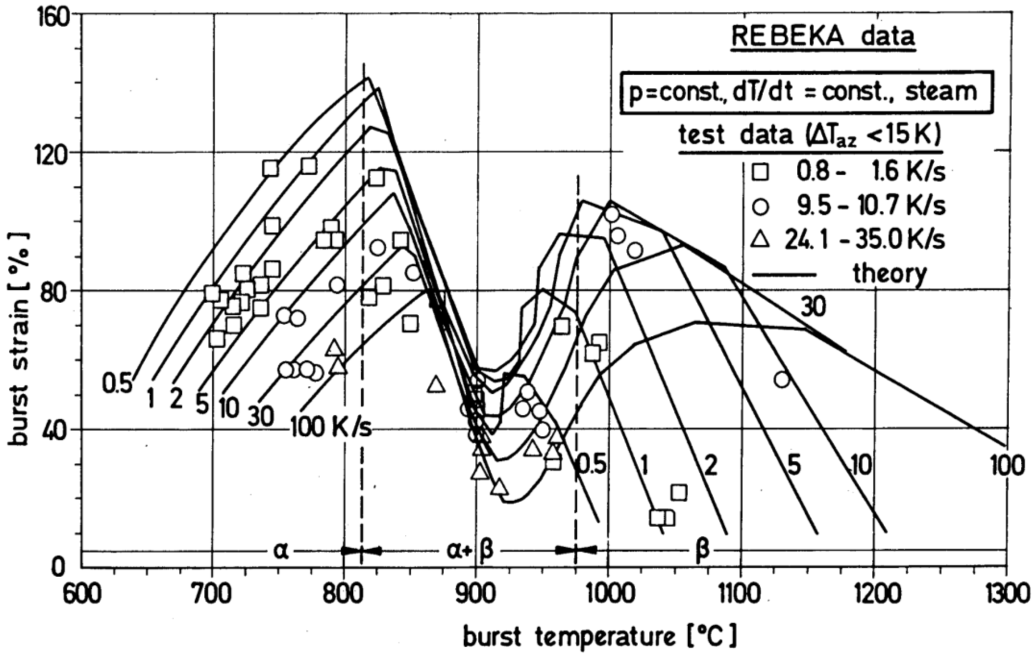


Figure 10
REBEKA-1 Circumferential strain of the nine inner rods and coolant channel blockage

6.1.2 Multi-Rod Burst Tests (ORNL)

The US NRC sponsored a series of tests at Oak Ridge between 1977 and 1981. A total of six tests were run, on rod bundles ranging from 4x4 through to 8 x 8. Bundle-heating tests were conducted cooled by steam with a fairly low flow rate. The heating was by internal electric heating elements within each rod, with an external shroud that could also be heated, as part of the attempts to simulate the effect of the rods being part of a larger bundle. Three spacer grids were provided.

A helpful summary of these tests is provided by [5] (Section 3.2.2, page 25), and by [2] (Section 5.3.2.3, page 129).

An indication of the general experimental arrangement is provided in Figure 11. In Figure 12 is shown a close-up of the cross section of the array, with its heated and insulated shroud.

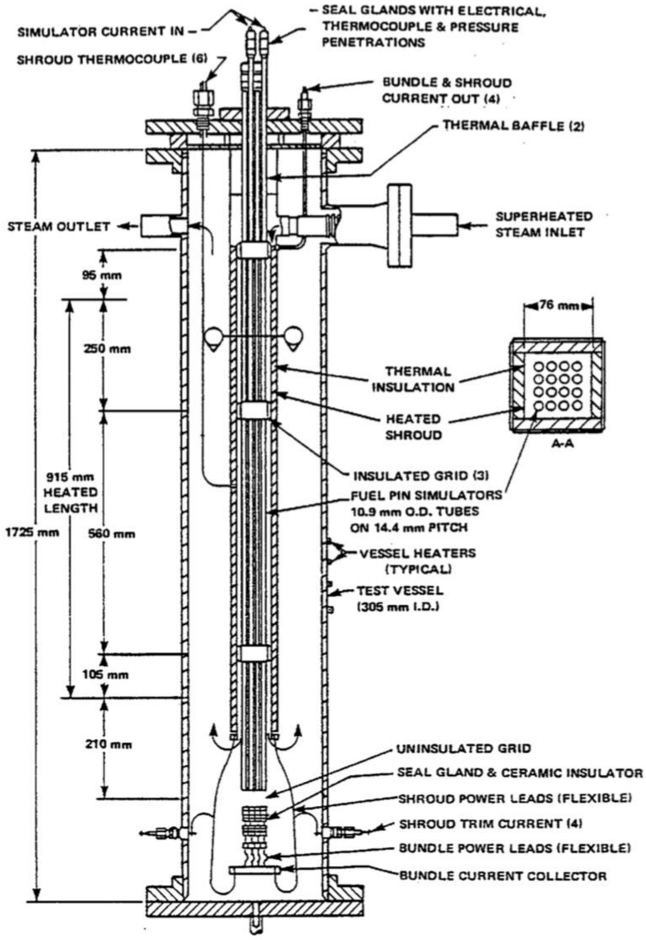


Figure 11
An overview of the experimental arrangements for the MRBT series.

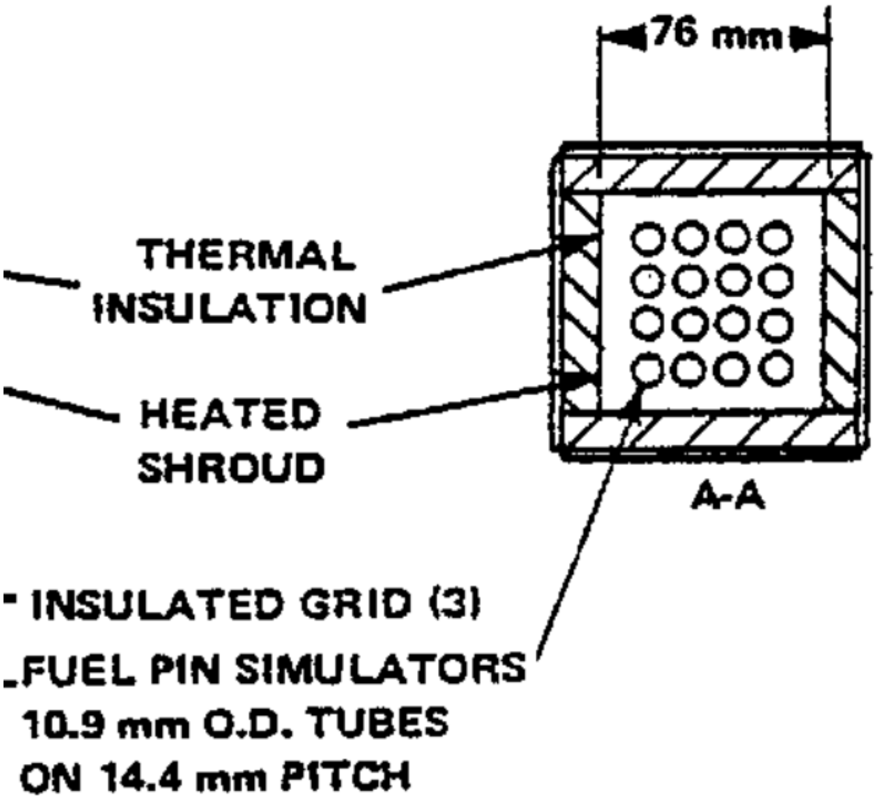


Figure 12
The cross section through the rod bundle, with the heated shroud insulated on its rear face, surrounding the internally heated fuel pin simulators.

Typical bundle post-test cross sections are shown in Figure 13.

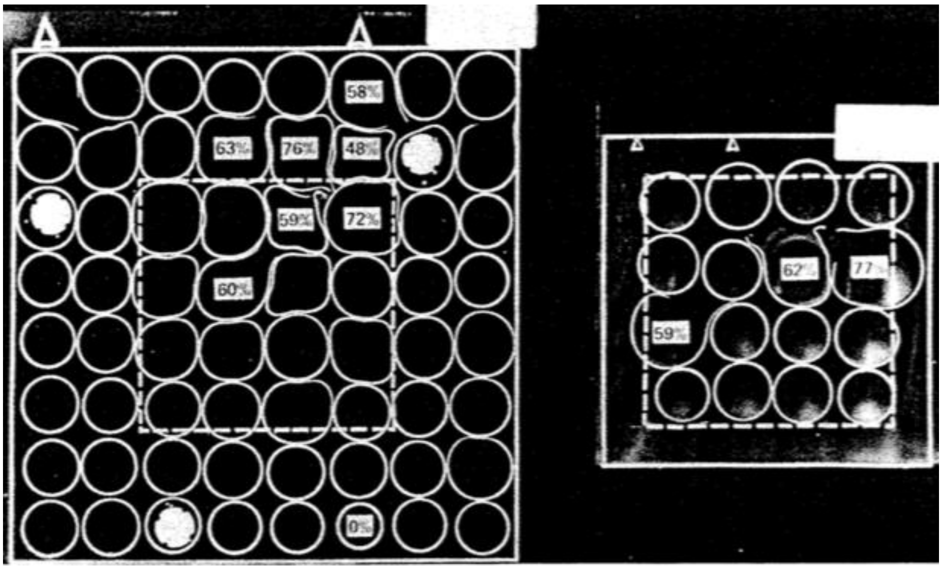


Figure 13
Sections from the tests of B3 (4 x 4) and B5 (8 x 8), showing high degrees of deformation, and some failures

Amongst the objectives of this series of tests was the determination the degree to which bundle size, and the (inevitable) use of relatively small bundles in experiments, influenced the ballooning behaviour. In part this was achieved by the use of a shroud surrounding heated rods, where the shroud itself could be heated.

The general conclusion was that significant mechanical interactions took place between rods, and that these interactions themselves had a big influence on the development of rod distortions. In essence, once a rod was prevented from expanding further radially at a particular axial location, the axial extent of its ballooned region tended to increase. The view was expressed that few-rod tests could give misleadingly low indications of the ballooning and blockage.

Tests from this series at Oak Ridge were amongst those used to provide the data from which the correlations in the NUREG 0630 document were derived. However, only the early tests B1 to B3 were used for this purpose. Arguably this rather weakens the correlation within the NUREG document. We will discuss NUREG 0630 more generally later.

6.1.3 JAERI

A series of test were conducted by JAERI in the late 1970s and early 1980s[11-13]. They were to investigate the behaviour of the typical 15 x 15 Japanese PWR subassembly, and used 7 x 7 simulant rod bundles. The heated length was 0.9 m, with mechanical restraint by two spacer grids. The simulators were internally pressurised to between 20 and 70 bars. Much as in the ORNL program, an external shroud that could be heated or left unheated was employed. Similarly, heating was conducted in an atmosphere of steam, but with a relatively low flow rate and cooling, conditions that are conducive to large deformations. A particular feature was the use in some tests of some unheated rods, broadly simulating the effects of fuel rods adjacent to unfuelled control rod guide tubes. A typical deformation pattern is shown in Figure 14.

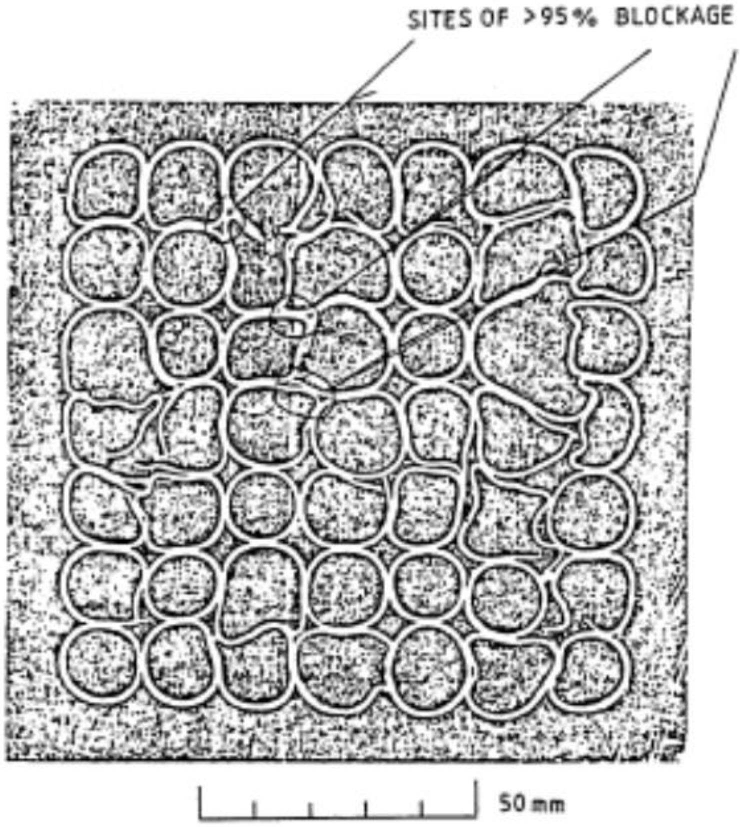


Figure 14
Deformation pattern at axial location of maximum blockage (JAERI test 24)

In many respects the JAERI are not unlike the MRBT's, with broadly similar conditions, and broadly similar observations. A notable extension offered by the JAERI tests was the investigation of the effect of unheated rods within the bundle, representing the effect of control rod guide tubes.

It was notable that the presence of an unheated tube, causing significant azimuthal temperature variations, did not result in reduced overall cladding strain or blockage.

All in all, the tests reinforced the conclusion that significant interaction between rods does take place, and that propensity to block cannot safely be predicted from single tests.

