INTERNATIONAL SEMINAR ON NUCLEAR CRITICALITY SAFETY
October 1987, Tokyo, Japan

CRITICALITY EXPERIMENTS IN THE DIMPLE REACTOR

G Ingram, J M Stevenson
UKAEA, AEE Winfrith, Dorchester, Dorset
0305-63111-2082

ABSTRACT

A large number of precisely defined fuel pin lattices have been studied in the DIMPLE reactor since the plant was refurbished in 1983. The assemblies range from simple geometry benchmarks to complex critical and sub-critical simulations of transport flask geometries. Techniques development has included sub-critical monitoring and the design of specialised assemblies for the study of irradiated fuel. The results from these assemblies compare satisfactorily with predictions based on current UK calculational methods, but have indicated a number of areas where improvements are required if higher levels of confidence are to be achieved.

I INTRODUCTION

DIMPLE is a versatile, low power, water-moderated reactor, located at AEE Winfrith. The plant was refurbished in 1983 and is currently being used for a criticality-related programme of experiments. Many of the experiments are of a generic nature but investigations funded by the industry and directed at specific problems, also feature prominently in the programme.

The main aims of the experimental programme are to:

(a) Validate the methods and data currently used in criticality assessments in the UK, in particular the MONK6 Monte Carlo code (1) and the WIMS codes (2), both developed at AEE Winfrith;

(b) Mock-up particular items of plant, under normal and accident conditions, and so reduce error margins in the criticality assessments;

(c) Investigate and develop sub-critical monitoring techniques for application in a plant environment.

Following a brief description of the reactor, the experiments carried out during the past three years are reviewed below.

II THE DIMPLE REACTOR

The main features of the reactor are illustrated in Figure 1. A more detailed description is given in reference 3. The design is simple and versatile. Assemblies of fuel pins are located by lattice plates attached to U-shaped beams within a large, aluminium reactor tank, 4m high and 2.6m in diameter. The tank is enclosed in steel-lined concrete shielding, which also provides secondary containment. Stainless-steel pipes link the reactor tank to dump tanks, which are located in a pit and which accommodate the water when the reactor is shut-down.

The reactor is controlled by water level alone, precise amounts of water being added or removed to balance the reactor at power. Shutdown is by means of a fast-dump system, which drops the water level in the reactor tank by 300mm in about 1 second. Although, in principle, DIMPLE can be operated with banks of conventional control-rods, the present arrangement is ideally suited to criticality studies, as it avoids any perturbation of the lattice.

