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DEVELOPMENT OF SUBCRITICAL MONITORING TECHNIQUES USING THE DIMPLE REACTOR

G INGRAM, A D KNIPE, B M FRANKLIN
Winfrith Technology Centre, UKAEA, Dorchester, Dorset
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ABSTRACT

The application and development of sub-critical monitoring forms an important part of the criticality programme in the DIMPLE reactor. Effort has been concentrated on the Modified Source Multiplication (MSM) technique , which has been tested rigorously in both simple and complex geometries. Encouraging results have stimulated investigations into how the technique might be broadened to apply generally in the nuclear field. A suitable route has been identified provided the neutron source strengths present in the fuel are sufficiently well-known. Current work in DIMPLE with irradiated fuel samples indicates that this stage has already been reached and that monitoring of k-values in fuel stores and reprocessing plants using MSM is achievable with Winfrith methods and data.

I INTRODUCTION

Measurements of sub-criticality have formed an important part of the reactor physics programme in low power reactors at Winfrith. The techniques applied include, for example, buckling measurements in source-driven exponential stacks[1], pulsed neutron source measurements[2], the Modified Source Multiplication (MSM)[2] technique and more recently neutron noise measurements[3].

Sub-critical monitoring is central to several studies in the low power, water-moderated reactor DIMPLE at Winfrith. The prime objective of the experimental programme is to validate the methods and data used in both criticality assessments and power reactor performance predictions. However, the programme is also aimed at developing sub-critical monitoring techniques that could be applied in a plant environment. To this end, the DIMPLE experiments have tended to concentrate on the simplest of the methods and the one most capable of wide application in the nuclear field, ie MSM. However, within the constraints of the reactor programme, time has also been allocated to noise measurements. These have been performed mainly because of their reactor physics interest, but also because it could be argued that noise analysis has yet to reach its expected potential.

Following a brief description of the DIMPLE reactor in the next Section, the sub-critical measurement programme to-date is reviewed below in Sections III and IV and an outline of the likely trends in future developments is given in Section V.

II DIMPLE

Briefly, DIMPLE is a versatile, low-power, water-moderated reactor, accommodating arrays of pin fuel in a large aluminium tank (4m high and 2.6m diameter). The layout of the plant is shown in Figure 1. In the conventional assemblies illustrated, the fuel pins are supported and precisely located by aluminium lattice plates. These are attached to U-shaped aluminium beams, which in turn are supported by a tubular steel chassis.

The standard fuel pins used in DIMPLE consist of sintered 3% (10.1mm diameter) and 7% (7.4mm diameter) enriched uranium oxide pellets. These are stacked in thin walled stainless-steel tubes to a fuel height of nominally 700mm. Reactor control is by water level alone, with shut-down by means of a fast dump system.

A more detailed description of the plant is given in Reference 4.

III THE MSM TECHNIQUE

The MSM technique applied in DIMPLE was developed in the low-power fast reactor ZEBRA at Winfrith. It was used there to measure shut-down margins in typical fast reactor cores and, on the basis of this earlier work, the technique is now used routinely during refuelling of the Prototype Fast Reactor at Dounreay.