6.2 Dynamic ballooning, in-pile nuclear heated

6.2.1 PBF-LOC Tests

Although single-pin in the substantive sense, in that the pins involved had no interaction with each other, in these tests[30, 31] four pins were simultaneously subject to fission heating and reduced cooling, simulating conditions expected to obtain in a loss of coolant accident. Pins were arranged to differ from each other in various respects, such as burnup and pressurization. A cross section of the arrangement is shown in Figure 15. The shroud around each pin simulates a sub-channel flow passage. Tests wee conducted under conditions designed to achieve temperature in the various zircalloy phases (α, ($\alpha+\beta$), β).

Quite dramatic images were obtained, of cladding and fuel pellets burst and fragmentation; see Figure 16.

Irradiated fuel experienced very much greater strain before failure. In LOC3 this was 42% compared to 20%, and in LOC6 74% compared to 31%. Similarly, axial extents of ballooning were much greater for irradiated fuel: see Figure 17 and Figure 18.

This was attributed to the fact that during the irradiation azimuthal variations tended to be eliminated, by effects such as gap closure due to creep-down and fuel swelling, and the resulting azimuthally more uniform system was then less prone to strain localisation and early failure.

An assessment of the results for the behaviour of the cladding, and a comparison with those from an ORNL set of tests, is provided in [5].

The important issue of fuel behaviour, and in particular fuel relocation upon fragmentation, was investigated in the PBF-LOC tests. The relocation into the ballooned region is plain in Figure 19. Confirmation of this is seen in Figure 20.

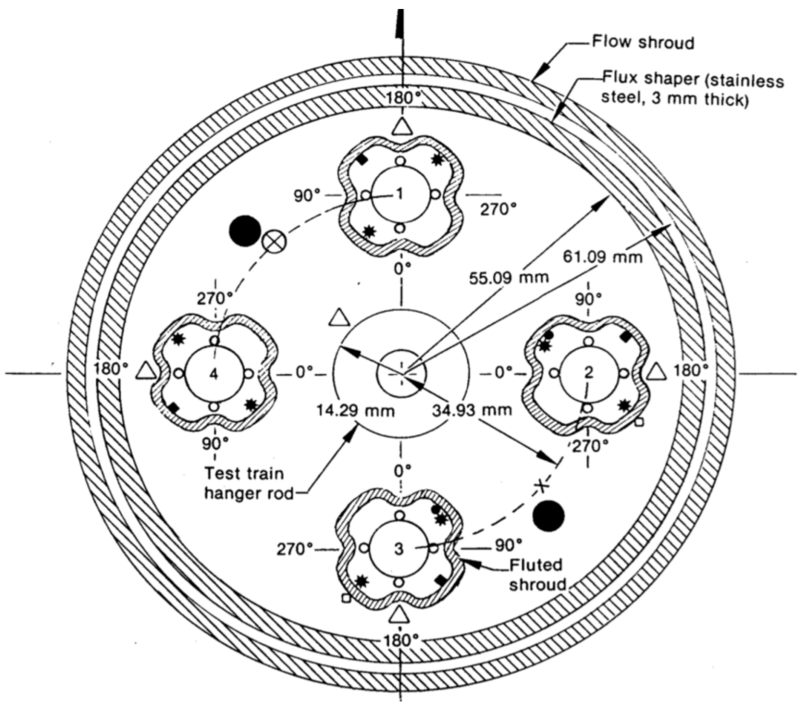
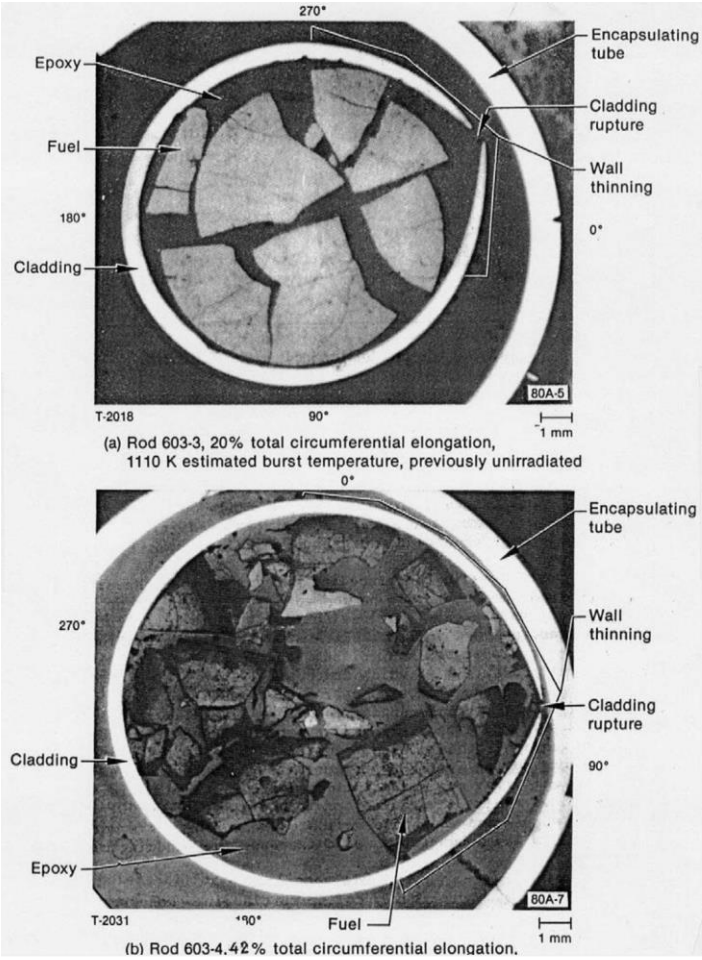


Figure 15
PBF LOC test arrangements



(a) Rod 603-3, 20% total circumferential elongation, 1110 K estimated burst temperature, previously unirradiated

(b) Rod 603-4, 42% total circumferential elongation,

Figure 16
Pellet and cladding failures for fresh (upper) and irradiated fuel

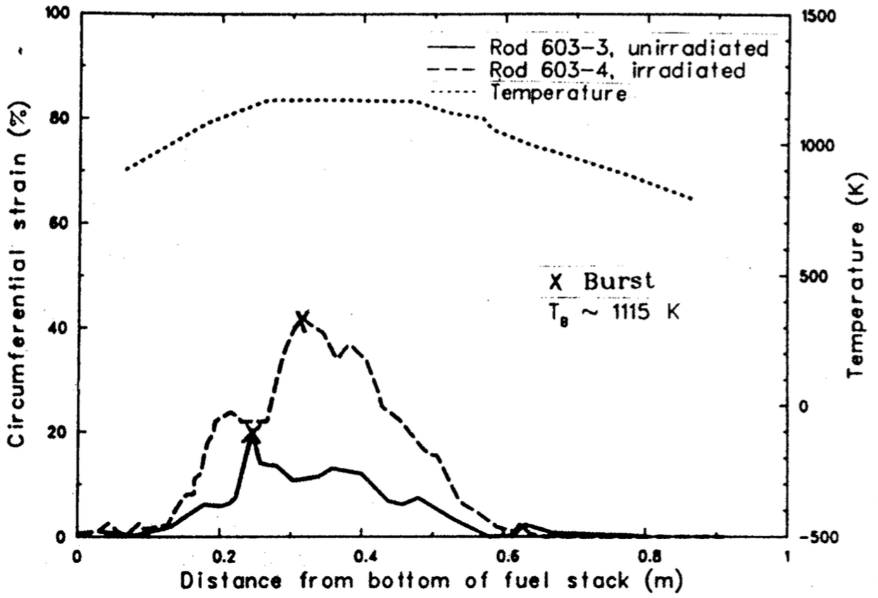


Figure 17
LOC3: Irradiated versus fresh fuel response

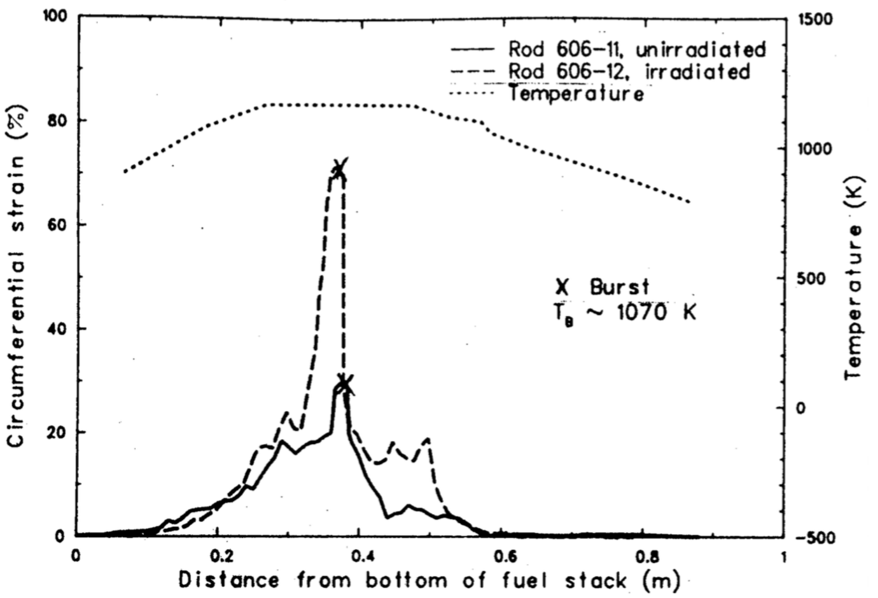


Figure 18
LOC6: Irradiated versus fresh fuel response

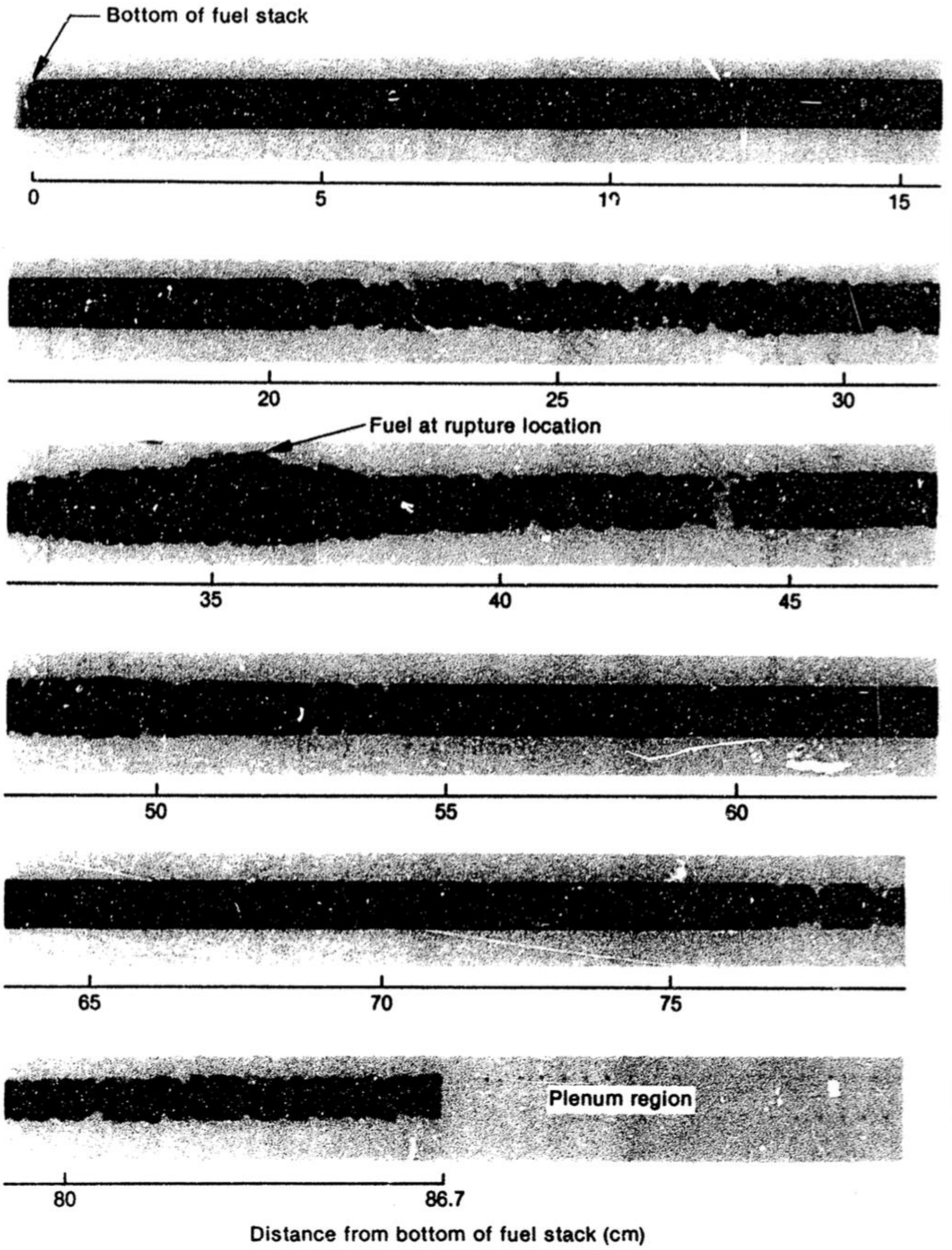


Figure 19
Neutron radiograph of Rod 12, LOC-6, showing the fuel fragment relocation into the balloon.