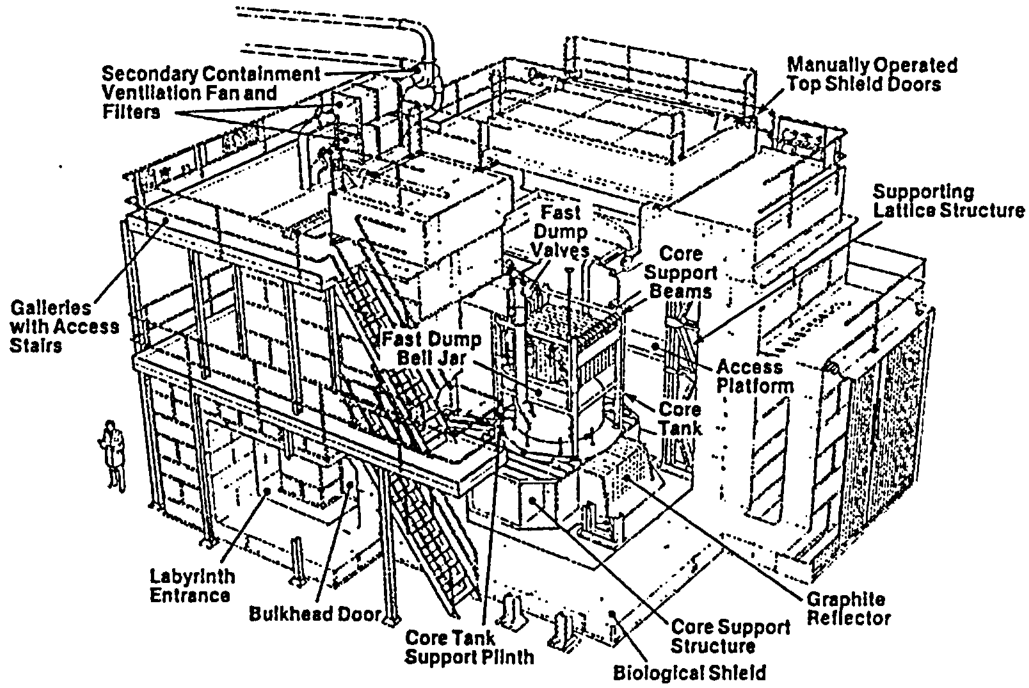

Figure 1 - General View of DIMPLE Reactor

III SIMPLE GEOMETRY ASSEMBLIES

The first assembly, in the present series of experiments, Assembly S01, was a re-build of an earlier DIMPLE benchmark (4). In this steel-clad 3% enriched pins were arranged on a square pitch in a simple cylindrical configuration, with a high neutron leakage, ie with over 20% of the neutrons leaking from the core. These studies served a dual purpose. Firstly they allowed the benchmark data to be re-assessed using up-to-date experimental techniques. Secondly they provided a clean-geometry reference assembly for the subsequent programme in Assembly S02, where nearly 20% of the neutrons were absorbed in a boron-steel walled transport/storage skip. A summary of the lattice parameters and experimental programme for Assembly S01, along with those for the other assemblies in the present series, are given in Table 1.

Assembly S03, which was developed for irradiated fuel studies, see Section VI, also provided an opportunity for a simple cylindrical geometry high leakage, benchmark, in this case with steel-clad 7% enriched uranium oxide pins.

The experiments in these assemblies included diagnostic reaction-rate measurements by the Reactor Physics Division Counting Laboratory, using well-established techniques (5). A selection of the results and associated uncertainties (1 standard deviation) are compared with prediction in Table 2, which includes, for completeness, data from Assembly S02. The most striking trend is that the WIMS data consistently underestimate the capture rate in U238 to U235 fission, which is measured relative to that in a thermal spectrum. This trend has been corrected in a recent revision of the data (6). A comparison of the calculated neutron balances suggests that the under-estimate is even larger in the case of the MONK6.3 calculations, which may explain the relatively large k-values obtained for Assemblies S01 and S03. There is also some evidence that the U238 fission rate relative to that in U235 is poorly predicted by WIMS, although the impact on the k-value in this case is small.

Future plans envisage extending the range of simple geometry assemblies to include mixed-oxide fuel.

Table 1 - DIMPLE Assemblies, 1983 to 1986

Assembly	Pitch (mm)	Number of Fuel Pins		Critical Water Height (cm)	Description of Measurement
		3% enriched	7% enriched		
S01	13.2	1565	-	49.4	Cylindrical, high-leakage benchmark; U235, U238 fission rate and U238 capture rate distributions and ratios.
S02 Critical	17.9	3920	-	50.3	Critical loading of 20 compartment, boron-steel wall, CAGR skip; benchmark measurements as in S01.
S02 Sub-Critical	12.7 to 25.3	1040 to 4418	-	Fully flooded	Sub-critical measurements in CAGR skip; 13 sub-critical arrays.
S03	13.2	-	376	53.8	Cylindrical, high-leakage benchmark; benchmark measurements as in S01; sub-critical measurements with reduced pin loadings; irradiated fuel assembly (resonance region accentuated).
S04/1	13.2	196	680	57.8	Annular core with central light water region; irradiated fuel assembly (residual fissile mass calibration); U235 fission rate distribution.
S04/2	13.2	424	680	56.7	Annular core with central heavy water region; irradiated fuel assembly (thermal region accentuated); U235 fission rate distribution.
S05/1	12.7 to 17.9	3268	248	52.0	Simulated loading error in central compartment of CAGR skip; U235, U238 fission rate and U238 capture rate distributions.
S05/2	12.7 to 17.9	3496	248	52.8	Simulated loading error in edge compartment of CAGR skip; reaction-rate distributions as S05/1.

