

Le présent document a été reproduit
avec l'autorisation de CANCOPY.
La revente ou la reproduction ultérieure
sont interdites.

THE APPLICATION OF DIMPLE IRRADIATED FUEL MEASUREMENTS TO
BURN-UP CREDIT VALIDATION

G Ingram, J Marshall

AEA Technology
Winfrith Technology Centre
Dorchester, Dorset, DT2 8DH, United Kingdom

SUMMARY

Benchmark measurements in the DIMPLE reactor of the relatively large loss in reactivity with burn-up are described in the context of studies with power reactor fuel irradiated to 20GWd/t(U). The results are discussed in the light of predictions obtained with current WIMS calculation methods, where it is shown that these provide a satisfactory level of agreement with the measurements.

INTRODUCTION

Commercial pressures are steadily advancing the case for the acceptance of burn-up credit in routine criticality safety assessments. Validation for this approach is being provided by a continuing programme of well-defined experimental benchmarks in the low power DIMPLE reactor at Winfrith [1], using small samples of irradiated fuel, discharged from a range of commercial power stations.

A key element of the programme is the measurement of the reactivity loss with burn-up in different neutron spectra, covering the range likely to be encountered in normal and accident criticality assessment scenarios.

Using a series of measurements with fuel from one of the UK's Commercial Advanced Gas-Cooled Reactors (CAGRs) as an example, this paper outlines the techniques employed at DIMPLE and discusses the application of the results to the validation of one of the current calculation methods, the WIMS code [2].

EXPERIMENTAL TECHNIQUE

A CAGR fuel element consists of 36 stainless steel clad uranium oxide pins arranged in three concentric rings and contained within a graphite sleeve. For the DIMPLE measurements three samples of fuel were chosen, one from each of the rings, that had been irradiated to 20GWd/t(U) and had cooled for about two years.

As in all DIMPLE irradiated fuel work, samples 120mm long were selected from regions of the fuel element where the burn-up had been relatively flat and where the power history was well-defined. These

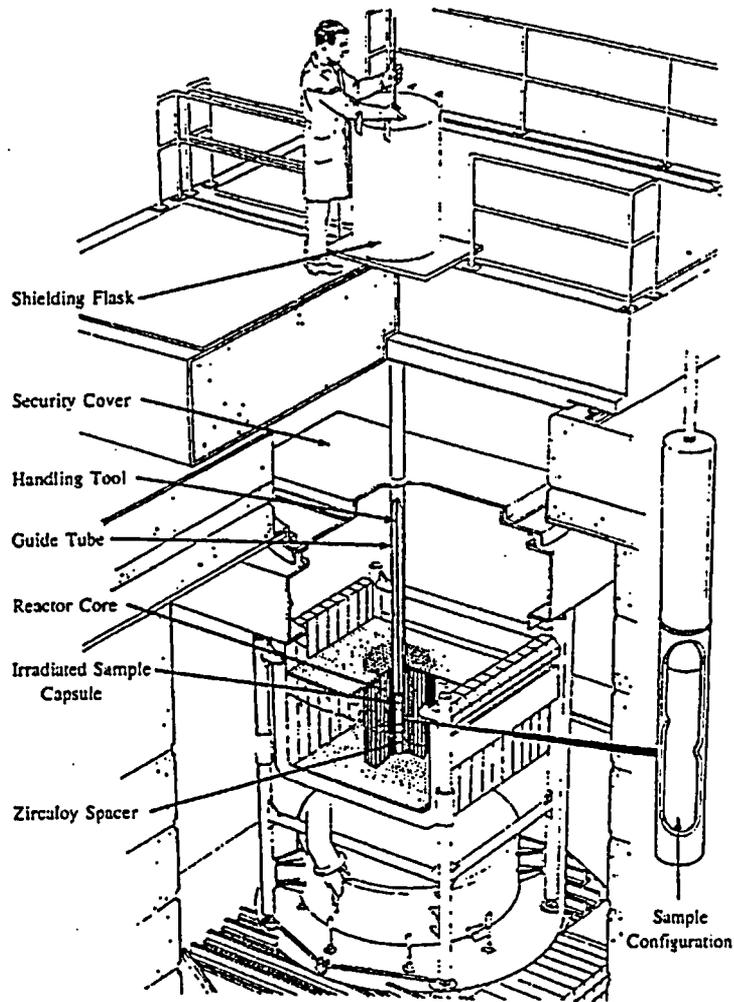


FIG.1 CUTAWAY SHOWING USE OF IRRADIATED SAMPLES IN THE DIMPLE REACTOR

II-100

samples, which included the stainless steel cladding, were then welded into low neutron absorption zircaloy cans. Chemical and isotopic analyses from adjacent regions were interpolated to establish the principal actinides and the total fissions in each of the samples.