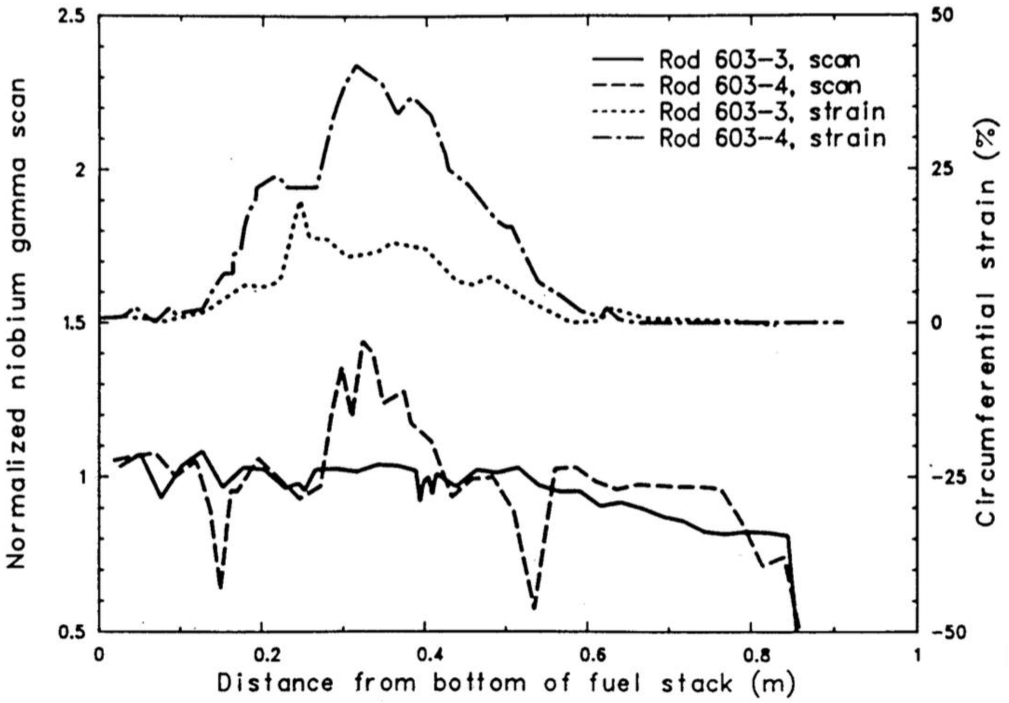


Figure 20
axial variations of gamma emission and circumferential strain for two rods, showing the association of large strain with fuel relocation.

6.2.2 KfK Karlsruhe FR2 tests

As a complement to the Rebecca out-of-pile single rods tests, a series of in-pile tests were performed in the FR2 reactor at KfK Karlsruhe[45]. The intention was to try to identify the influence of both the reactor environment, and the use of both fresh and irradiated fuel.

The test matrix, reproduced from [5], is shown here as Figure 21, to indicate the approach taken.

Test Type	Test Series	Number of Irradiated Rods	Number of Tests	Target Burnup (GWd/tU)	Range of Internal Pressure at Steady State Temperature (bar)
Calibration, Scoping	A	-	5	-	25 - 100
Unirradiated Rods (main parameter: internal pressure)	B	-	9	0	55 - 90
Irradiated Rods (main parameter: burnup)	C E F G1 G2/G3	6 6 6 6 6	5 5 5 5 5	2.5 8 20 35 35	25 - 110 25 - 120 45 - 85 50 - 90 60 - 125
Electrically-Heated Fuel Rod Simulators (main parameter: internal pressure)	BSS	-	8	-	20 - 110

Figure 21
Test matrix for the FR2 tests (reproduced from [5])

The general conclusion was that as regards to rupture, the FR2 tests exhibited only modest differences between the behaviour of the electric simulator rods, fresh rods, and radiated fuel. Systematic differences between categories lay within the variations within each category. This is apparent in Figure 22. Similarly, there is little clear relationship between burst strain, temperature and rod type; see Figure 23.

There was no real discernible dependence of circumferential strain at first on the azimuthal temperature difference at that elevation. In Figure 24 the circumferential strains observed in the FR2 tests are plotted, along with the Rebeka burst criterion. The two tests do not really provide support to the criterion associated with the Rebeka series.

Significant post-ballooning fuel relocation was observed in these FR2 tests.

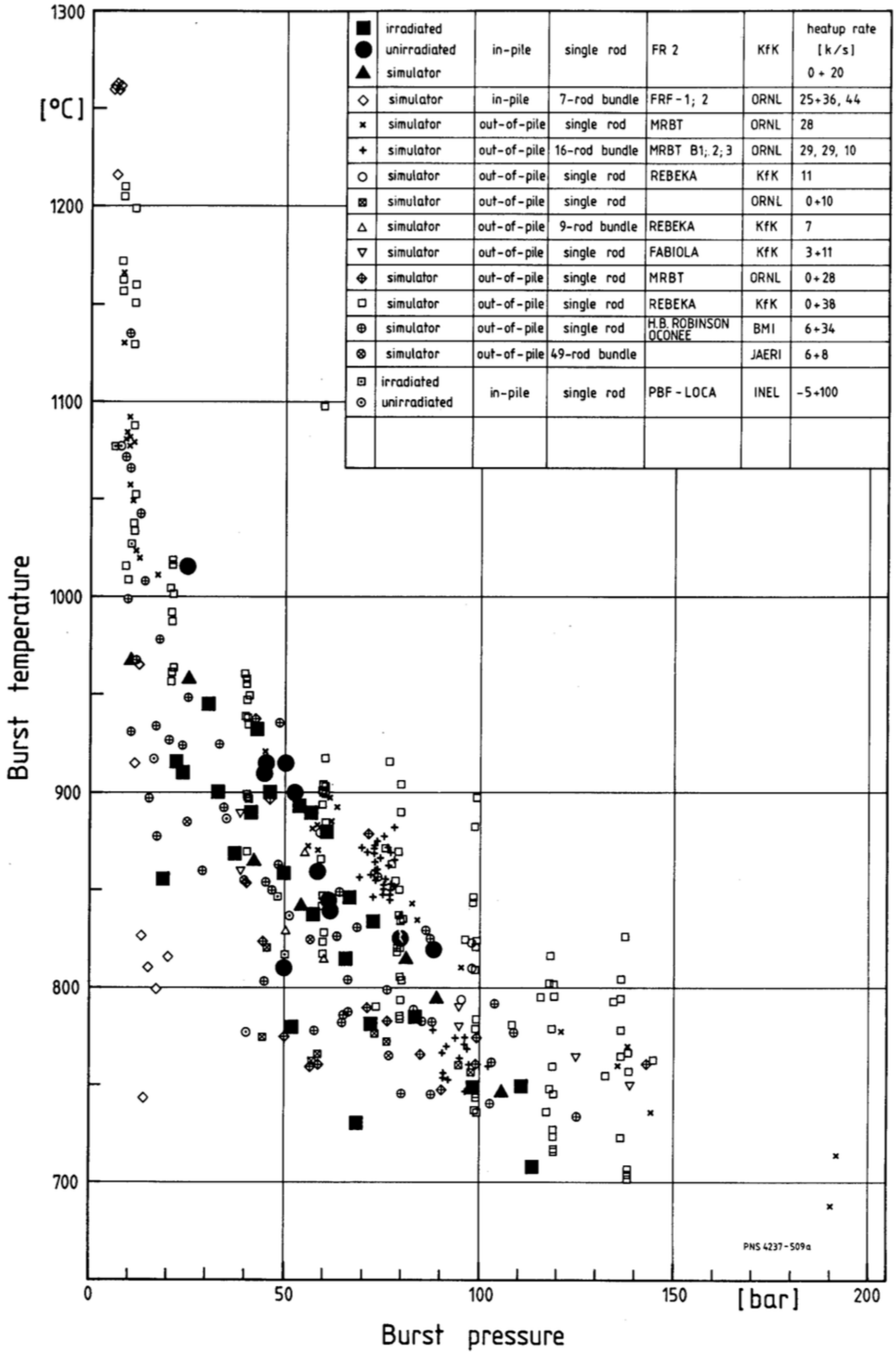


Figure 22
The relationship between burst pressure and temperature over the range of pins investigated in the FR2 tests

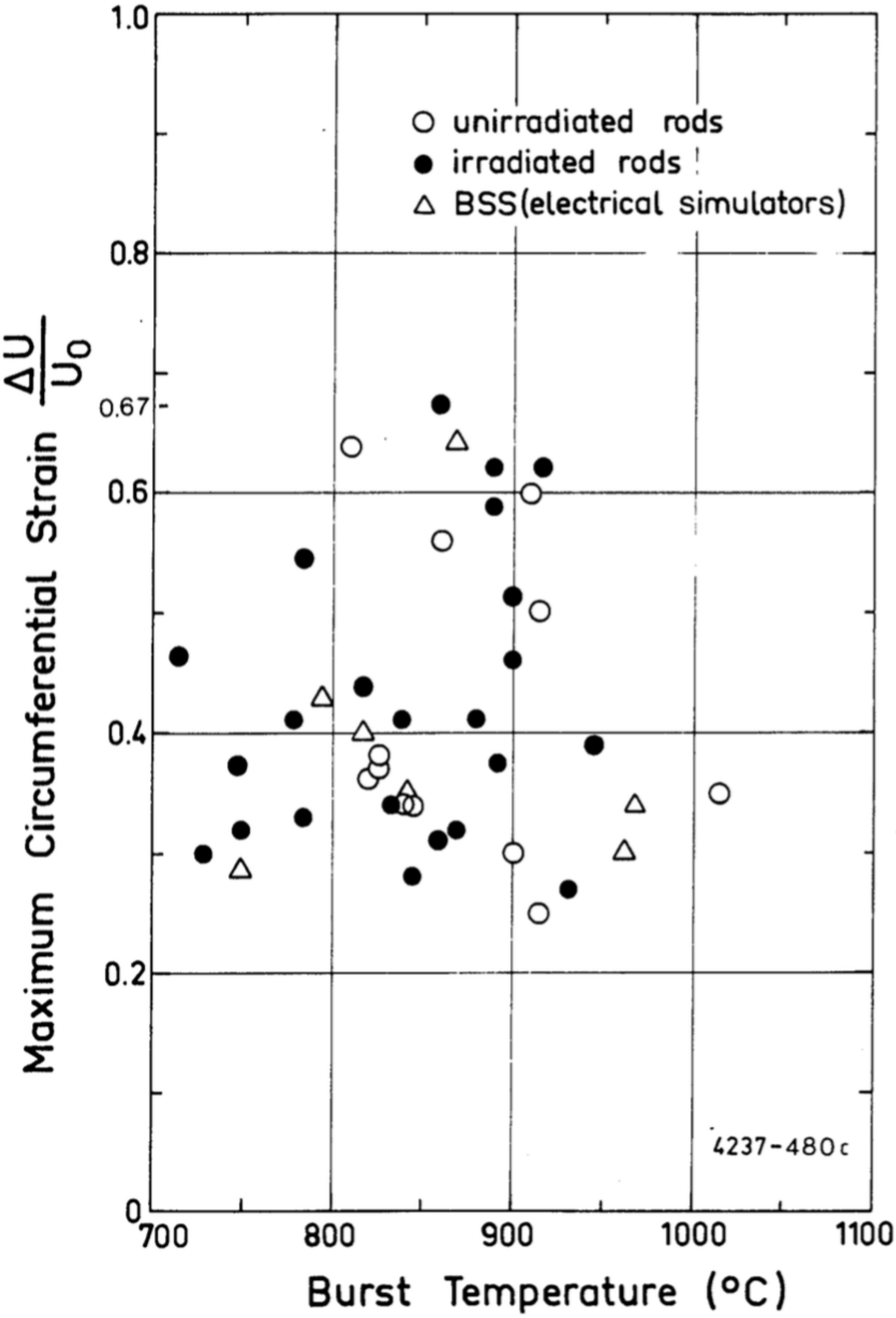


Figure 23
The relationship between circumferential strain and temperature over the range of pins investigated in the FR2 tests

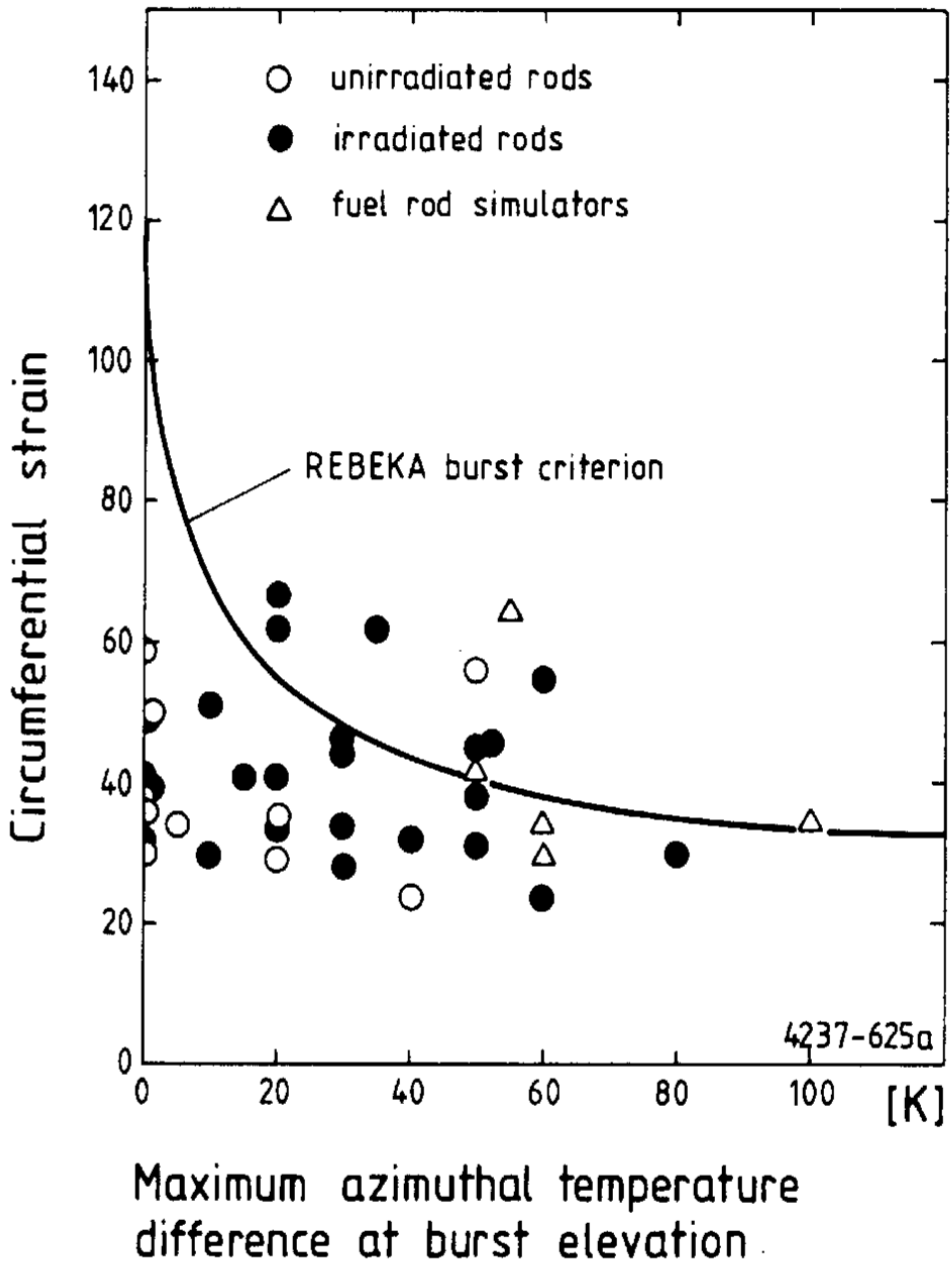


Figure 24
The observed FR2 dependence of circumferential strain at burst on the azimuthal variation of temperature at that axial location. Also shown is the REBEKA burst criterion