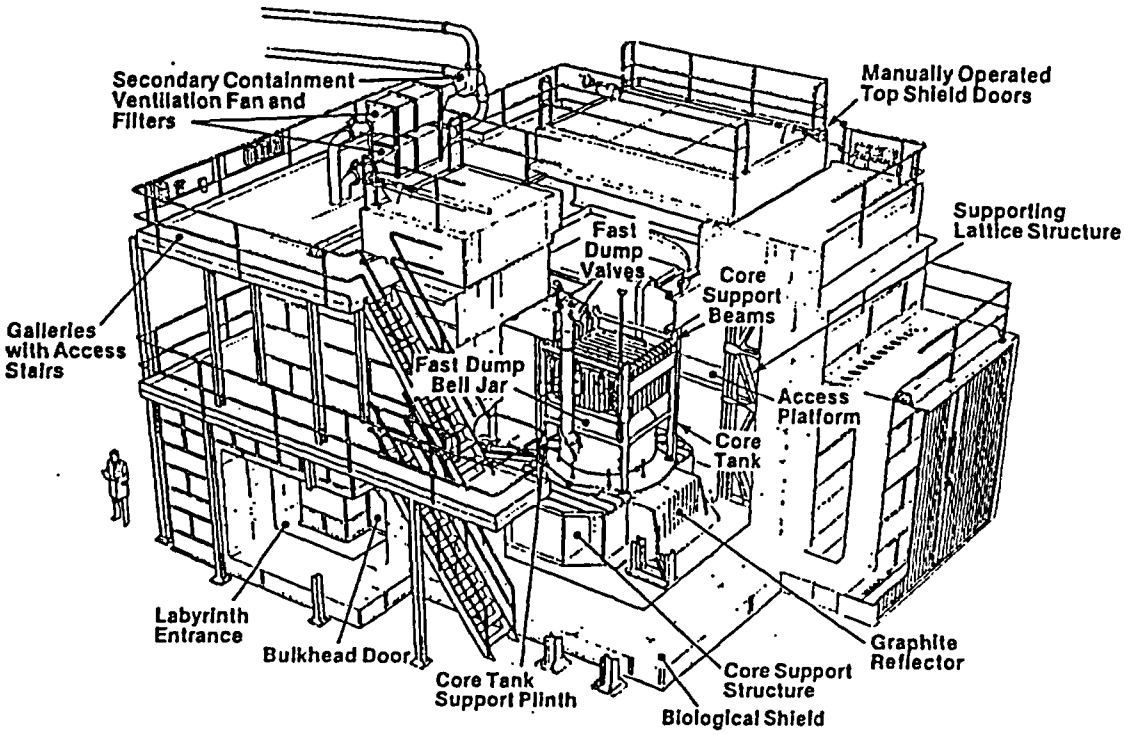


Figure 1 - General View of DIMPLE Reactor

In DIMPLE, neutron sources are placed within a sub-critical assembly and the resultant power level is monitored with neutron detectors placed in and around the assembly. In a sub-critical system close to critical, the relationship between the sub-criticality, the fission source, ie the power level, P, and the neutron source strength, S, can be expressed:

$$\rho = \frac{1}{k} - 1 = \frac{S}{P\nu} \qquad (1)$$

where ρ is the sub-critical reactivity, k is the k-value of the system, ν is the average number of neutrons per fission. For simplicity it has been assumed that the neutrons from the source and fission have the same spatial distribution and energies. As the assembly becomes more sub-critical, and the power deviates from the eigen-mode distribution, it becomes necessary to take account of changes in the efficiency of the detector used to monitor the power level and changes in the fission source distribution relative to the neutron source. Thus, as the assembly becomes more sub-critical, the product of the reactivity and detector count-rate are no longer constant. In comparing two sub-critical states this departure can be obtained from calculation as:

$$F = \left[\frac{\rho_2 R_2}{\rho_1 R_1}\right]_{calc} \qquad (2)$$

where the reaction rates, R, are from source mode calculations and, to make the results consistent with normal calculation methods, the reactivities are from eigen-value calculations.

Using measured count rates, C, the unknown sub-criticality of a system can be obtained relative to a known one:

$$\left[\frac{\rho_2}{\rho_1}\right]_{expt} = F \left[\frac{C_1}{C_2}\right]_{expt} \qquad (3)$$

The first measurements in DIMPLE were with 3% enriched uranium oxide fuel pins in a 20 compartment, boron-steel walled skip, approximately 1.4m long, 1m wide and 0.9m high. The