The layout for the reactivity measurements in DIMPLE is shown in Figure 1. The period technique was used throughout, the reactor periods resulting from inserting and removing the samples being converted to reactivity using the Inhour equation. To establish a reactivity scale which spanned unirradiated and irradiated CAGR samples, measurements were also made with a range of fissile and neutron absorbing standards. These included uranium and plutonium with various isotopic compositions, to facilitate identifying the source of any discrepancies between the predicted and measured irradiated sample reactivities.

To establish the reactivity change with burn-up in different neutron spectra, measurements were made at the centre of three assemblies, all of which were light water moderated. The first, see Figure 2, was a compact array of stainless steel clad 7% enriched uranium oxide pins, where a large proportion of the neutron absorption was in the resonance region. In the second, see Figure 3, a central light water region was enclosed by annuli of stainless steel clad 3% and 7% enriched uranium oxide pins. Here, because the neutron absorption in the samples was negligibly small compared with that in the light water, the reactivity changes were directly related to the fissile content of the samples and so provided a cross-check on the chemical and isotopic analyses. Figure 4 shows the third assembly, which was similar to the second except that the central region consisted of an aluminium tank filled with heavy water, providing an environment where most of the neutron absorption took place at thermal energies.

In addition to the reactivity studies, measurements were also made of the neutron and gamma emissions from the irradiated fuel. These investigations, which are relevant to burn-up monitoring, have already been described elsewhere [3], [4].

CALCULATIONS

The WIMS-E code [2] and the 1986 WIMS library [5] were used to predict the results of the experiments. The burn-up calculations were based on a model of a CAGR cluster in its surrounding graphite, irradiated in accordance with the station power history reduced to histogram form. For DIMPLE, the various contributions to the sample reactivities were obtained from a perturbation edit routine. This combined the fluxes, number densities and microscopic cross-sections from the WIMS-E model for each sample in each of the three assemblies with adjoints appropriate to the unperturbed configuration obtained from SNAP [6] calculations. Further details of the methods are given in Reference 3, which also discusses the results from a reactor physics point of view.

TOR

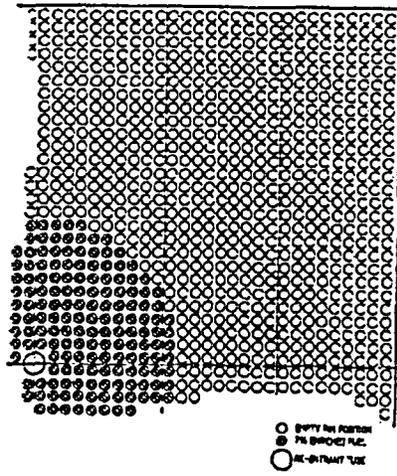


FIG. 2 PIN LAY-OUT OF ASSEMBLY A

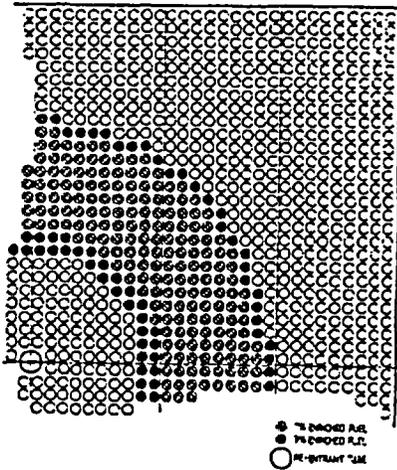


FIG. 3 PIN LAY-OUT OF ASSEMBLY B

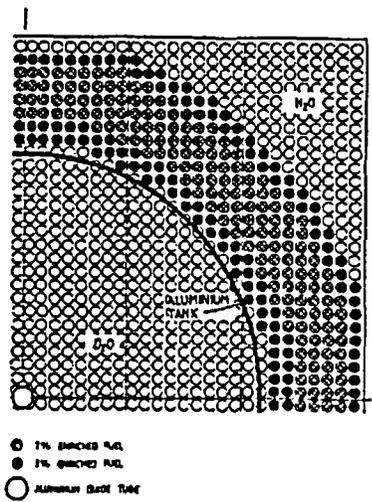
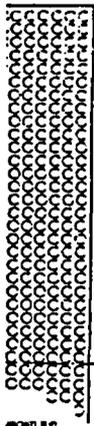