6.3 Fixed geometry: Electrically heated

6.3.1 ACHILLES

The ACHILLES programme was undertaken at the behest of the United Kingdom regulator following the Sizewell B pre-construction report submission. In part it was a successor to the THETIS programme[6, 34], designed to address in particular more relevant rod diameters, and the effects of spacer grids. The program was undertaken at the Atomic Energy Establishment Winfrith (AEEW), as then was. The program comprised heat transfer investigations on an assembly of 69 electrically heated full-length simulated fuel rods, in both partially blocked and "normal" configurations. The facility is described by Denham[35]. The cross section of the test section is shown in Figure 25, with the 16 optional sleeves, used to simulate a ballooned geometry, shown. These sleeves simulated a blockage ratio of 80% over an axial extent of 100 mm, with upstream and downstream tapers over 200 mm and 50 mm respectively. The fuel rods were supported by a total of eight spacer grids.

Details of the test, and the results, are reported in [46].

A helpful précis and discussion of the main results is provided in[6], and we will not re-summarise.

This is arguably one of the better designed and characterised experimental facilities, but the complexity of the processes at work is very apparent and rather daunting. Explanations for differences between tests, whilst sounding plausible, are necessarily to degree almost hand-waving assertions. Taking one example from Grandjean's summary:-

"*In the experiments with a more realistic decreasing flow rate, cooling was much better than in the experiments with a constant flow rate and the blockage rewet earlier. This was explained by the fact that the amount of water that accumulated above the blockage during the initial surge was greater than in the constant flow rate experiments. Furthermore, as this levitated liquid mass could not be supported after the surge end, it fell into the top of blockage causing it to rewet. No blockage penalty was observed for the combinations of rod power/ rod initial temperature that were selected for these experiments.*"

This sounds plausible, and may well be correct, but just how secure is even the qualitative, let alone quantitative evidence of this? And if it was, how generic and predictable and reliable is this helpful effect, or was it a peculiarity of the particular conditions in that test? Since our ability to predict, let alone our ability to engineer, such finely varying reflood flow rates does not exist, what comfort can we gain from this anyway?

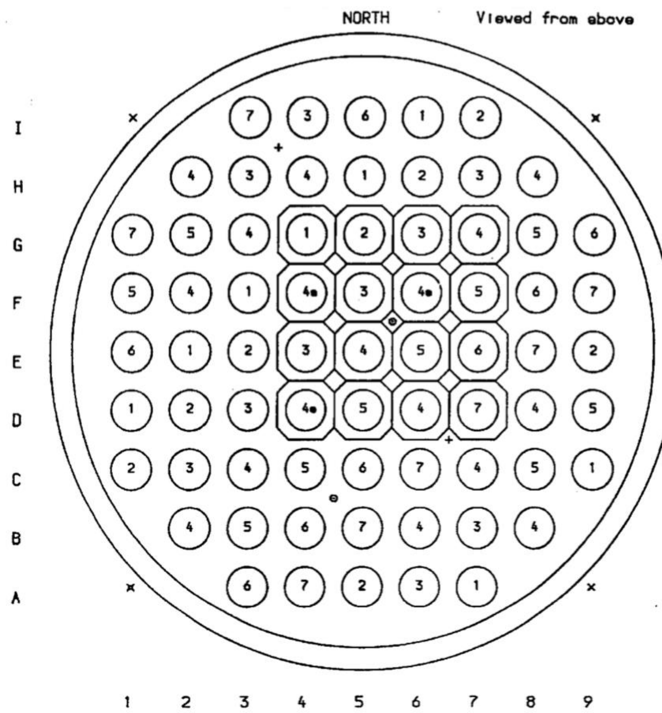


Figure 25
Cross section of the ACHILLES test section

6.3.2 PERFROI

The PERFROI project[39] is run by IRSN, supported by EdF, and involves collaboration with Areva. It is aiming (at this stage) at addressing flow blockages within bundles, and the effect of this on coolability. Within that, there is one strand addressing the mechanical behaviour, deformation and rupture of cladding, and the second strand studying the thermal hydraulics of a partially blocked region of the core during water injection. It is a large multi-facetted project, with both smaller scale experiments, and a larger scale (~ACHILLES-like) activity 'COAL'[40]. This will involve flow and heat transfer measurements around simulated fixed balloons.

It is not a formal reference, but a helpful summary is provided on the IRSN website, from which the following is a summary:

6.3.2.1 The experiments:

The 'mechanical' strand addresses thermal-mechanical deformation and rupture of fuel rods under LOCA conditions, to provide data and validation for the DRACCAR fuel modelling code, a typical (for present purposes) simulation from which is shown in Figure 26.

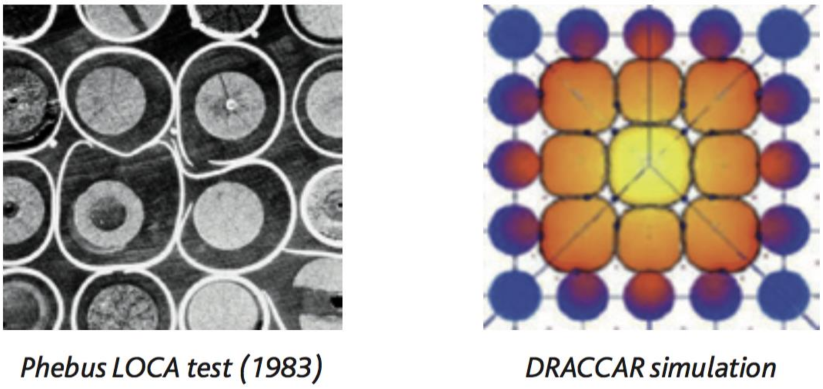


Figure 26
A typical simulation performed by the DRACCAR code

Experiments (termed the ELFE and COCAGNE tests) will be conducted to characterise the thermal mechanical properties of the cladding, and to develop the numerical simulation models needed to simulate cladding deformation and bursts. These models will be integrated into the DRACCAR simulation tool in order to evaluate potential blockage formation in a LOCA transient. In particular, these models will take into account the physico-chemical behavior of hydrogen within the cladding and its potential impact on the mechanical behavior of the fuel rods.

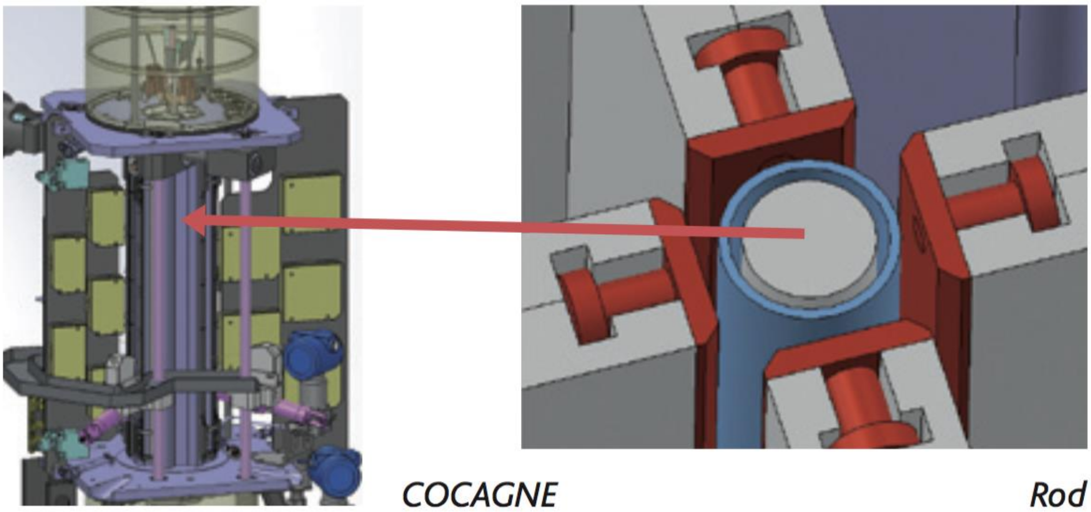


Figure 27
The COCAGNE facility and the planned tests under PERFROI

In COCAGNE small rod sections are heated and pressurised to investigate the deformation in the presence of neighbouring rods by taking into account the contact between rods observed in many experiments in a bundle configuration (such as Phebus LOCA). The COCAGNE experiments are performed with 60 cm long rod sections with internal pressure up to 200 bar and heated to ~1000°C. The test device is equipped with a UV pyrometer for temperature measurements, laser telemetry for deformations, and ultrasonic signals to detect contact 'time and position'.

Thermal hydraulic aspects will be investigated experimentally in collaboration with Areva in Germany, under a programme termed 'COAL'. Experiments devised by IRSN will be performed in the Areva Benson† thermal hydraulics loop facility, where 7 x 7 electrically heated rod bundles will be subject to the thermal hydraulic conditions associated with low and intermediate pressure re-flooding. In particular, the ballooned geometry of the rods, and the degree of blockage formed, will be observed. The issue of the relocation of fragmented fuel has been identified by IRSN as inadequately covered in work thus far, and this features in COAL.

6.3.2.2 The modelling programme of PERFROI

This is centred around the validation of and development of models for the DRACCAR fuel modelling code, combined with the thermal hydraulic modelling capabilities within CESAR, a component of the Astec severe accident code. This will be discussed at greater length later.

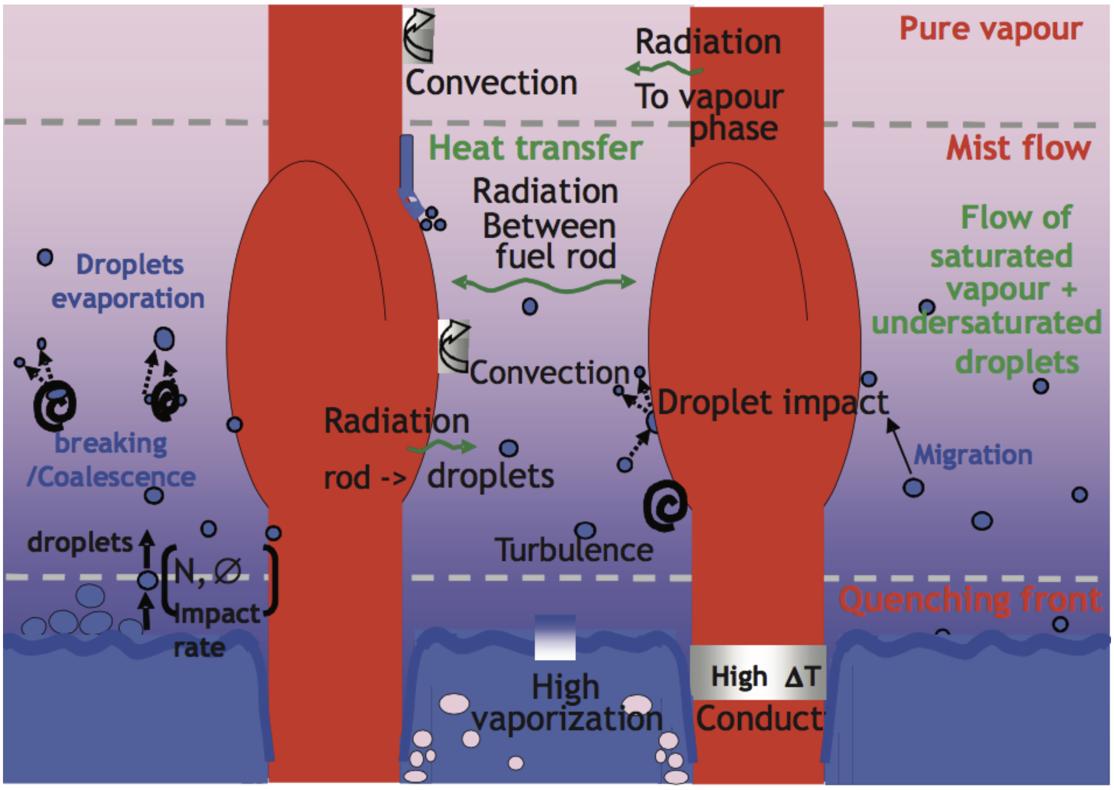


Figure 28
Local phenomena involved in cooling during the reflood process. (Reproduced from [39]

We reproduce from [39] as Figure 28 their identification of the phenomena of interest.

