Potential Fuel Damage Within a Shipping Cask Following a Postulated Impact Accident

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INTRODUCTION

British Nuclear Fuels plc has, for many years, been involved in the transport of spent nuclear fuel to its Sellafield reprocessing and waste management complex in Cumbria, UK. The Company operates through its subsidiary, Pacific Nuclear Transport plc (PNTL) for sea transport from the Far East, and is a major shareholder in Nuclear Transport plc (NTL), covering transport from Europe.

The Nuclear Criticality safety of spent fuel transport requires engineered safeguards which are effective under both normal and accident conditions. The IAEA Transport Regulations provide the framework for safety assessments, and the large scale mechanical integrity of shipping casks is demonstrated by various methods, including drop tests.

The criticality safety of a cask usually relies on fixed neutron absorbing material engineered into the fuel element support framework, and the geometry and layout of fuel elements inside the framework. Theoretically, if fuel elements were to undergo massive fracture following an impact accident, releasing large quantities of UO_2 fragments to the flask cavity, a criticality hazard may arise if fragments settle in a water-filled free volume. Criticality survey calculations show that the reactivity effects are strongly determined by the amount of fuel assumed to fragment, and the specific cask design which determines the interaction between intact fuel elements, the assumed UO_2/water slurry, and fixed neutron absorbers.

There is a clear need therefore to evaluate an estimated upper limit to possible UO_2 fragment release for criticality safety evaluations under assumed accident conditions. The sensitivity of some designs to the amount of assumed UO_2 fragments meant that arbitrarily pessimistic assumptions on UO_2 release were not sensible. Advice from BNFL Fuels Division (the fuel manufacturing Division of the Company) was that nuclear fuel is mechanically robust, and unlikely to suffer massive fragmentation in a 9m drop accident. To arrive at a representative assumption for UO_2 release, a series of simple demonstration drop tests were carried out on unirradiated fuel. These were very severe tests, and served to test the general advice given on fuel impact resistance. These tests were then followed up by a study based on Post Irradiation Examination (PIE) experience, which extrapolated the conclusions to irradiated fuel.

SINGLE PIN DROP TESTS

An initial very simple test was carried out on three 5cm lengths of PWR zircaloy tubing. The hollow tube samples had a dummy "end-cap" push fitted into one end, and a 1.2cm diameter, 2.3kg steel bar fitted to the other end to simulate the UO_2 pellet stack (Figure 1). The samples were then dropped 9m down a 2.5cm bore tube onto concrete. This is an extremely severe test, as in the real situation with a welded end-cap and a UO_2 pellet stack inside the tube, much of the stress would be transferred up the rod. In this test, all of the impact energy is concentrated in the 5cm hollow tube length. The results showed no tendency for high hoop stress to burst the zircaloy tube. Damage was largely limited to deformations around the end-cap, and in one case where the end-cap split the tube, the crack did not propagate along the tube length, and appears only as a result of the circumferential stress caused by driving the end-cap into the tube.

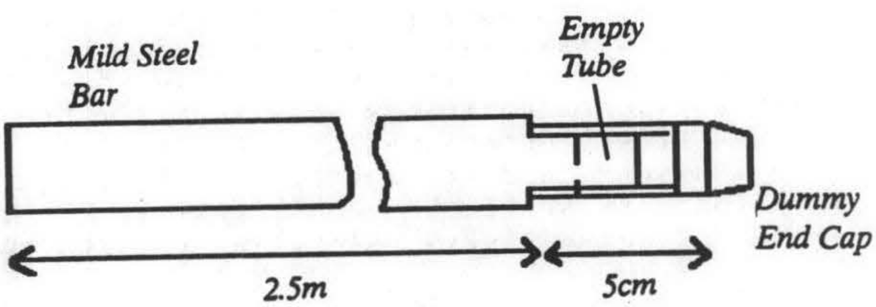

Figure 1 Arrangment of rod used for 9m drop tests.

These simple tests were followed by drop tests on actual fuel pins. The pins were zircaloy clad and contained UO_2 pellets, giving a pin weight of about 3.5kgm. Pin length was 378cm, diameter 1.22cm and zircaloy wall thickness was 0.66mm. The pins were dropped inside a 2.8cm diameter copper guide tube through 9m onto a rigidly supported steel bar of 360kg mass. Five pins were dropped, and high speed films were taken of five of the drops for subsequent analysis. Pin number 5 was dropped three times, to gain experience of the filming and timing techniques. Pin number 4 had a deliberate defect spark machined into the cladding. This took the form of a longitudinal slot about 12mm long, 0.3mm wide and 0.3 to 0.5cm deep (up to 75% of the wall thickness), located about 2 cm from the bottom end of the pin. The defect was meant to simulate a pin which was an imminent in-core failure.