FIG. 4 PIN LAY-OUT OF ASSEMBLY C



WORKED A.D.
WORKED A.D.
STAND TUBE

BLY B

TABLE I
K-INFINITY BREAKDOWN FOR UNIRRADIATED AND IRRADIATED 2.55% ENRICHED CAGR FUEL
IN DIMPLE ASSEMBLIES

Contributors	Thermal Neutron Spectrum (Assembly C)				Resonance Neutron Spectrum (Assembly A)			
	Unirradiated		Irradiated		Unirradiated		Irradiated	
	Absorption	Production	Absorption	Production	Absorption	Production	Absorption	Production
Cladding	0.085	-	0.107	-	0.065	-	0.069	-
Fission Products	-	-	0.072	-	-	-	0.067	-
Uranium	0.915	1.575	0.536	0.697	0.935	1.276	0.590	0.550
Plutonium	-	-	0.280	0.473	-	-	0.268	0.364
Other Actinides	-	-	0.005	-	-	-	0.006	0.001
Total	1.000	1.575	1.000	1.170	1.000	1.276	1.000	0.915

The DIMPLE models have also been used to illustrate the large reductions in k-infinity with burn-up that apply in a wide range of neutron spectra. Table I shows a breakdown of the k-infinity values for unirradiated and irradiated samples from the central ring of pins in the CAGR cluster, where to simplify comparisons the absorption has been normalised to unity. In the assembly where thermal neutron absorption predominates, it can be seen that k-infinity is reduced from 1.58 to 1.17. The values are correspondingly lower when resonance absorption predominates, where there is proportionately less U235 fission and more U238 capture. Here the values are 1.28 for the unirradiated sample and 0.92 after irradiation.

Essentially the reduction in k-infinity corresponds to a change in the uranium enrichment from 2.55% to 0.88%, accompanied by a build-up in plutonium events to a level where the neutron production approaches that in the residual U235. In fact for the sample from the outer row of CAGR pins, Pu239 is the major source of fission neutrons. The other actinides, neptunium, americium and curium make a negligibly small contribution to k-infinity. Neutron absorption in the stainless steel cladding is broadly the same throughout and this is also the case for the fission product absorption, where in the two neutron spectra, it represents about 7% of the total.

DISCUSSION OF REACTIVITY RESULTS

Table II compares the WIMS-E predictions with the irradiated fuel reactivity measurements. Two sets of irradiated sample compositions were used. The first set was obtained directly from the burn-up calculations with the station power history. The second used the most precise data available. Thus, the predominant actinides were taken from the chemical and isotopic analyses for the samples, and the

fission product inventory and the less important actinides came from a burn-up calculation where the station power history had been normalised to the total fissions derived from the measured Nd148 yield in the samples.

TABLE II
COMPARISON OF MEASURED AND PREDICTED SAMPLE REACTIVITIES
(10^{-5} dk/k)

DIMPLE Assembly	Sample Enrichment and Cluster Location	Measurement	WIMS Prediction	
			Precise Data	Station Data
Central Light Water Region (Assembly B)	2.01% outer	36.2±0.4	36.4	36.1
	2.55% middle	50.6±0.4	50.9	50.6
	2.55% inner	53.2±0.4	53.5	53.1
Central Heavy Water Region (Assembly C)	2.01% outer	4.1±0.8	4.0	3.9
	2.55% middle	29.6±0.8	28.8	29.1
	2.55% inner	34.4±0.8	33.5	33.7
Compact 7% Enriched Array (Assembly A)	2.01% outer	68.0±0.5	65.0	64.6
	2.55% middle	52.4±0.5	49.7	48.7
	2.55% inner	48.7±0.5	46.4	45.5

In practice the differences in the two sets of compositions were relatively small. The correction to the total fissions required an adjustment of only a few percent to the power level in the burn-up calculation. For the isotopic contents there was a tendency for the calculations to underestimate the amount of uranium present, in particular the U235, and to overestimate the amount of Pu239; the other plutonium isotopes were underestimated. However, the overall impact of these changes was small, generally corresponding to less than 1% in k-infinity.

The expected agreement between the calculated values from both sets of compositions is generally confirmed by the results in all three DIMPLE assemblies in Table II. Furthermore, the very good agreement with experiment in the assembly with the central light water region shows that the fissile content of the samples is being predicted to better than 1% (1 standard deviation), confirming that there are no unexpected sources of neutron production in the samples. In the assembly where thermal neutron events predominate there is again very good agreement between both sets of predictions and experiment, indicating that the neutron absorption, including that in the fission products, is being predicted satisfactorily. Both sets of predictions show significant levels of disagreement in the assembly

7C-II

where resonance events predominate, where a detailed analysis indicates that the fission product neutron absorption in the resonance region may be larger than predicted [3].