It is notable that a major objective of the PERFROI / COAL activity is explicitly stated as the validation of a fuel modelling code (DRACCAR[47]). It is obviously not a binary issue, but this is something of a shift in emphasis from using experiments as a way to see how one's plant will behave.

† http://www.areva.com/activities/liblocal/docs/BG%20reacteurs-services/BG%20RS%20MONDE%20-%20activit%C3%A9%20-%20Centre%20technique/19%20-%20BENSON.pdf

7 FUEL MODELLING CODES

7.1 Introduction

"Fuel rod modelling codes" have pretty much as long a history as nuclear power itself. This reflects the fact that whilst ostensibly a very simple device, a fuel rod in the form of a metal tube filled with fissile oxide pellets is actually remarkably complex, or perhaps more correctly, is a device of which the behaviour is the result of the interaction of a large set of individually remarkably complex phenomena. Without further discussion, we will simply reproduce here in Figure 29 a very early attempt to depict this complexity.

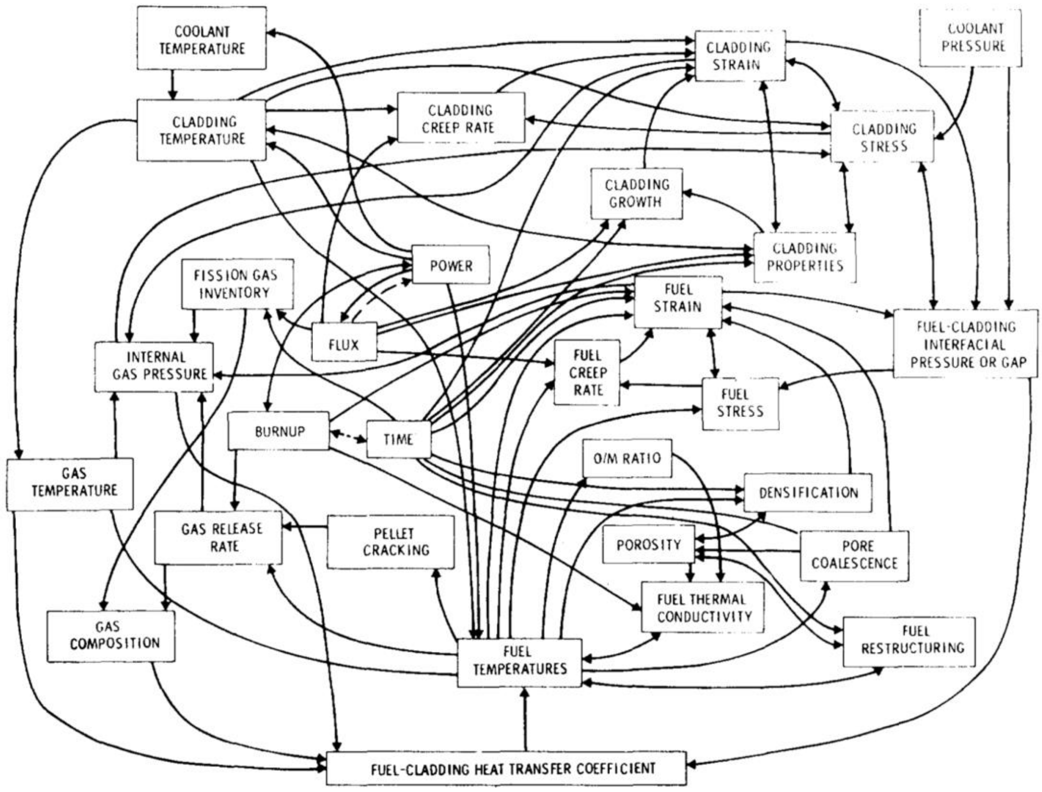


Figure 29
Some of the issues and phenomena involved in nuclear fuel modelling (from Beyer et al, Batelle-Pacific Northwest Laboratories (1975), cited by [48])

We will not attempt at all to provide a review or history of all this here. Rather, we will confine ourselves to commenting on the characteristics of some of the main present day fuel modelling codes under active development, with a particular emphasis on those that are candidates for the modelling of the fuel response during reflood and so on.

As will be clear from the discussion of the physical phenomena above, there is however rather more to modelling than "fuel modelling". At is simplest, modelling of the response of the fuel requires a knowledge of the thermal conditions to which it is subject and if that were the end of the story one will also need to be considering the computational tools available for predicting these thermal hydraulic conditions.

It is rather more complex than that, as well. There is good reason to believe that the ballooning response of the fuel itself changes the thermal hydraulic conditions, so the coupling between these two areas of analysis also comes into play.

In the sections below we will note the main codes used for modelling of the various ballooning experiments discussed above, and then in subsequent sections discuss a very small number of codes that seem candidates for development and application to future studies.

7.2 Previous LOCA / Ballooning modelling studies

There is a large number of fuel pin modelling codes that have addressed the ballooning issue. Principal amongst these are the two Westinghouse codes BART and TAPSWEL, used for analysis of the UK Sizewell plant[49], the KFK code SSYT code[50] and the UKAEA-developed MABEL code[50] and CANSWELL[51]. The FRETA-B code[52] is notable for present purposes in that it is an extension of FRETA to analyse multiple pins.

A review of the ISP-14 (REBEKA-6 test) modelling is provided in [5]. As the simplest way to indicate the range of codes used for this we reproduce from this document its Table 10, as Figure 30. Similarly, a review of ISP 19 (PHEBUS 218 Test) is provided, and the participating codes are given here in Figure 31.

Participant	Identifier	Analyzed Rods	Model	Calculation Particularities
CEA	K	49, 29	CUPIDON 4.0	Heat transfer coefficient calculated with Dittus-Boelter during steam cooling
AEEW	L	"average rod"	MABEL-2D	Fluid dynamics calculated by TRAC-PD2
EG&G	M	49, 20, 29	FRAP-T6	Measured cladding temperatures as input
VTT	N	49	FRAP-T6	Fictive fuel material data
ÖFZ	P	49	BALO-2A	Separate empirical gas pressure model

Figure 30
Participants and codes in ISP 14 (REBEKA-6 Test) (Table 10 of [5])

Institution	Country	Corresponding Experts	Computer program	Identifier
Gesellschaft für Reaktorsicherheit (GRS), Köln	Germany	I.A. Keusenhoff	TESPA	A
Österreichisches Forschungszentrum (ÖFZ), Seibersdorf	Austria	G. Sdouz G. Sonnek	BALO-2A	B
Swedish Nuclear Power Inspectorate	Sweden	J. Mattson O. Sandervag	TOODEE-2 (Swedish version)	C
UKAEA, Springfields	United Kingdom	T.J. Haste	MABEL-2D	D
National Nuclear Corporation (NNC), Whestone	United Kingdom	R.E. Haigh, M.E. Shanawany	SWEM / BART	E
Swiss Federal Institute for Reactor Research, Wurenlingen	Switzerland	H. Gautier	FRAP-T6	F
Energy Research Foundation, ECN, Petten	Netherlands	Th. Van der Kaa	FRAP-T6	G
CEA / DMECN, Saclay	France	M. Fillatre	CATHACOMB	H
CEA / IPSN, Cadarache	France	I. Drosik	FRETA	I
VTT Technical Research Center, Helsinki	Finland	S. Kelppe	FRAP-T6	J

Figure 31
Participants and codes in ISP 19 (PHEBUS 218 Test) (Table 13 of [5])

None of these seem likely to form a sound basis for further studies and development. French efforts are now focused on DRACCAR, and the US has strong contender in BISON. We will discuss the Mabel code a little below, but not in the context of its being itself a candidate for future development.

7.3 TRANSURANUS

The TRANSURANUS code was developed at the European Institute for Transuranium Elements (TUI). It is a general-purpose code for the modelling of fuel rod behaviour under various conditions, both normal, of normal and faults. It is able to deal with both long-time ("burn up"), and very short-term, millisecond transients. The overall approach is broadly similar to that taken in DRACCAR, and which is discussed at more length below. Rather than a finite element type to discretised solution of the governing equations, perhaps can be similarly likened to a numerical solution of semi-analytic approximate treatments. The code is quite old. The first substantive publication describing it is from 1992, but obviously development predates this. The code was made available for general use in 1992, since then has been widely applied to the analysis of age large number of different types of fuel and different circumstances.

A small sampling of quite recent papers employing the code is [53-69]. One noticeable feature is the recent work there has been on coupling to DYNA 3D[63, 66, 69]. TRANSURANUS seems not to have been used for LOCA ballooning analyses.

7.4 MATARE

7.4.1 MATARE modelling objectives

In essence, these were;

If pins balloon during reflood, they reduce the coolant flow passage area around them, which will influence the mass flow, temperature, and heat transfer coefficient associated with the fluid flowing past them in neighbouring subchannels. MATARE takes this into account, reducing the relevant subchannel area in the thermal hydraulic model as a pin balloons.

If all pins are identical, and are subject to the same conditions as each other, the above coupling could perfectly well be modelled by considering only a single representative pin. However, there are numerous systematic and stochastic differences between pins. Stochastic ones include effects such as manufacturing tolerances, wall thickness variations from the drawing process, pellet dimensional tolerances and defects, pellet eccentricity within the cladding and so on. More systematic differences arise from cross core (radial) variations in power, on a core-wide length scale, and differing proximity to unheated control rod guide tubes, and to subassembly to subassembly gaps, both at subassembly edges and indeed at subassembly corners. By modelling multiple pins via individual instances of a pin modelling code these differences can be taken into account.

The MATARE model includes some important changes to RELAP to represent better the dry-surface heat transfer. This includes modifying the single-phase convection correlations and introducing an empirical model of the turbulence induced by spacer grids, together with a mechanistic model of spacer-grid quench and wet-grid evaporation effects.

The droplet size is tuned to provide a good representation of high-void fraction dispersed droplet flows through the rod-bundle geometry, but is not able to represent quench-front progression late in reflood.

The model also assumes that momentum cross-product effect can be neglected, which is reasonable for predominantly axial flows.

Overall, these models make MARATE suitable for predicting the development of the ballooning, but not the coolability of severe blockages.

7.4.2 MATARE overall structure

'MATARE', **MA**bel-**TA**link-**RE**lap', is not a code in its own right, but refers to the coupling of Mable and Relap, using Talink, to form a composite tool to study ballooning. It was developed at Imperial College in the early 2000's, supported by the (then) NII and British Energy. It is reported fairly fully in [20], and in the journal paper [70]. Its use to analyse the MT-3 experiment is reported in [21], and to study a hypothetical PWR reflood in [71]. Its application to cases with an advanced 1% Nb alloy is presented by Jones[72].

In essence, the MATARE model considers multiple pins. Each pin is analysed by a single instance of the fuel pin modelling code Mabel. All of the subchannels representing the space around the pins considered are modelled by RELAP. Cross flow between the subchannels is permitted within the RELAP model, via conventional crossflow junctions. The general computational structure is shown in Figure 32.

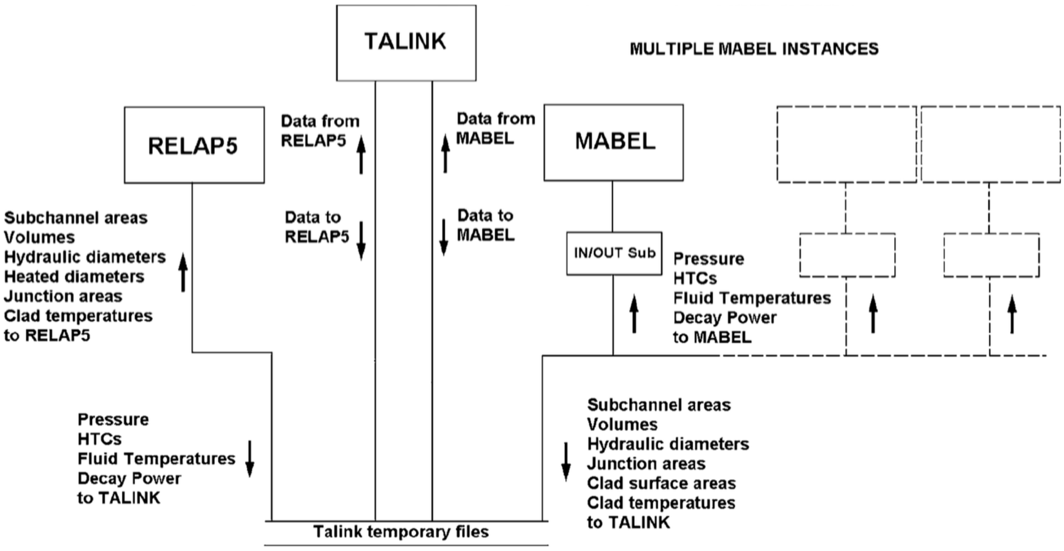


Figure 32
The computational structure of MATARE

7.4.2.1 Relap

The RELAP code was developed for best- estimate transient simulation of light water reactor coolant systems during postulated accidents. The code was developed at Idaho National Engineering Laboratory (INEL) for the Nuclear Regulatory Commission (NRC).

The code is based on a non-homogeneous and non-equilibrium model for the two-phase system that is solved by a fast partially implicit numerical scheme. The RELAP analysis is not a full 'CFD' treatment. Rather, the code is one-dimensional and solves six basic field equations for six independent variables: pressure, specific internal energies for liquid and vapour, void fraction and liquid and vapour velocities. The constitutive relations include models for defining flow-regimes and flow-regime-related models for inter-phase drag and shear, the coefficient of virtual mass, wall friction, wall heat transfer, and inter-phase heat and mass transfer.

A boiling curve is used in RELAP to govern the selection of the wall to fluid heat transfer correlations. The code boiling curve logic is based on the value of the heat slab surface temperature. These correlations are based on fully developed steady-state flow, while entrance effects are considered only for the calculation of the critical heat flux.