None of the fuel pins fractured in any way, including the "defect" pin. Permanent axial distortions occurred, with circumferential wrinkling of the cladding at the pellet interfaces adjacent to the buckled region of the pins. Bending occurred in the region 5cm to 90cm from the impact end, with pin distortions up to about 1.4cm from the original centreline. Analysis of the film showed that pin number 1 did not reach theoretical impact velocity, presumably due to fouling on the guide tube. All other drops were unimpeded. The buckling wavelength of the distortions in the pins was found to correspond broadly with theoretically predicted values.

Although more representative than the initial tests, these drops are still very severe compared to the real situation. In a complete fuel assembly there are several substantive support grids which would limit axial buckling. Also, much of the impact would be absorbed by the cask, the inner "multi-element bottle" support frame and the fuel element top or bottom fittings.

The main conclusion which can be drawn for unirradiated fuel is that fuel is indeed quite robust, and impacts similar to a 9m drop will not result in fragmentation of pins and loss of UO_2.

EFFECTS OF IRRADIATION

The next task was to assess the effect of irradiation on the impact resistance of nuclear fuel. Work at Sellafield and elsewhere on Post Irradiation Examination (PIE) of fuel has been routinely carried out for many years. It was decided to use the Sellafield experience to advise on the effects of irradiation, and help come to a conclusion on representative assumptions for UO_2 release.

PIE experience shows in general that irradiation increases the yield and tensile strengths of zircaloy. Irradiation damage and hydrogen pick-up tend to decrease ductility however, but an elongation to failure of typically 8% or so may still be expected. The UO_2 pellets crack due to differential thermal expansion the first time the fuel goes to power. The degree of cracking is proportional to rating, so more severe cracking with smaller mean fragment sizes will occur in the axial centre region of the fuel. A typical histogram of UO_2 particle size is shown in Figure 2. A typical particle size of 3-4mm is apparent, with roughly 10% of fuel less than about 1mm size.

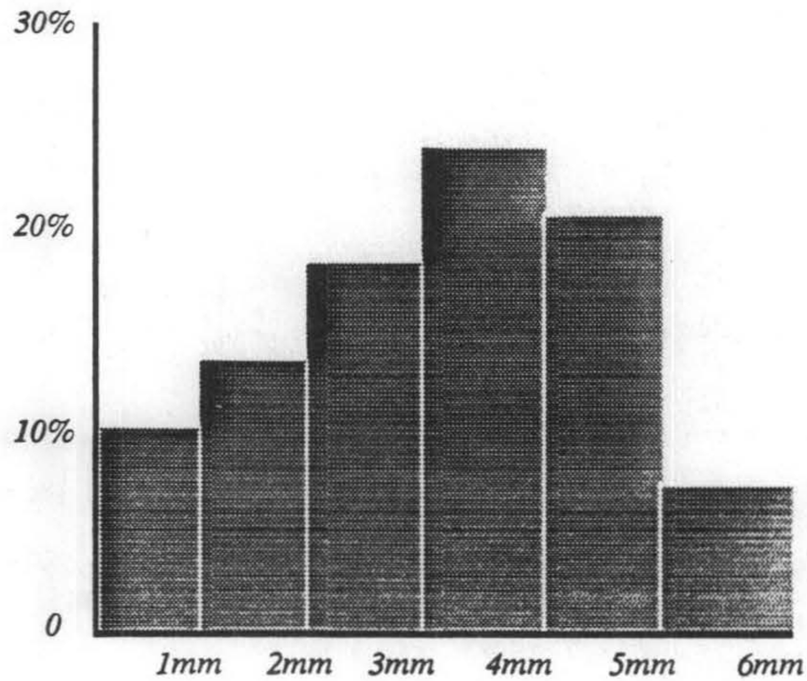

Figure 2 Typical LWR Particle Size Distribution.

During irradiation, pellet-cladding mechanical interactions occur, particularly at pellet ends where there may be strong contact between pellet and cladding. Strong fuel-cladding bonds have been observed to preferentially occur in the hottest and most highly rated regions of fuel pins. In addition, chemical bonding has been observed, formed by a complex oxide of U, Zr and Cs, which may be present even in low rated fuel. Bonds can be very strong, and PIE experience reports examples where fuel has had to be axially rammed, vibrated, or even prised free from cladding to break the bond.

