The Criticality Implications of Taking Credit for Fuel Burn-Up

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INTRODUCTION

The first problem usually facing a criticality engineer is data specification. The design engineer and operator need a design and operating ``envelope'' of data which allows practical operation, but the criticality engineer must ensure that key parameters remain within specific limits to maintain criticality safety. Given the high safety standards demanded in the nuclear industry, a naturally cautious view is taken and pessimistic assumptions of parameter values are therefore made. This explains why assessments of spent fuel transport and storage assume unirradiated fissile compositions (the ``fresh fuel'' assumption).

Looking ahead, the pressure to achieve more economic designs and the need to extend design specifications is likely to increase. The large decrease in spent fuel reactivity resulting from fuel burn-up provides a potential way of meeting these requirements.

The challenge is to develop ways of taking credit for fuel burn-up with <u>no reduction</u> in safety. This raises various issues which are discussed briefly in this paper, together with a scoping study to investigate the scale of reactivity change with burn-up for an existing BNFL cask design.

CURRENT PRACTICE

The fissile content of spent nuclear fuel is one of a number of key parameters directly affecting nuclear criticality safety as well as many other aspects of spent fuel management.As such it requires is careful attention through transport, receipt and storage, plant feed, reprocessing and final product finishing and storage operations.

When carrying out a criticality assessment, ideally all variable parameters affecting criticality may be assumed to be at an optimum value. This ensures that the system is at its maximum achievable reactivity condition and if the system is still safely sub-critical, a simple and robust ``deterministic'' safety case may be made; ie ``Even if everything else goes wrong, you can't go critical!'' For example, natural uranium (magnox) fuel shows a small initial increase in neutron multiplication factor K with burn-up, as the build up of Pu initially outweighs the effect of U235 depletion. Criticality assessments therefore assume magnox fuel is at the optimum burn-up, ensuring that the maximum reactivity condition is acceptable during underwater handling and storage.

Usually it is not possible to allow every parameter to be at its optimum, and key parameters may be engineered or selected to be within a limiting value. For example, the diameter of a cylindrical vessel may be engineered to be below the critical diameter. The ``fresh fuel'' assumption is equivalent to selecting an enrichment to be no greater than a limiting value equal to the initial enrichment (prior to burn-up). Careful engineering and selection can maintain a robust safety case without placing onerous constraints on operations.

For many applications, positive criticality control of key parameters may be necessary. This requires operator action to monitor and maintain values within safe limits. For example, the concentration of a soluble neutron absorber added as a criticality control mechanism to a fuel dissolver must be maintained above some minimum safe level. Adequate safety is achieved by high reliability, and the safety case may use risk assessment to demonstrate this.

SCOPING CALCULATIONS

As LWR reactors are refuelled off-load, the burn-up of fuel may be quantised into reactor cycles. A first approach to scoping the potential credit for burn-up, and the effect of uncertainties, is to estimate the number of reactor cycles required to reduce cask reactivity to some target level. A simple scoping study was therefore carried out to estimate the burn-up required to extend the design specification of an Excellox 4 shipping cask with 7 PWR elements, from the original $3.5^{w}/o$ U_{235} limit to $4.5^{w}/o$ U_{235} initial enrichment.

A 16 x 16 KWU/KKU PWR element design was selected for calculations purposes, identical to the $3.5^{w}/o$ U_{235} case except for enrichment, with 1.08 cm (OD) pins on a 1.43 cm pitch.

Control rod thimble tubes were represented by twenty vacant fuel pins at appropriate positions in the element. Seven elements were modelled inside a Multi-Element Bottle (MEB), this being a water-filled, lidded inner container divided by neutron absorbing inner partitions into seven compartments. The MEB fits inside the Excellox 4 cask, which is a finned steel outer cylinder, within which is an annular lead liner. Figure 1 is a cut-away illustration.

The study consisted of two main stages:

- Reactor Lattice burn-up calculations to generate isotopic data
- Cask reactivity calculations using isotopic data from stage one.