In the quench front model, the code uses a fine mesh model that inserts additional nodes into the heated wall if significant axial temperature gradients exist along the wall. This allows a more accurate representation of the true localised energy release from a localised portion of a heater or nuclear rod rather than the energy release from all the structures within the fluid node. Fine mesh heat transfer cells for axial and radial conduction are superimposed on the coarser hydraulic computational cells usually used for the heat transfer analysis.

No specific models are used to simulate the spacer grids in the hot rod bundle. In particular, there are no models to include the effect of heat transfer enhancement induced by the grid. Entrainment and de-entrainment is only calculated for annular-mist or horizontal stratified flows.

7.4.2.2 Mabel

The MABEL code[73], developed by the UKAEA, carries out a radial and azimuthal (R–θ) solution of the temperature and strain field, in a single pin, at all axial nodes. The behaviour of surrounding rods is taken into account to some degree by the use of input parameters that determine the extent to which the behaviour of the other rods reflects that of the rod analysed. It considers heat transfer between the four sub channels that border each pin, and between the eight other pins that surround it.

The MABEL thermal–hydraulic routine is capable of modelling the flow in four sub-channels surrounding the pin of interest. Some account is taken of flow diversion due to ballooning or bowing. The MABEL model, however, assumes a homogeneous, quasi-steady two-phase flow, so it is normally used to perturb input thermo-hydraulic conditions obtained from best-estimate codes such as RELAP5 or TRAC.

The pellet stack is assumed to remain intact and the eccentricity of the pellets relative to the cladding, both direction and magnitude, is prescribed by the user.

7.4.2.3 TALINK

Both of MABEL and RELAP need to run simultaneously, and to pass information from one to the other as the computation progresses.

The code TALINK (Transient Analysis code LINKage) is a proprietary utility code, designed to control the data transfers required for the execution of a set of coupled transient analysis codes performing their calculations in separate operating system processes. The data transfers are linked on many-to-one and one-to-many bases with the TALINK code being the central component in this structure and the other codes being regarded as client codes. As required, TALINK will perform simple computations on the data transferred, for example, converting values from one set of units to another.

Figure 32 shows schematically the interactions between the codes.

7.4.3 Sample MATARE results

7.4.3.1 MT-3

MATARE has been used to model the MT-3 experiment[21].

Obviously more details are available in the reference cited, but we show in Figure 33 the observed and predicted cladding cross section. In that the direction of pellet eccentricity within the cladding simply cannot be known, that was used as an input, with the directions inferred from the test provided to the code. Other than that, the simulation was effectively blind. The heterogeneity between rods is well reproduced.

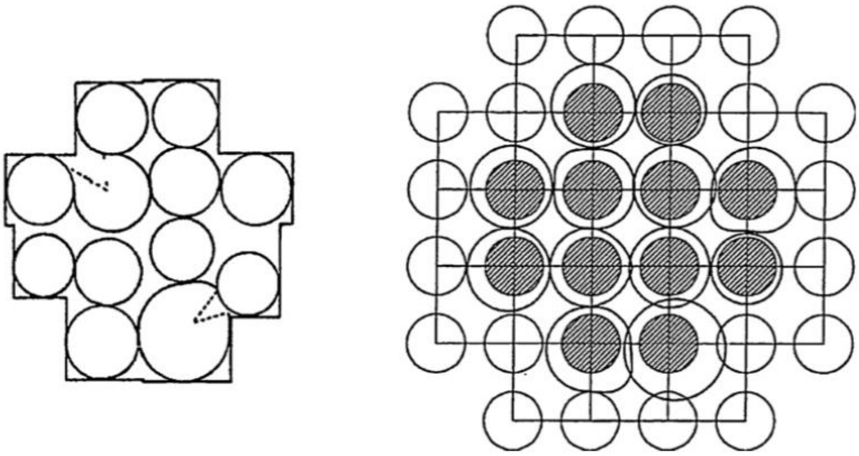


Figure 33
Observed MT-3 clad geometry, and the MATARE prediction

7.4.3.2 Generic PWR

In [71] a generic PWR subassembly reflood was analysed.

The regions studied are indicated in Figure 34.

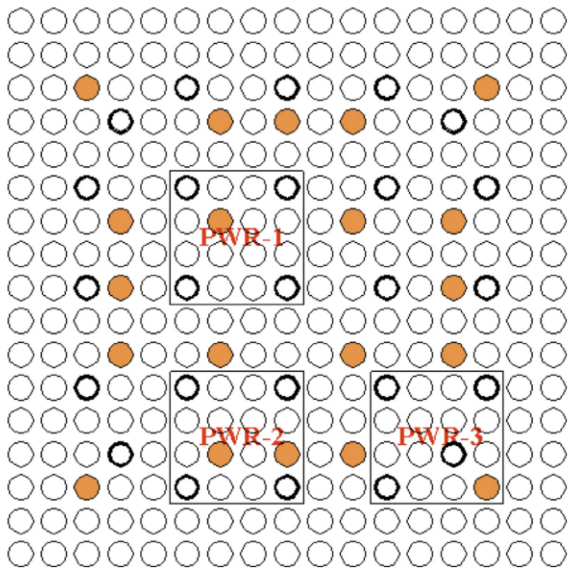


Figure 34
PWR sub-assembly regions analysed (fuel with burnable poisons in orange; guide thimbles in annular black)

Exploiting a degree of assumed symmetry, the overall subassembly ballooning was computed as shown in Figure 35.

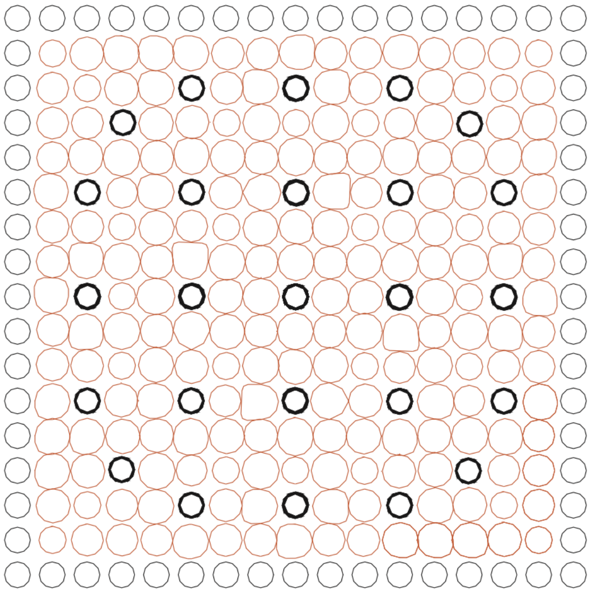


Figure 35
Reconstruction of fuel assembly blockage.

7.4.4 Closing remarks

The work reported here was initiated approximately 15 years ago. The approach adopted, of the coupling of a one-dimensional, hydraulic code with a classical fuel pin structural mechanics code, to allow the interactions between multiple fuel rods to be treated, was novel at the time.

This approach has now been picked up, and is being followed in the context of the DRACCAR and CESAR coupling, being developed by IRSN.

7.5 DRACCAR

The DRACCAR[74] code is being developed by IRSN. A recent description of the status of this code, and in particular of its ability to model the mechanical response of fuel rods under ballooning conditions during reflood, is given by [47].

7.5.1.1 Structural mechanics

From a structural mechanics point of view, the DRACCAR code is a relatively conventional nuclear fuel pin modelling code, albeit modern and pretty much state-of-the-art. The structural mechanics model is largely two-dimensional *r*-theta, with some degree of axial interaction through quantities such as internal pressure, and a 3-D creep material model. The treatment is 'semi-analytical':-

"Mainly, the mechanical modeling is based on analytical stress–strain relations devoted to thin-wall cylinder (not necessarily with a circular base) loaded with internal and external pressures and thermics.

-at each axial slice, each cladding of the rod is azimuthally discretized,

-on each node, equilibrium of a thin cylinder is solved with local values (i.e. nodal) of radius of curvature, pressure difference and thickness of the cladding, radial, hoop and axial stresses, solutions of this problem allow to obtain an equivalent stress (von Mises or Hill criteria depending on isotropy of the material)"

This is fairly normal for 'nuclear' fuel codes.

Azimuthally non-uniform deformations, and the ballooning of clad to the point where it interferes with the cladding on neighbouring pins, is treated. This requires the thin shell theory on which the code is based to be applied to the more or less flat geometries bounded by portions with high curvature, associated with touching ballooned rods, and require it to deal with the mixed, essentially non-normal traction, boundary conditions accompanying such contact. Anisotropy cannot be treated.

The code has sophisticated bespoke material models for the deformation of the cladding.

There is mention of the need for treatments of the breakup and relocation of fragments, and it is stated that DRACCAR does this parametrically.

The code is able to model multiple pins simultaneously. Influenced by computational costs, the code has targeted simultaneous evaluation of 1/8 of a subassembly.

7.5.2 Thermal hydraulics

Thermal hydraulic modelling is provided by the CESAR[75] code, the two-phase, thermal hydraulic component of the Astec severe accident analysis code. CESAR is a two-phase five-equation model, including a drift flux model. It does not cover dispersed droplet phase flows, which are regimes that generally obtain during reflood.

The CESAR thermal–hydraulic modeling is based on a 1D, 2-fluid, 5-equation approach. As a result, 6 differential equations and 1 algebraic equation are solved[75]:-

• 3 mass-balance equations, one for the gas mixture, one for the incondensable gas and one for the liquid phase;

• 2 energy-balance equations, one for the gas phase and one for the liquid phase;

• 1 mixture (liquid and gas phases) differential momentum balance equation;

• 1 algebraic equation that models the interfacial drag between the liquid phase and the gas phase.

One-dimensional thermal -hydraulic codes of this type are the workhorses of nuclear reactor thermal analysis. They have a pedigree at least as long as the fuel modelling codes. The most widely used is probably RELAP, originating under the auspices the US NRC, but many other jurisdictions active in reactor development have now produced home grown equivalents. The CESAR code mentioned above is of broadly this same type, but with a feature and model set tuned for severe accident analysis.

7.5.3 Coupling

The coupling is described as computationally 'deeply nested'.

"First, the DRACCAR code computes the wall contribution to the fluid variables (three fluxes on the liquid, gaseous phases …(It is not quite clear, to the present writer, at least, just what is done. Through wall fluxes are surely zero?)

Second, with these source terms (and the inertia effect inserted in its own Jacobian matrix), the thermal–hydraulic code solves the corrected problem to issue fluid variables increments.

Third, the DRACCAR code deduces the wall's temperature.

However, there is no reason to think that the coupling implementation is not correct and thoroughly competent.

7.5.4 Application & validation

DRACCAR has been used[76] to simulate various integral tests, including Phebus-SFD B9+, PERICLES, Phebus-LOCA 215-R, REBEKA-6, CORA 13. However, as the authors point out, in such complicated coupled problems 'validation' against integral tests does not really allow modelling fidelity to be investigated properly.

Some more detailed comparison has been made[77] between the structural part of DRACCAR and a two finite element codes, albeit with the finite element codes caused to use two-dimensional plane strain approach to match DRACCAR. However, the whole range of interaction has not been assessed.

7.5.5 Demonstration computation

As a qualitative demonstration, rather than an attempt at validation, the authors present the use of the coupled codes to model a notional LOCA experiment. Besides the obvious testing of the code itself, the interest was in observing whether or not the local mechanical deformation of the rods influenced the local thermal hydraulics, and to see whether or not the thermal hydraulics and flow regime in turn influence cladding embrittlement, through its effect on the wall temperature.

A 3m long small bundle is considered, with one half of it modelled; see Figure 36. Only rods 1 to 6 are heated, all at the same rate as each other, with a series of cases with different rates studied. The heating rates were adjusted to achieve various maximum (in any one subchannel) flow blockages.

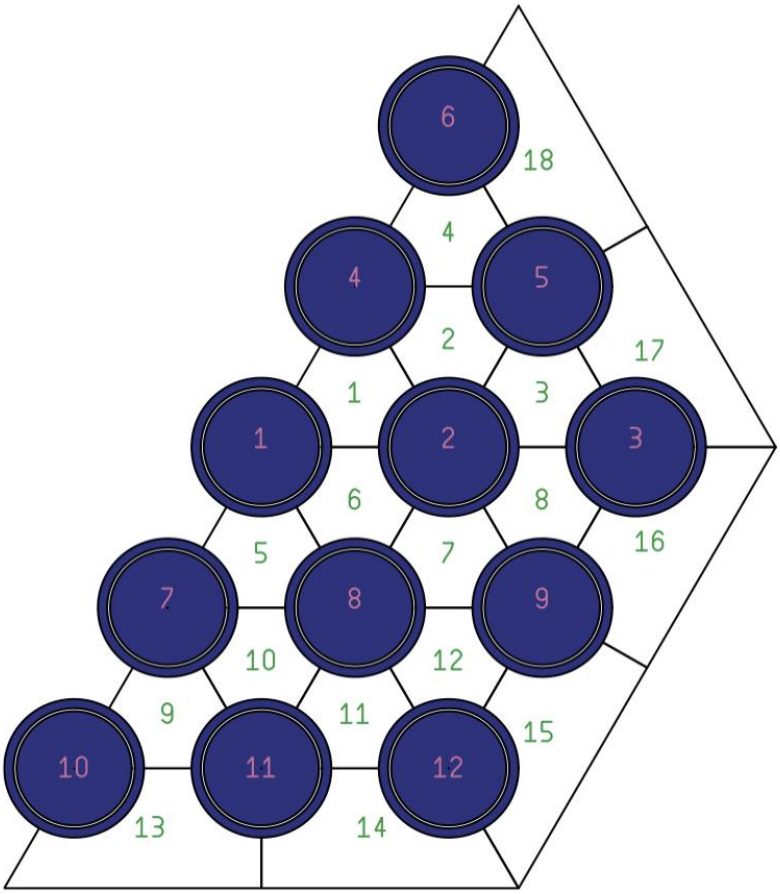


Figure 36
The half-bundle analysed in the DRACCAR demonstration computation[77].