IV SUB-CRITICAL EXPERIMENTS

There is a wide range of data for critical experiments but very few sub-critical benchmarks. To meet this need, a unique series of measurements has been completed in DIMPLE, with 3% enriched uranium oxide fuel pins in a 20 compartment, boron-steel walled, skip. This is used inside a steel container for pond storage of irradiated fuel from the Advanced Gas-Cooled Reactors (CAGR's) and this in turn is placed inside a lead-lined, steel flask for fuel transport.

The arrangement of the skip in DIMPLE is shown in Figure 2. A total of 13 sub-critical benchmarks have been studied, with k-effective ranging from 0.9 down to 0.7. Early measurements concentrated on the importance of fuel location and included a number of arrays for BNFL (7). Later measurements have included an investigation of the influence of voiding and a very

Table 2 - Comparison of DIMPLE Experiments with Prediction

Assembly	Calculated k-value		Reaction-rates relative to U235 fission					
			U238 Capture (lattice to thermal)			U238 fission		
	LWRWIMS	MONK6.3	LWRWIMS	Expt	C/E	LWRWIMS	Expt	C/E
S01	1.001	1.022 ±0.003	4.175	4.284	0.975 ±0.004	0.002848	0.003038	0.938 ±0.033
S02	1.002	1.003 ±0.004	2.396	2.426	0.988 ±0.006	0.001404	0.001459	0.962 ±0.033
S03	1.004	1.012 ±0.003	3.959	4.031	0.982 ±0.009	0.002867	0.002861	1.002 ±0.033