The lattice burn-up calculations were carried out using the 2D deterministic LWR-WIMS code (Halsall 1982) with a 69 energy group cross section library from the 1981 WIMS nuclear database (Halsall, Taubman 1983). The code models the fuel element in the reactor fuel core lattice at power. Keeping the element rating fixed, the calculated neutron flux is applied to burnable fuel materials to calculate build-up and depletion reaction rates. The neutron flux at constant fuel rating is re-calculated at steps through the calculation to accommodate the change in flux level and spectrum as the fuel composition changes. A simple fuel cycle was assumed, consisting of three full power cycles of 290 days, with 60 days at zero power between cycles to represent refuelling shut downs. At the end of each cycle, fuel data was transferred to a second LWR-WIMS calculation which included the materials required for the cask calculation (eg the neutron absorbing partitions and the lead liner). This second calculation allowed fuel to cool for a 365 day period to accommodate isotopic changes due to radioactive decay. At this stage, fission products were omitted from the calculation. Data was thus generated for and of cycles one, two and three, representing discharge burn-ups of 11 GWD/Te, 22 GWD/Te and 33 GWD/Te respectively. This data consisted of all cask, MEB and fuel materials (except fission products), processed into the 69 energy group WIMS format.

The cask criticality calculations were carried out using the MONK 5W code (Hutton 1979). This is a monte carlo code with full 3D geometry capability, and the ability to accept WIMS group averaged data. Results are shown in Table 1, and displayed in Figure 2. Note that SD is the code standard deviation and a value of keff + 3SD is usually assumed as an upper bound to the monte carlo calculation.

Fuel Burn-up GWD/Teu	Keff ± SD	Keff + 3SD
0	0.9536 ± 0.0031	0.9629
Cycle 1 11.02	0.9129 ± 0.0031	0.9222
Cycle 2 22.04	0.8752 ± 0.0034	0.8854
Cycle 3 33.06	0.8205 ± 0.0043	0.8298

Table 1 Monte Carlo Results

The results show an almost linear decrease in cask keff with burn-up, with a slope over the first two cycles of about -0.0036 K/GWD/TeU. The uncertainty in the results is almost as important as the results themselves in this case. The uncertainty in the isotopic composition must be translated into equivalent margins on the fuel burn-up level, and combined with the uncertainty in the reactor operator estimates of fuel burn-up at discharge. The nominal one standard deviation uncertainties in LWR-WIMS isotopic predictions are ± 1% for U_{235} and ± 3% for Pu/U ratio. The nominal one standard deviation uncertainty in fuel burn-up prediction was taken to be typically ± 6%. Translating these into equivalent uncertainties around the 4 GWD/TeU value gives roughly ± 2.1 GWD/TeU, at a nominal 99% confidence level. An upper bound value for minimum irradiation is thus approximately 6.1 GWD/tU, to meet a safety criterion of keff + 3SD $\leqslant$ 0.95.

For this particular cask/MEB/fuel combination, it appears that one full power cycle is ample to allow extension of the design base to 4.5 $^{W}/o$ U_{235}. The slope of the burn-up line and its uncertainty are of key importance of course. Generally, it is expected that for most BNFL cask/MEB designs, one cycle of burn-up would be enough to accommodate foreseeable increases in enrichment, even with a generous allowance for uncertainties. Specific detailed assessments would, of course, be necessary to confirm this for any particular case. Any monitor to check burn-up would, in this case, only need to show that the spent fuel is radioactive, provided the reactor has been on full power for at least one cycle (and no premature discharges of fuel have occurred). Although estimates of uncertainty are approximate, it is clear that the significant level of uncertainty involved would require very careful assessment if new designs of cask rely on multiple cycles or full burn-up of fuel.

FUTURE DEVELOPMENTS

The use of multi-element bottles for transport and storage of spent fuel has the criticality advantage that any MEB assessed safe for transport is also safe for underwater storage. Extending the safety case for storage to accommodate higher initial enrichment fuel therefore becomes a relatively simple extension of the transport safety case necessary to get the fuel to Sellafield for storage.

The THORP (Thermal Oxide Reprocessing Plant) dissolver will be controlled by dissolved neutron poison, allowing a wide range of enrichments to be processed safely by changes in poison concentration. The fissile content of the spent fuel will be monitored in the feed pond just prior to shear and dissolution, which gives the potential ability to match poison concentrations quite closely to the fissile content of the fuel charge. The THORP design therefore already contains features to take credit for fuel burn-up, once the performance of the fissile content monitor is demonstrated by early experience.