We will not detail the results here, as a full description is available in [77]; we will just show a few sample images; Figure 37.

Again, this is only put forward as a 'demonstration', but it is notable that the predicted blockages did not affect the quench time. In Figure 37 the black arrows indicate cross-flows between subchannels. How such flows are computed by a 1-D code is unclear, but one assumes some form of semi-empirical 'cross flow junction' or equivalent is employed. Note that CESAR is fundamentally a 1-D code.

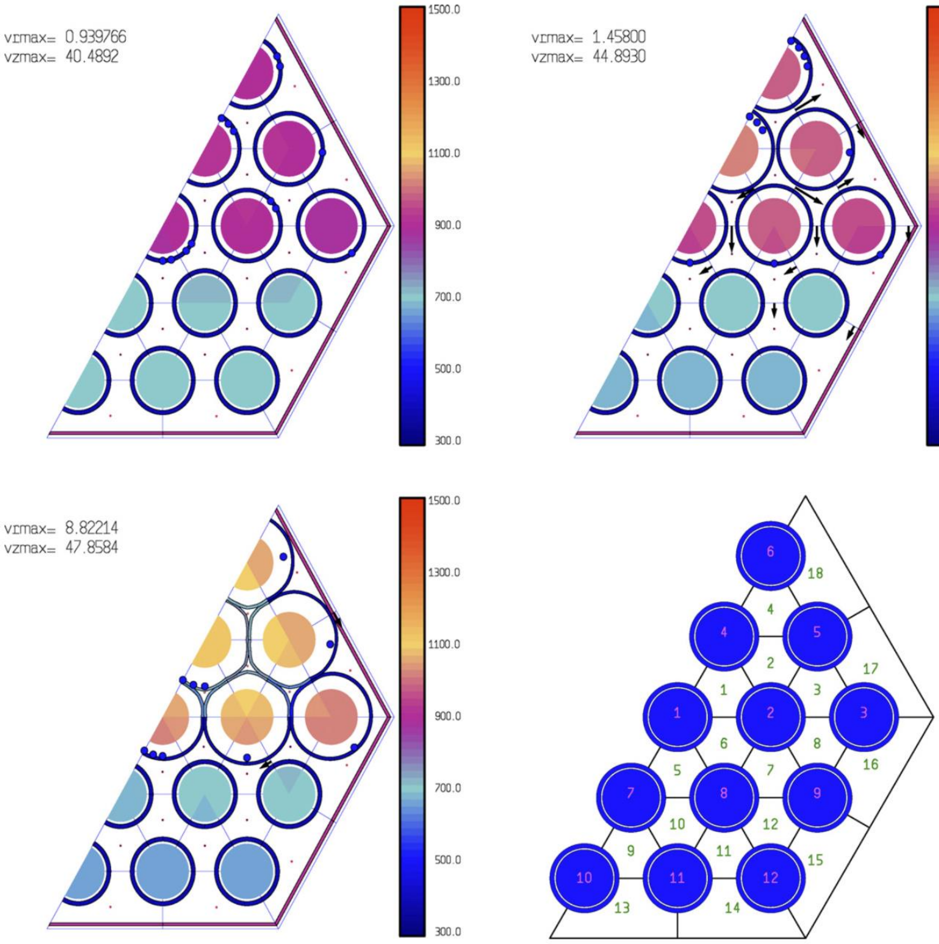


Figure 37
Temperature distributions associated with maximum blockages of 25%, 65% and 95% at 100 seconds.

7.5.6 Closing remarks

Concluding what was, when all is said and done, an early report of a very large and complex piece of work, the authors conclude that the modelling is working largely as intended, and that in general the correct physical behaviour is being predicted. Thermal hydraulics influences the ballooning, and the ballooning influences the thermal hydraulics. Various areas where, quite reasonably, further work is required are identified.

7.6 BISON

7.6.1 Overview

In the comments above about the DRACCAR code, in several places we attempted to bring out that in some senses the code was a numerical evaluation of some almost "analytical" approximate models of structural mechanics; things such as thin shell approximations for the behaviour of the cladding, for example. There is no particular intent here to single out DRACCAR; it is actually one of the more sophisticated codes of its kind, and essentially all current and historical fuel-modelling codes adopt this approach.

Arguably, nuclear fuel modelling codes have continued along broadly the same technical path that they set out on some 30 years ago. Meanwhile, more general structural analysis applications in essentially all other branches of engineering, and indeed in other aspects of nuclear engineering, have moved to a reliance on finite element methods. In essence these take the fundamental governing equilibrium equations, along with suitable constitutive laws, and discretize and solve them directly. As the name implies this is done using the finite element technique; essentially projecting the solution onto a space of basis functions, each basis function having support over only a very small region of the solution domain; the finite element. There are now of course many highly capable, very general purpose finite element codes commercially available, incorporating for example 3-D nonlinear material behaviour, and special element types and the like for fracture mechanics analysis. ABAQUS is one such, perhaps the best known, but there are several on a par.

The BISON code represents a distinct break from the classic nuclear fuel modelling approach, and it too adopts the finite element technique to solve the fundamental elastostatics equations describing the fuel behaviour.

There are by now many papers reporting the use of BISON. Two that provide a general overview of the code's capabilities and approach are [78] and [79]. The following short summary is extracted from these, but for a fuller picture these papers themselves should be consulted. In particular, we do not give any of the mathematical background here, but a summary is available in [78].

Bison has been under development at the Idaho National laboratory since about 2009, in part under the auspices of the United States CASL programme. It is a very general-purpose code, and can be used to analyse time dependent and steady spherically symmetric, cylindrically symmetric, and fully and three-dimensional systems. It has already been used to investigate a variety of fuel forms, including TRISO coated-particle fuel, metallic fuel in rod geometries and plate geometries, and LWR oxide fuel.

The mathematical and computational underpinnings of bison use a general-purpose finite element based solution framework developed at INL, known as Moose. This framework has been constructed to be able to be used on very large parallel machines, allowing computationally large problems to be tackled, such as a full LWR rod where each individual pellet is modelled in three dimensions. Attention has been paid to the software structure, and the code is highly object-oriented, so that new material and other behavioral models can be added relatively easily. This is of course a particularly important characteristic in any nuclear code, as the diagram above (Figure 29) makes abundantly clear.

As well as the 'basic' structural mechanics, BISON has a comprehensive fission gas behaviour and swelling model.

7.6.2 Sample BISON analysis.

As noted, there are many frankly rather impressive examples of the use of BISON, and we will select one here to illustrate.

We show here the results of a simulation of temperature, stresses and strains and in a fuel rod associated with what is known as a "Missing Pellet Surface defect". Essentially this relates to an occasional manufacturing defect where part of the outer periphery of a pellet is missing; see Figure 38.

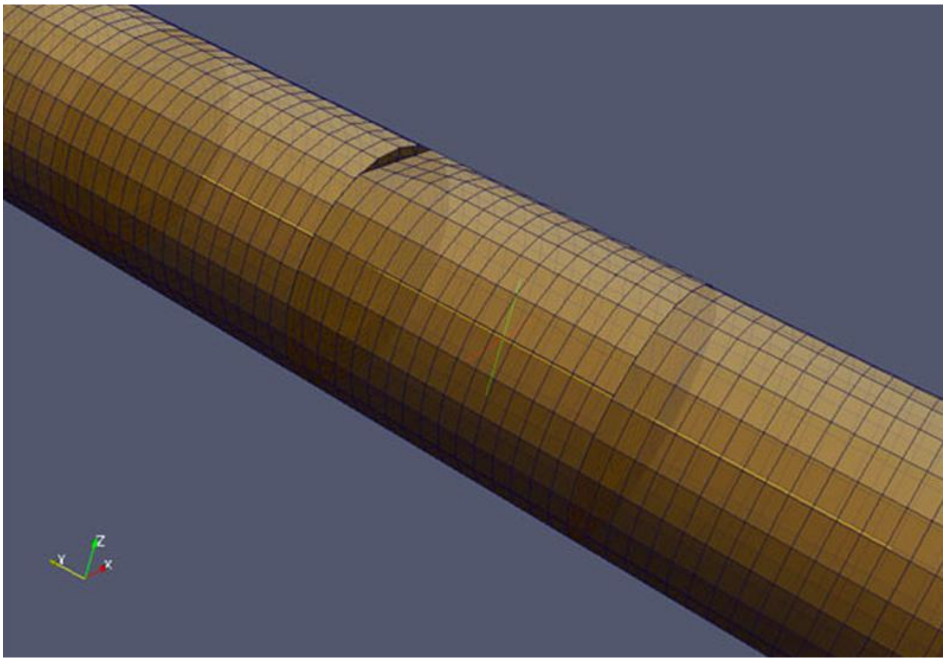


Figure 38
A typical 'Missing Pellet Surface' defect (from CASL)

The following results are taken from [79]. Figure 39 shows the temperature predicted at the end of a power ramp, for each of two different defect depths. The right-hand pair of images shows the deformed shape of the cladding, with the deformations magnified, showing clearly the marked axial dependence associated with pellet ends in general, as well as the local perturbation associated with the missing pellet surface. Figure 40 shows the stresses and deformation in more detail.

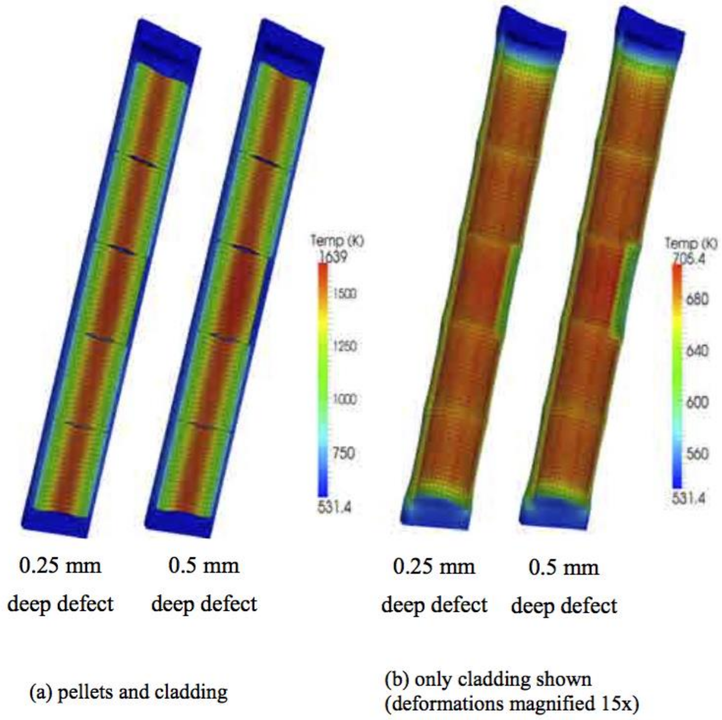
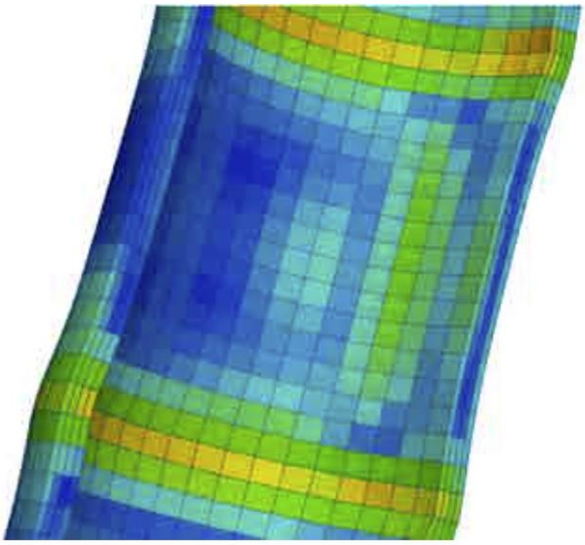


Figure 39
Predicted temperature at the end of power ramp



Zoomed-in view of 0.5 mm deep defect

Figure 40
The predicted displacements and von Mises stresses at the end of the power ramp

7.6.3 Closing remarks

BISON seems to be a highly capable, very 'modern' fuel code, under active development. It is also free to use.

7.7 Fuel Modelling codes: Conclusions

The classical approach to fuel modelling, epitomised from the early days in codes like TRANSURANUS, Mabel, FRUMP, TRAFFIC, has perhaps been brought to a peak in the present DRACCAR.

These developments have largely taken place in isolation from more general advances in structural mechanics modelling, in particular involving the now ubiquitous use of finite elements for sophisticated three-dimensional structural mechanics, including large deformations, fracture, plasticity creep and so on.

The approach taken by the developers of the BISON code is radically different. It seems to have embraced these more modern methods, and use them to build a framework into which the needed multiple complex physics models of nuclear reactor fuel modelling can be incorporated.

As such, it seems to offer much better prospects for longer-term advanced use. As one very minor caveat, we are not aware of any use of BISON for large deformation ballooning (although it may well have been so used, and we are simply not aware of it). However, even if it has not been we are confident there is fundamental in (or that could not be added to) the formulation that precludes it.

It seems to us by far the most attractive avenue for fuel modelling at this point.

It is our understanding that the BISON code is available without charge, as is TRANSURANUS. Access to DRACCAR is uncertain; it has not to our knowledge been made 'open' in the way the other two have. Access might well require (paid) participation in programmes that involve it. How this would apply (differently?) to (say) a commercial operator, a Regulator, or a university, we cannot say.