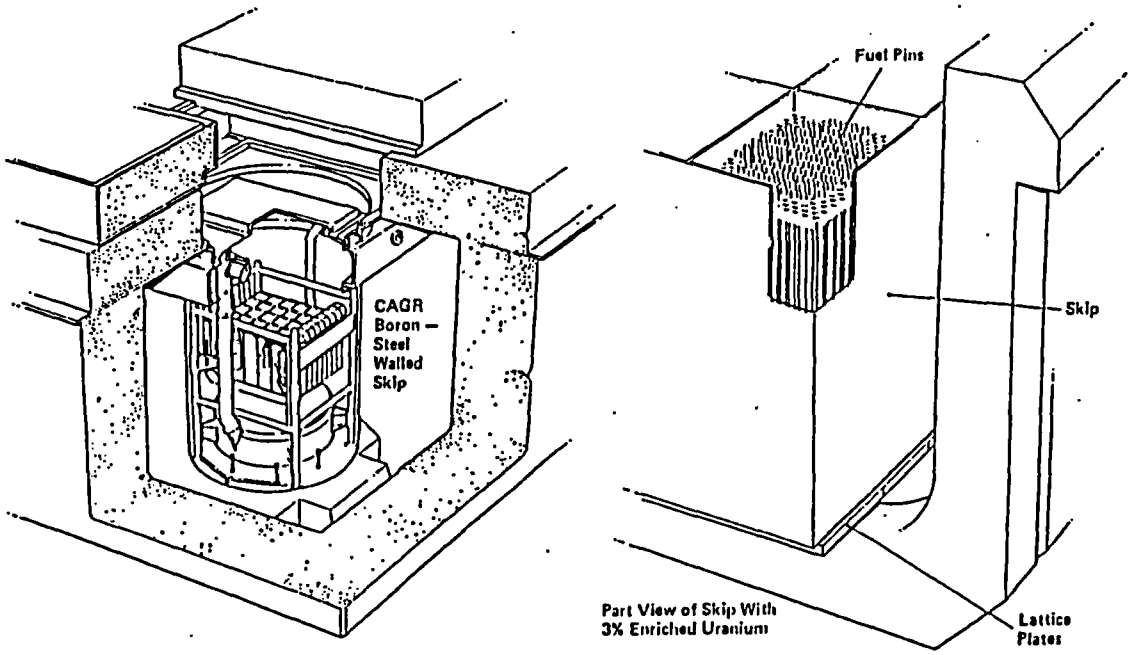


Figure 2 – Layout of CAGR Skip in DIMPLE

boron-steel walls were 5mm thick and contained 1 w/o of boron. This type of skip is used for the transport and storage of fuel from the UK Commercial Advanced Gas-Cooled Reactors (CAGR's). The arrangement in DIMPLE is shown in Figure 2. The majority of the sub-critical experiments are described in Reference 5, with two further assemblies being included in Reference 6. For completeness a summary of all the sub-critical results is given in Table 1.

In the reference configuration, with 196 pins spread in a square array across each of the compartments, the skip was critical at a water height of about 50cm. For the sub-critical measurements the skip was fully flooded with light water and the pin patterns changed, firstly, to vary the reactivity of the components within each compartment, secondly, to vary the interaction between compartments and, finally, to represent potential accident scenarios where pins were dropped into gaps between the fuel clusters. The 52, 112 and 221 pin clusters, which all had an outer diameter of 215mm, are examples of changing the reactivity in each compartment, with loadings going from over-moderated, through optimum moderation to under-moderated respectively. The 137 pin cluster and the 172 and 184 pin clusters are further examples, where optimum moderation was maintained and the cluster diameter gradually increased to nearly fill each compartment. For the two very large clusters it was necessary to lower the water level to 49cm to keep the assemblies sub-critical. The interaction between compartments was varied by moving the clusters towards the compartment walls, firstly in groups of four and then with all the clusters as close as possible to the centre of the skip. The interaction was also varied by increasing the fuel loading at the edge of the fuel clusters to give the 152 pin cluster and by introducing simulated voids, in the form of aluminium plates, in the water gaps between clusters.

Because the complex geometries of the skip provided a severe test of the MSM technique, a diagnostic approach was adopted in the experimental programme. This established the sensitivity of the results to source and detector location and to the methods used to calculate the corrections in equation 2. The conclusions from these studies have been given previously.[7] Briefly, diffusion theory was found to provide satisfactory results when the detectors were located in fuel regions, but, not surprisingly, gave poor predictions of fine-structure in the water/boron-steel regions of the skip compartments and poor predictions of the attenuation

Table 1

Comparison of Measured k-values for the Subcritical Skip Assemblies with Prediction