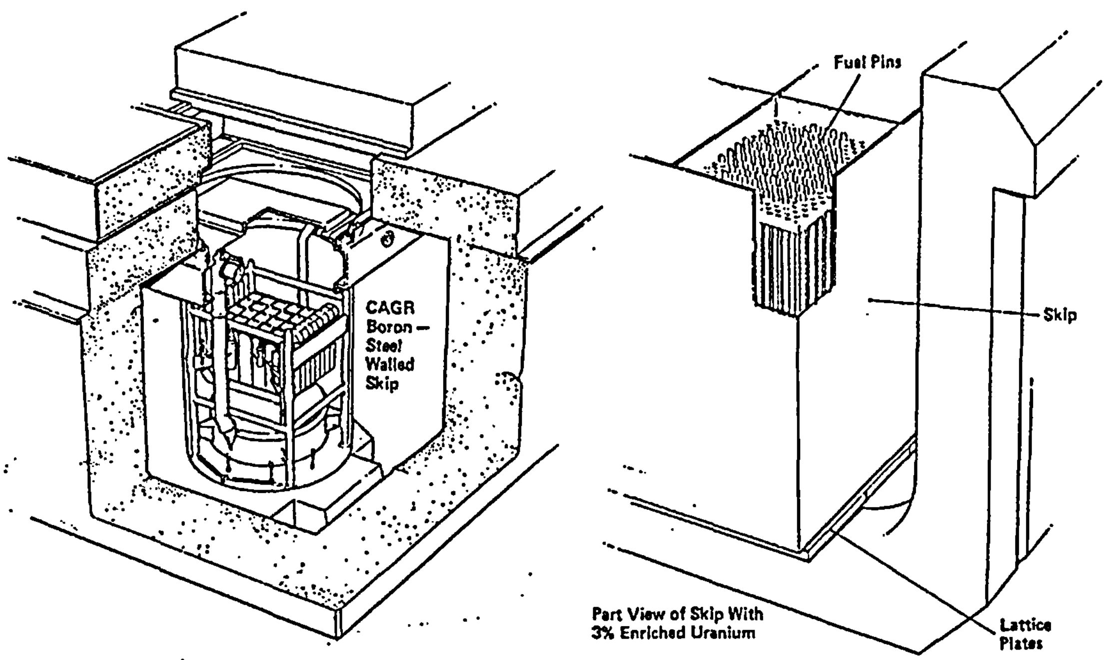

Figure 2 - Layout of CAGR Skip in DIMPLE

sub-critical loading, which has been used as a "blind benchmark", to test the consistency of criticality calculation methods throughout the UK nuclear industry. Details of this work are summarised in a companion paper (8), where it can be seen that, in general, current calculation methods provide satisfactory k-value predictions for these sub-critical assemblies.

One of the aims of the DIMPLE programme is the development of techniques to monitor sub-criticality. The standard method used throughout the CAGR skip work is the Modified Source Multiplication technique (MSM) (9). Here the response of BF_3 counters in or around an assembly to a neutron source placed at the centre are compared with the corresponding values in a state where the sub-criticality is known. Source-mode calculations are used to relate the source and detector efficiencies in the two cases. The feasibility of adapting this approach to a plant environment is under investigation.

A second technique, based on both cross- and auto-correlation noise analyses has been investigated in Assembly S03, the number of 7% enriched fuel pins present being steadily reduced to increase sub-criticality. k-values were deduced using calculated prompt neutron lifetimes and compared with MSM measurements and LWRWIMS-TWOTRAN predictions. The results are summarised in Table 3. It can be seen that the k-values from the noise analysis tend to be higher than those from calculation and MSM, both of which relate to k-values from fundamental mode calculations. The difference between the noise results and prediction is reduced when the k-values are derived from neutron balances of the more appropriate source-mode calculations.

V ACCIDENT SIMULATION

An important feature of the DIMPLE programme is the ability to study sub-critical and critical arrays of fuel, allowing both normal conditions and extreme accidents to be represented.

An example of accident simulation is provided by recent measurements in Assembly S05. These simulated a loading error in a transport flask, a high enrichment cluster being introduced into one of the compartments of the CAGR skip, the others being loaded with low enrichment fuel. The experiments, which were designed in collaboration with the CEA at Fontenay-aux-Roses, received financial support from the Commission of the European Communities (CEC).

As already seen in Section 3, important diagnostic data are provided by measurements of fission rates and neutron capture rates in the critical assemblies. The very large tilts in power distribution measured in Assembly S05 are shown in Figure 3. The results obtained from this work are summarised in a companion paper (8) and demonstrate that MONK6, the corresponding French Monte Carlo code, MORET, and deterministic LWRWIMS calculations provide satisfactory predictions of k-values under

Table 3
k-value Comparisons for Cylindrical 7% Enriched Pin Assemblies

Number of Pins in Assembly	k-values			
	Fundamental mode Calculation	MSM	Cross- and Auto-Correlation	Source Mode Calculation
386 (Critical)	1.003	-	-	-
318	0.958	-	0.960±0.002	0.963
258	0.911	0.915±0.007	0.919±0.003	0.927
210	0.863	0.866±0.010	0.901±0.007	0.885
166	0.809	0.819±0.013	0.869±0.014	0.833
134	0.760	0.790±0.015	0.815±0.024	0.788