8 IS CLAD BALLOONING A NON-ISSUE?

8.1 Is there a problem?

For completeness, the first question needing to be addressed is whether or not the phenomenon of clad ballooning could credibly occur at all, and if it were to occur would it generate problems.

The answer to this is unequivocally yes. Multiple credible experiments, over many years, have demonstrated that when fuel is subject to the sorts of conditions that plausibly could occur in a reactor under perfectly credible accidents, ballooning is observed. Just how much ballooning, and under just which circumstances, is a complicated issue, but that is not the point. Incoherence in the ballooning, and ballooning turning out to be non-co-planar, may save the day, but we cannot be confident in this, so for present purposes it is not the point either. There are multiple circumstances under which significant ballooning has been observed.

8.2 Does anything need to be done about it?

In that we have had the problem, by definition, for the last 50 years it is not self-evident that anything needs to be done about it now. What harm has this potential problem caused us?

Have we just been lucky and not experienced the kind of accident where ballooning could make the outcome worse?

As discussed elsewhere, there is a wide range of circumstances and events that can lead to conditions where ballooning may occur. Happily we have experienced a tiny number of accidents, but that does mean that we have only very sketchy coverage of this parameter space. We are simply not in a position to say that absence of harm associated with ballooning is other than just good fortune.

Regulators and operators are of course well aware of this, and together have adopted pragmatic approaches to try to live with these circumstances. The regulators and operators are the ones who will know best just what are these pragmatic approaches, and what if any economic costs accompany them. Whatever are these pragmatic approaches, we can of course not be certain that they will eliminate any possibility of harm from ballooning, but equally that is of course not an objective or requirement. Reasonableness, in whatever form it takes in various jurisdictions, comes into play.

We have considered above the (strikingly extensive) range of experimental investigations, and the associated development of modelling and predictive capability.

It seems to us that despite this vast effort and expense, the robustness of evidence and understanding available to guide and underpin pragmatic approaches to accommodating the ballooning hazard within plant design and operation is still actually rather thin.

From a 'reasonable safety' point of view, there is a strong case to be made that efforts to gain further understanding are justified.

As noted, we cannot be sure of the cost associated with 'accommodation' measures, but it seems to us quite likely that a greater understanding could have worthwhile economic benefits, by reducing margins, increasing operating envelopes, or whatever.

We will take as our starting point for the next section the position that a greater understanding of, and ability to predict, ballooning is indeed worthwhile. We will discuss how in our view this might be achieved, and attempt to relate this to our understanding of current activities.

9 THE WAY FORWARD

9.1 Opening remarks

Early on in this report we discussed at some length the physical phenomena that seem likely to be important in ballooning. These were complex, numerous, and coupled.

We also discussed the breadth of the range of circumstances that could be lead to ballooning and the range of conditions from which ballooning could begin.

Amongst other things, this led us to the early observation that "demonstration" experiments, attempting of themselves to show directly how a plant will respond, are almost by definition doomed to be inadequate. It is simply not possible to conduct these on an adequate scale, and with adequate coverage of the wide parameter space, to provide the needed broad confidence.

This leads us to the view that a greater emphasis is needed on developing models and predictive capability. This is by no means, of course, easy or a panacea, but it seems to us that this offers the only way credibly to cover the needed parameter space and the associated difficulty in covering the space adequately with realistic experiments.

This is obviously not of itself a new idea. Also, one gains a sense from the publications (and activities) from, in particular, IRSN that there are thoughts in this direction also. There seems also to be something of a renewed interest in smaller scale, individual-phenomenon experiments, which is consistent with this.

9.2 A modelling approach†

A modelling approach that will handle all of the issues in a ballooning subassembly during reflood will need to be able to treat all the complexities of nuclear fuel undergoing large-deformation straining. It will need to be able to handle the chaotic, two-phase three-dimensional fluid flow that is providing such cooling as there is. It will also need to be able to accommodate the probably intimate coupling between these two sets of phenomena. We will address these issues in turn.

9.2.1 Fuel

There are many reasons, at least some of them good and understandable, for the very pronounced inertia and longevity associated with analysis and modelling in a nuclear context. This is very apparent in the area of the modelling of nuclear fuel. Our own work in the area, albeit itself a decade or so ago, used a code that predated our work by some decades, and other codes in current use now are of that same vintage. We have discussed fuel-modelling codes above.

If one were starting from scratch, with no history, it is hard not to conclude that an obvious starting point would be one of the several very capable general-purpose solid mechanics finite element codes. These can handle linear elastic behaviour in 3-D (obviously), but also can deal with nonlinearities such as creep and plasticity, and fracture. There are many additional phenomena that a nuclear fuel model needs to take into account (as evidenced in the rather wonderful diagram from 1975, reproduced here as figure X.) A well written basic

†And we consciously address this first, when habit and practice would generally have us first discuss experimental programmes.

framework, able to do all these things by default, would seem to provide a good structure into which to insert what are in essence further materials models, of things like swelling and gas release, that a nuclear fuel code needs. Given that, the choice made in the United States to build first an underlying general-purpose solver ('MOOSE'), and then to build around this a fuel code (BISON) seems surprising. However, we are not aware of the assessment that lay behind this choice, and are happy to assume that the assessment was performed appropriately. The important point is that the full capabilities of a 3-D modern finite element treatment, augmented with nuclear-specific models, is surely the obvious way to model fuel.

9.2.2 The fluid flow

From its earliest days, now some 50 years ago, it was essential for the nuclear industry to have the means to model complex transient two-phase flows in (inter alia, but also in particular) slender, one-dimensional piping systems. This need led to the development of the enormously competent collection of one-dimensional system codes they have done and will continue to serve for this extremely well.

Their ubiquitous nature, their competence, and the familiarity of the community with these codes has led to them being used in circumstances that arguably stretch the envelope of their validity. Indeed the fact that such application is commonly successful and useful is testament to their underlying capabilities. However, there is a tendency to attempt to add refinement and refinement to what is by modern standards fundamentally not the most capable approach now possible.

Reflood is probably one such an example. The modelling flow of steam and entrained droplets up a subassembly, with periodic spacer grids, followed by a region where the flow passages may have shrunken considerably, forcing flow diversion, does sit very ill on the starting point of the models of a code like RELAP.

It is hard not to conclude that starting with a clean sheet of paper, the starting point would be other than the very extensive two-phase flow capability currently exhibited by commercial CFD codes. That is not for a moment to suggest that this would be a magic solution. Even the modelling of single-phase flow up a subassembly, taking proper account of the complex three-dimensional spacer grid geometry, is a challenging task. Modelling this flow with entrained droplets, let alone with larger bodies of entrained water, depends amongst other things on a heavy input of empirical models derived from experiments of inevitably indirect applicability. However, these codes do at least provide a rigorous three-dimensional treatment, with rigorous enforcement of at least the various conservation laws in that geometry. As such, they probably provide the most reliable, comprehensive and flexible structure into which to insert the necessary and necessarily empirical models of phenomena that cannot be treated wholly from first principles.

Commercial CFD codes, as the name implies, are not free. However, most serious nuclear operators or developers will already have licenses to candidate codes. We do not know the position with regard to say a Regulator. However, if the cost is not trivial compared to the perceived value of the programme, the programme should not be undertaken. There are also freely available 'open source', 'academic' or similar codes.

9.2.3 Coupling

The need then arises for the coupling of the computation of the changed geometry of the fuel rods with that of the flow and heat transfer within the sub channels, which are of course changing in geometry in the same (strictly, inverse!) fashion.

Compared to the complexities of the computations being coupled, this is relatively straightforward.

There is a considerable "software engineering" requirement, associated with the running of multi-pin fuel models. This presumably can be by running a single instance of a fuel code, which has itself been written to solve multiple pins simultaneously, or via the approach of multiple instances of a single pin code. The choice no doubt matters a lot to the people doing the work, but from one step removed it is a fairly minor issue.

The other coupling is between the fluid flow code and the fuel code. The main information required by the fuel codes is the external heat transfer being caused by the two-phase flow past it. Computing this is routine for the CFD model. Indeed, there are interesting possibilities where one might even overlap the solution domains, as the computation of conjugate heat transfer into the solid boundaries around a fluid flow is a normal capability now for CFD.

The information required by the CFD code is the changed geometry. An ability to cope with a dynamically changing geometry, including re-meshing if required, is now widespread in CFD codes. (An extreme example is the computation of combustion processes in a reciprocating internal combustion engine, where obviously the CFD volume changes by a factor of perhaps 15 to 1 during a cycle.) Compared to this, the geometry changes associated with ballooning are modest, slow and "smooth".

9.3 The above approach in the context of extant and previous modelling activities

The best, indeed only, approach to implement multi-pin modelling is from the DRACCAR-CESAR combination discussed at length above (Section 7.5)

This itself is an incremental improvement on the work of Ammirabile & Walker of about 15 years ago. It is a much bigger program, and as such must be better, albeit there are some issues such as eccentricity and spacer grids that perhaps are not as well treated. Anyway, it is where we are and what we have.

The Ammirabile & Walker work involved the UK fuel code Mabel. Whilst in many respects DRACCAR and Mabel have the same underlying approach, which is very much not the "general, 3-D, finite element" approach recommended above, DRACCAR is undeniably more current and more capable than Mabel.

The fluid flow treatment within the coupled DRACCAR approach is based on CESAR, which is essentially a derivative of a one-dimensional systems code approach, albeit with many additional features incorporated. They do note a desire to incorporate a better fluid flow treatment.

All in all, the work of Ammirabile & Walker 15 years ago, and its replication and improvement currently at IRSN, give confidence to the view that a coupled, multi-pin treatment, combined with a fluid flow analysis, is what is needed. We believe that the approach we have suggested above, which takes what is probably the most capable and flexible current fuel modelling code, and a modern two-phase general-purpose CFD treatment, represents the most promising way to achieve this objective.

9.4 Experiments

If we had a capable coupled modelling capability we would be able to demonstrate this readily, by successful reproduction of the wide range of experimental measurements that have been reported over the last decades. Whilst none of these obviously are exact replicas what will happen in a reactor, the tests all embody various subsets of the conditions that will be found in a real reactor, and any competent modelling capability should thus be able to reproduce them adequately.

We do not have such capability, and that is the lack that has been addressed in the section above. One might almost say that at this stage we do not actually need more measurement results, but we rather need an ability to predict the ones we already have.

However, these comments apply to integral experiments, of course.

We have discussed elsewhere in this report the broad alternative objectives of experimental programs, from experiments that tell us how our plant will behave just by watching the experiment, to experiments designed to give us confidence that our more general modelling tool will make correct predictions under circumstances relevant to our real plant.

Once this changed motivation for experiments is accepted, it obviously causes different measurements to be undertaken. This is not the time for devising a detailed programme, but probably the best value would come from small-scale separate effects tests on issues to do with fluid flow, droplet (as opposed to vapour) diversion, and droplet impingement and breakup. Ultimately our interest is in modelling these phenomena under conditions relevant to re-flood, but one of the benefits of this more mechanistic, physically-based approach is that a great deal of information and confidence in the modelling can be gained from experiments under much less onerous and expensive conditions.

10 CONCLUSIONS

The possibility of clad ballooning, and the formation of uncoolable regions of the core, following various loss of coolant accidents, was known of relatively early in the development of light water reactors. However, the issue probably first really only came to prominence in association with the building in the United Kingdom of its first (and only) PWR.

That recognition spawned a series of tests attempting to determine experimentally just how much of a problem this was. Having been raised in this fashion, it is fair to say that the issue has if anything grown in prominence in the intervening decades, with concern extending to other jurisdictions. A correspondingly extensive series of experiments has by now been undertaken. Associated with this, there have been extensive attempts at modelling the phenomenon.

On reviewing the material, the most obvious conclusion that is forced upon one is astonishment at the size and longevity of these programs, taken in aggregate. A sceptic might feel that if the problem has not been sorted out by now, (a) does it really need to be sorted out, and (b) will it ever be? This latter comment he might feel was particularly justified since in many respects there is remarkably little difference between the work now being done to address this problem, and the work that was done some decades ago.

Having considered this, we have concluded that in our view, despite everything, this is a real world problem that will not go away, and where there is some real-world benefit in trying to resolve the issue. The fact that it is hard to resolve does not make this any the less so.

The complexity of the problem, and the wide range of conditions over which we need to be confident in the ballooning behaviour, are such that gaining this confidence via simply observing the behaviour of experiments is unlikely to be possible. We believe that a shift in emphasis of experimental programs towards those very explicitly aimed at (probably phenomenon by phenomenon) validation of first principles models is needed. Predictions by these models, would then play a greater role in gaining overall confidence in plant behaviour under the range of conditions necessary

There are nonetheless and high-quality efforts being made to tackle ballooning, in particular under the IRSN PERFROI programme. It is however notable that the modelling activities do betray the very great inertia and introspection so often associated with methods and approaches in the nuclear industry. (And indeed, these embody only incremental advances on very similar work begun ~15 years ago.)

We have reviewed the various modelling codes that are used in this area. This is complicated by the fact that modelling of this problem really requires two distinct forms of code; a fuel modelling code, and a very advanced thermal hydraulic capability.

The most modern, and what we believe would be the most capable fuel code, BISON, is not being used. (That comment perhaps sounds more critical than it is intended to; the timing of the availability of codes, and the inception of the programs in question, as well as nationalities, make this quite understandable).

The thermal hydraulic capability being used is based upon one-dimensional system code treatments. These have a very long pedigree in the nuclear industry, and tend to be used almost by default, but for what is very much a three-dimensional problem, with complex two-phase aspect, it is hard to think that modern computational fluid dynamics would not represent a more attractive approach. We finish by outlining a possible approach to modelling that makes maximum use of advanced methods, particular those not originating in the nuclear industry.

11 ACKNOWLEDGEMENT

It is a pleasure to acknowledge very helpful discussions with Tim Haste.

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