In practice, although the differences between prediction and experiment are an important pointer to shortcomings in the nuclear data used in the analysis, these differences are relatively small in the present criticality context. Thus, using the k-infinity values in Table I and assuming that, to a first approximation, for similar samples the deviation of k-infinity from unity is proportional to the reactivity, it follows that the average discrepancies between prediction and experiment in both assemblies are satisfactorily low i.e. less than 0.01 in k-infinity. The associated uncertainty is less than ±0.005.

In the light of the above analysis it is clear that the current WIMS methods and data can provide good predictions of burn-up inventories and fuel reactivities for 20Gwd/t(U) CAGR fuel and could be applied with a reasonable level of confidence in burn-up credit assessments.

FUTURE PROGRAMMES

The measurements with the CAGR fuel were followed by a similar programme with LWR samples from the Swiss Beznau and Spanish Zorita power stations. The burn-ups in these cases were in the range 20 to 50Gwd/t(U). Analysis of these measurements is scheduled for completion later this year and should help to provide a better understanding of the discrepancies between prediction and experiment at resonance energies.

Early next year further measurements are planned as part of the CERES programme, which is being undertaken in collaboration with the CEA. This entails the exchange of samples between the AEA (Winfrith) and the CEA (Cadarache) and a complementary programme of experiments in the DIMPLE and MINERVE reactors. The combined data-bank from these studies will go a long way towards alleviating the current shortage of Quality Assured benchmarks in this field and will provide a powerful tool for future validation of methods and data, where attention will be focussed primarily on the new JEF-2 nuclear data library.

CONCLUSIONS

Measurements of the relatively large reactivity change with burn-up have been completed in a range of neutron spectra in the DIMPLE reactor, using samples of CAGR fuel irradiated to 20Gwd/t(U). The experimental uncertainties corresponded throughout to about ±0.005 in k-infinity (1 standard deviation).

The experiments have been compared with predictions from current WIMS methods. Satisfactory agreement was obtained, the differences between calculation and experiment corresponding to less than 0.01 in

to from a
normal-
d in the

ion
tion
ta
.1
.6
.1
.9
.1
.7
6
7
5

is were
d an
-up
r the
parti-
ther
mpact
n le in

both
ll
j
: water

that
ples.
s
xperi-
the
of
bly

SC-II

k-infinity, confirming that the WIMS methods can be used with a satisfactory level of confidence in burn-up credit assessments for 20GWd/t(U) CAGR fuel.

The data-base will be broadened later this year, when analysis of further measurements with irradiated LWR fuel with burn-ups in the range 20GWd/t(U) to 50GWd/t(U) is completed.

Early next year, CERES, a major collaborative programme on irradiated fuel will be undertaken by the AEA and CEA, facilitating a powerful coordinated approach to the validation of the new JEF-2 nuclear data library.

ACKNOWLEDGEMENT

The irradiated fuel samples were provided by what was formerly the Central Electricity Generating Board and is now Nuclear Electric. The authors acknowledge the support received.

REFERENCES

1. INGRAM G. DIMPLE and its Current Experimental Programme. International Seminar on Criticality Studies, Programs and Needs, Dijon, 1983.
2. HALSALL M J; POWNEY D and MOROH M R. LWRWIMS, The WIMS Code for Light Water Reactors. Proceedings of the International Conference, Physics of Reactors: Operation, Design and Computation, Marseille, April 23-27, 1990.
3. MARSHALL J, INGRAM G, GULLIFORD N T and DARKE M. Irradiated Fuel Measurements in DIMPLE. Proceedings of the International Conference of Physics of Reactors: Operation, Design and Computation, Marseille, April 23-27, 1990, 2, XI, 45-53.
4. INGRAM G, KNIPE A D and FRANKLIN B M. Development of Subcritical Monitoring Techniques Using the DIMPLE Reactor. Proceedings of the International Topical Meeting on Safety Margins in Criticality Safety, San Francisco, California, November 26-30, 1989, p104.
5. HALSALL M J and TAUMAN C J. The '1986' WIMS Nuclear Data Library. AEEW-R2133, 1986.
6. McCALLIEN C W J. SNAP-3D a Three-Dimensional Neutron Diffusion Code. TRG 2677 (R), United Kingdom Atomic Energy Authority, 1975.