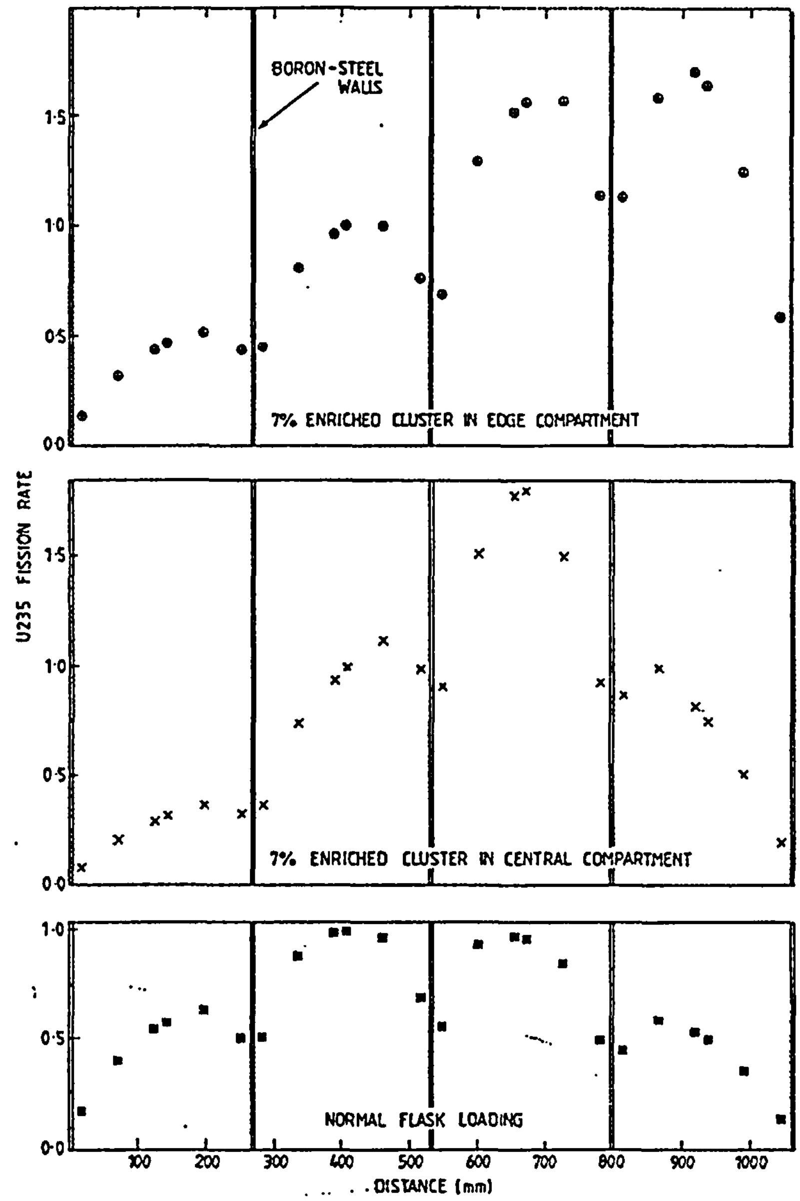

Figure 3 – Radial Power Distributions in CAGR Skip

these extreme conditions. However, it is noted there that the prediction of the most extreme tilt in power distribution is relatively poor, suggesting that there are nonetheless deficiencies in the calculations.

VI IRRADIATED FUEL MEASUREMENTS

In response to a request from the nuclear industry, DIMPLE has recently developed a capability for handling and making measurements with gram quantities of irradiated fuel. The measurements compare the reactivity of an irradiated sample with a range of fissile and neutron-absorbing standards at the core centre and lead to estimates of the residual fissile content of the sample and of the neutron absorption in the fission products.

The technique is based on one developed with the low-power HECTOR reactor at Winfrith. A provisional analysis of HECTOR data is illustrated in Figure 4. This shows WIMS predictions compared with measurements with irradiated fuel from the High Temperature Gas-Cooled Reactor, DRAGON (10). The deviation of the irradiated sample results from the fissile and absorbing standards in this case corresponds to calculation overestimating the fission product absorption by (24±8)%.

In DIMPLE, Assembly S03 is designed to accentuate fission product absorption in the resonance region and in a version of Assembly S04 with a central tank filled with heavy water, events in the thermal region predominate. A second version of Assembly S04, with a central light water zone is used to establish the fissile content of the irradiated samples.

Experiments with irradiated fuel samples from a range of power reactor types are scheduled to begin in DIMPLE later this year.