Initially, credit for fuel burn-up is likely to be used to extend the design base of existing designs, using a simple and robust approach which relies on coarse estimates such as full power cycles rather than precise estimates of burn-up. Before taking full credit for multiple cycle or full burn-up of fuel, a rigorous analysis of uncertainties is necessary. The validation of isotopic concentration and burn-up prediction codes would assume a criticality safety-related status in this case. It is likely that more PIE data will be necessary to quantify uncertainties and given the high cost of PIE work, collaborative ventures in this area may be of great benefit. Although for simplicity this scoping study neglected fission product poisons, Oak Ridge work (Parks 1988) has shown convincingly that selected fission products can be included without risk of reactivity increase with cooling time. (Reliance on fission products, however, does mean more isotopes for inclusion in the uncertainty analysis and PIE work).

A greater operating burden will be placed on reactor storage pool operators as correct identification of fuel and accurate burn-up records will become pre-requisites for transport safety. However, this should not present major new problems given the high standards of fuel handling in the industry. The need for a physical check on burn-up prior to cask loading requires careful consideration, but it may be that a fairly simple instrument which effectively counts reactor cycles may be sufficient.

Overall, in the longer term, it is certain that credit for burn-up is too valuable an option to discard. The ways and means of maintaining safe operations must be cautiously and rigorously developed over a period of time, and initially simple and robust methods with generous allowance for uncertainty should be applied.

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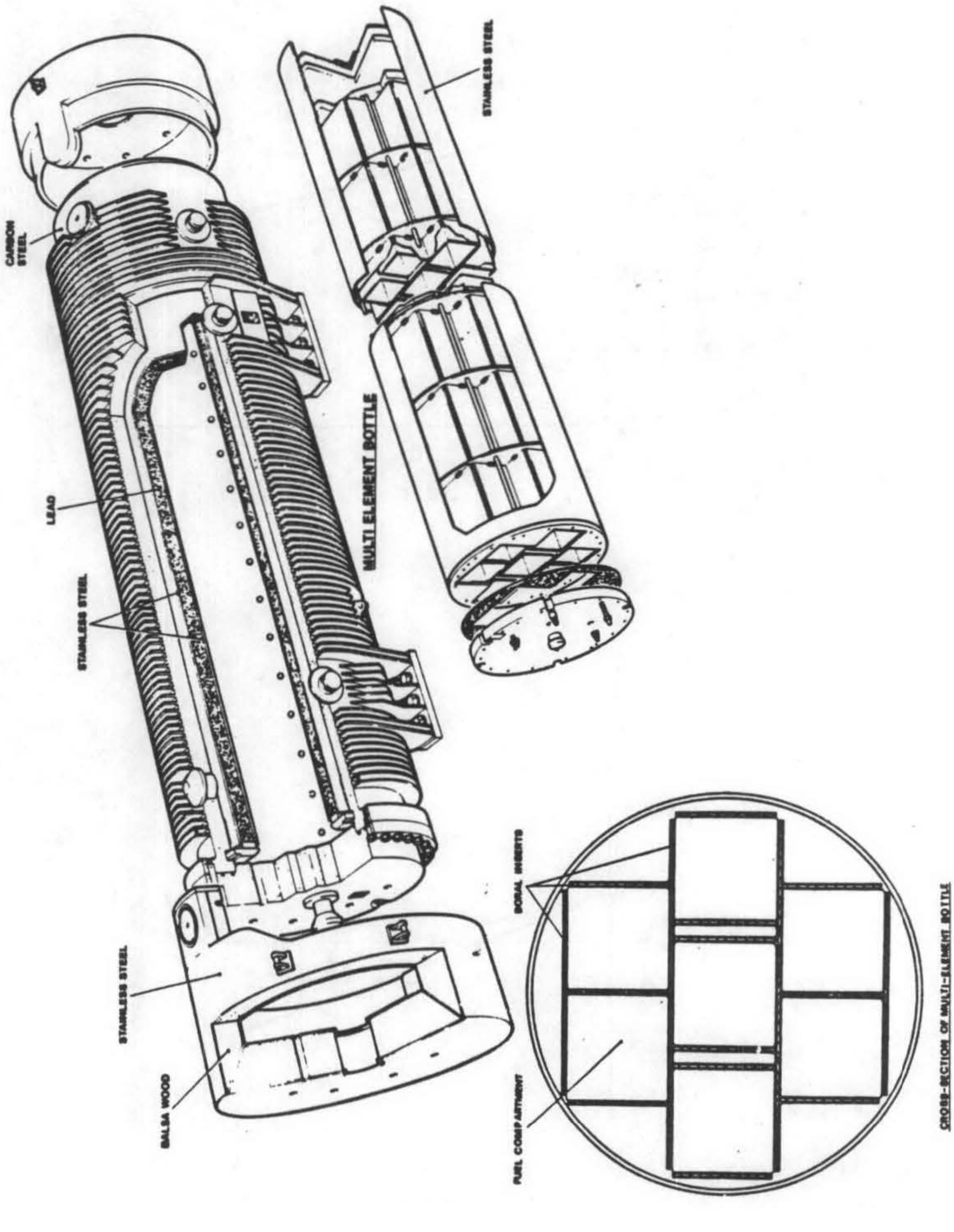