A small number of fuel pins may fail during irradiation, a failure being defined as a perforation in the cladding which allows fission products to escape, and ingress of coolant water into the pin. The perforation may range from almost pin-hole size to multiple cracks. No significant quantity of UO_2 will be lost through such perforations directly, but the ingress of water may severely weaken the fuel pin. PIE experience shows that water inside a fuel pin may cause local hydriding at several places on the inner surface of the cladding, and perhaps the end-cap. Unlike gradual hydrogen pick-up which occurs during the irradiation of sound cladding (with little effect on impact properties), this localised secondary hydriding can severely embrittle the pin at the point of hydriding. Handling experience in PIE hot-cells shows that hydrided failed pins are brittle, and may fracture transversely on impact, usually around the axial centre. Conversely, handling experience with sound rods shows them to be quite robust.

ESTIMATION OF FAILED PINS AND FRAGMENT RELEASE

The problem now reduces to estimating the likely number of failed pins in a typical shipment, as these are clearly the only ones at risk of fracture, and the fraction of UO_2 likely to be released from fractured pins.

The bonding of pellet fragments to cladding, and the size and irregular shape of the fragments themselves, will inhibit the loss of fragments from a broken pin. Also, it would be expected that a substantial proportion of released fragments would be trapped between the fuel pins and the grid structures. To test this, a simple experiment was carried out in which a gravel simulate was used representing the UO_2 fragments. A 5 x 5 grid section (a reduced section of a PWR grid) was loaded with PWR zircaloy tubes to represent sound pins. The centre tube was loaded with the gravel simulate, and a transverse break simulated above the grid by unplugging the base of the centre tube. On average, about 60% of the gravel was retained by the grid structure. After violent shaking to simulate recovery operations, about 15% of the gravel remained trapped by the grid. As there are typically seven grids in a PWR element, it is likely that fuel fragments will face a very tortuous path to escape from the fuel element structure. The view emerges therefore that UO_2 is certainly not a free flowing powder, but an irregular granular material which is likely to bind and block, and become trapped by the fuel element grids even if released.

The number of pins failing in reactor cores has reduced over the years, as the industry has matured, and failure rates are now typically between one in a thousand and one in ten thousand pins, ie 0.1% to 0.01%. Pin failures have been found to group together, such that several pins may fail in one element.

In order to arrive at a representative estimate of UO_2 release which will give a sensible upper bound, it is now necessary to make some rather sweeping assumptions. If we assume that one element in each shipment contains failed fuel, but that 30% of all failed pins at the cycle discharge are concentrated into that element, this should give a pessimistic estimate of the number of pins at risk. The actual number of assumed failed pins in a particular cask will depend on the number of elements discharged in the reactor cycle, and the flask payload. Using pessimistic estimates of failure rate and surveying a wide range of reactors and BNFL cask designs gave an estimate of not more than 1.8% of payload to be assumed failed pins. It was considered prudent to add a further 1% to cover uncertainties in the number of failures, and a further 1% to cover other unknowns, giving approximately 4% as an upper bound. It is clear that even if 4% of pins fractured in an impact, much of he UO_2 would be retained in the element. However, it is difficult to quantify what the UO_2 release fraction would be, so pessimistically we may assume that all the UO_2 from fractured pins is lost to the cask cavity. A final upper bound estimate of 4% UO_2 loss under impact conditions may therefore be made.

SUMMARY AND CONCLUSIONS

- Criticality assessments of shipping casks must show nuclear criticality safety is maintained under impact conditions similar to a 9m drop.

- Criticality survey calculations have shown that for certain cask designs, the nuclear reactivity of the system is very sensitive to the amount of UO_2 fragments assumed to be released into free water volumes from fractured fuel.

- Work was carried out to establish a representative upper bound estimate of assumed UO_2 release.

- Advice that fuel is robust and impact-resistant was confirmed by simple but severe single pin drop tests.

- Advice from Post Irradiation Examination experience was that irradiated fuel was in general also robust, but that failed pins were brittle and at risk of fracture.

A pessimistic assessment of UO_2 release based on conservative assumptions of pin failure rate and distribution gave an upper bound value of 4% UO_2 release for use in criticality analysis.

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