The irradiated fuel measurements have been designed primarily to check predictions of burn-up and fission product absorption in power reactors. However, the work has a much broader context, as the need to take account of burn-up in criticality assessments is becoming of increasing importance as higher enrichment fuels are introduced in power reactors.

VII CONCLUSIONS

A wide-ranging experimental programme has been completed in the first three years of criticality-related studies in the DIMPLE reactor.

The experimental data obtained, while generally giving confidence in the methods currently used in the UK for criticality assessments, have indicated a number of areas where improvements are required, if higher levels of confidence are to be achieved.

A capability for monitoring sub-criticality has been developed and future plans envisage extending the scope of the studies and also adapting the techniques for plant environments. The sub-critical monitoring programme, that associated with irradiated fuel and proposed measurements with mixed-oxide fuel are all relevant to the impending need to take account of burn-up in criticality assessments.

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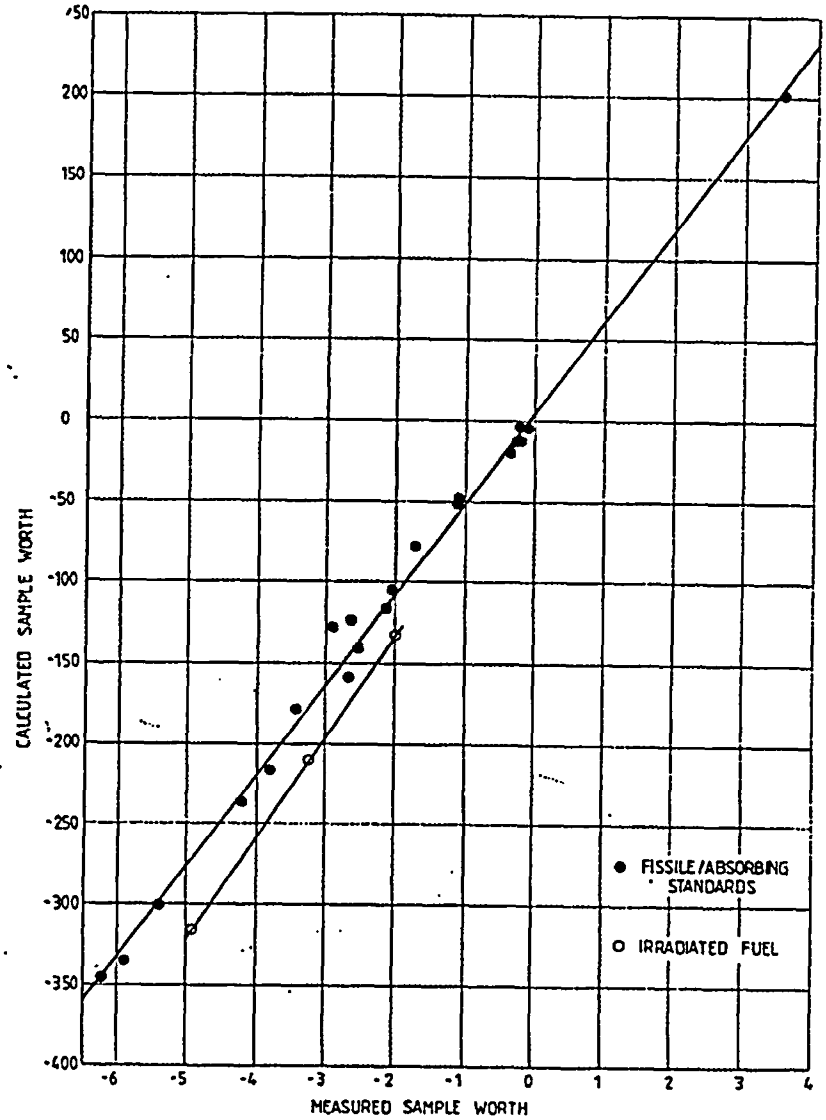

Figure 4 - Reactivity Correlation in HECTOR