FIG. 1 EXCELLOX IV IRRADIATED FUEL TRANSPORT FLASK

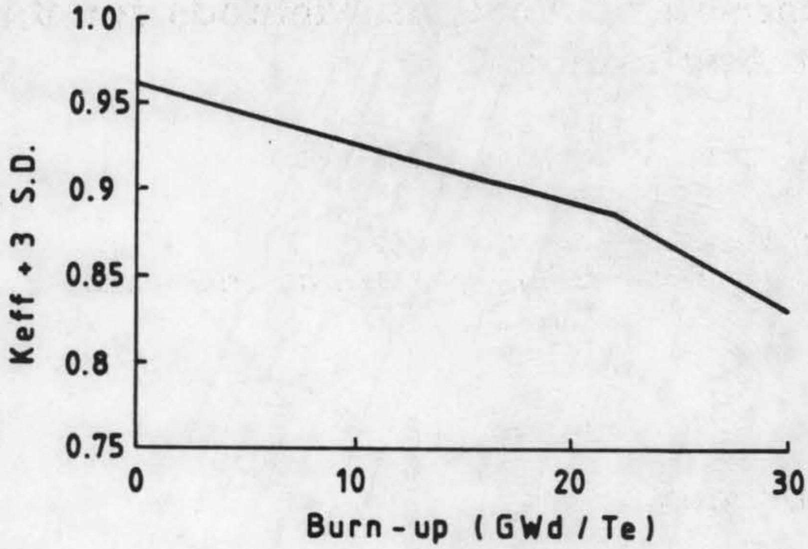

Fig. 2a

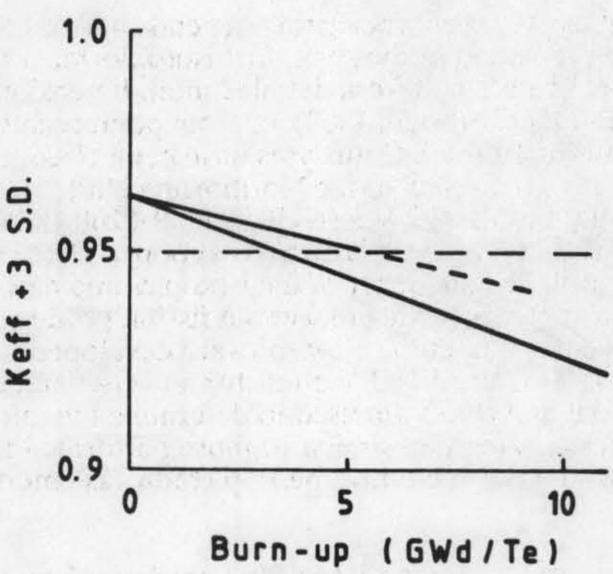

Fig. 2b

FIG. 2 SCOPING CALCULATION RESULTS