Array	Measured k-value	Predicted k-value	
		LWRWIMS-TWOTRAN	MONK6.4 ($1\sigma = 0.006$)
112-pin clusters, centred	0.843±0.014	0.844	0.841
112-pin clusters, groups of 4	0.863±0.011	0.861	0.866
112-pin clusters, closest to skip centre	0.864±0.011	0.867	0.865
112-pin clusters, closest to skip centre +92 dropped pins	0.913±0.006	0.923	0.927
221-pin clusters, centred	0.778±0.020	0.778	0.778
221-pin clusters, closest to skip centre	0.801±0.015	0.801	0.823
221-pin clusters, closest to skip centre, edge pins removed	0.804±0.015	0.798	0.809
221-pin clusters, closest to skip centre, edge pins removed +92 dropped pins	0.843±0.011	0.855	0.871
221-pin clusters, closest to skip centre, edge pins removed +208 dropped pins	0.867±0.008	0.890	0.894
152-pin clusters, centred	0.865±0.011	0.875	0.854
137-pin clusters, centred	0.899±0.007	0.916	0.915
52-pin clusters, centred	0.707±0.015	0.720	-
112-pin clusters, centred +34 aluminium plates	0.866±0.007	0.877	-
184-pin clusters, centred	0.970±0.002	0.973	
172-pin clusters, centred	0.946±0.003	0.953	

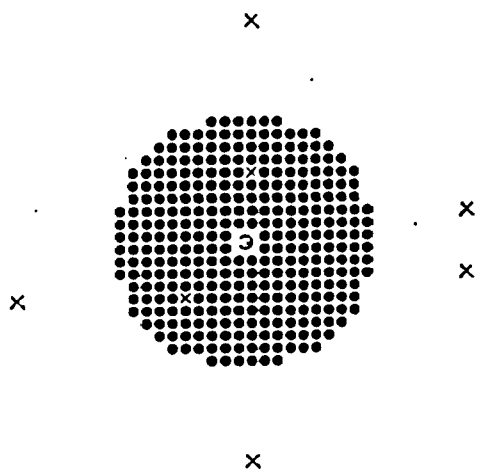
of neutrons through the skip. As a result, with k-values of about 0.8, detectors located outside the skip were tending to overestimate the k-value by about 3%. Transport theory on the other hand gave satisfactory predictions of k-value irrespective of source and detector location.

The k-values for the skip configurations studied ranged from 0.71 to 0.97 and the uncertainties in the measurements of sub-criticality were typically about ±7%, where ±5% came from the calibration of the reference reactivity by period measurements and the remainder were from a combination of counting statistics, neutron detector reproducibility and the influence of asymmetries within the skip.

The main purpose of this series of measurements was to validate the deterministic WIMS code and the MONK Monte Carlo code. The mean values of (C-E) from Table 1 and the deviation of the individual results about this mean are 0.007±0.008 for WIMS and 0.009±0.0013 for MONK, indicating agreement within the accuracy of the comparison. It should be noted that these conclusions are on the basis of early versions of the WIMS and MONK codes. A comparison is in progress using the current versions[8], although this is unlikely to change the overall conclusion that the WIMS and MONK codes provide satisfactory predictions for the complex skip geometries.

The MSM technique has also been applied to simple geometries in DIMPLE. The measurements

were made in a cylindrical loading of 7% enriched uranium oxide pins on a square pitch of 13.2mm. The critical loading, with a central guide tube to accommodate a neutron source, was 388 pins and a water height of about 50cm.



o Source Location
x Detector Location

Figure 3

For the sub-critical measurements, a single Cf252 source was placed at the core centre and counts were taken on two in-core BF_3 detectors and five larger BF_3 detectors located in the water reflector. Figure 3 shows the arrangement for a 318 pin loading. The results for a progressive reduction in pin loading with a fixed water height are shown in Table 2, which is reproduced from Reference 5. Diffusion theory was used throughout to obtain the detector correction factors and the results quoted are the mean from all seven detectors. Also shown in the table are the results of neutron noise measurements, which are discussed in the next section.

IV NOISE MEASUREMENTS

Although MSM has been used predominantly in DIMPLE to measure sub-criticality, other techniques have been employed from time to time to provide a cross-check. Chief among these has been noise measurements. To gauge potential problems in this method, the approach has been relatively simple and based on the cross-correlation technique.

In a source-driven sub-critical assembly, individual fission chains are started by source neutrons and subsequently die away. This leads to correlated events which can be observed by studying the relative responses of two neutron detectors in or around the assembly. It can be shown that the variation of the sum of the product of the count, C_1, in a given time interval for one detector and the count, C_2 in a second detector for a similar interval but delayed by time Δt, satisfies an equation of the type:

Table 2

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

$$\sum_t C_1(t)\, C_2(t+n\Delta t) = A + Be^{-\alpha n \Delta t}$$

where n is an integer and α is normally taken as the prompt neutron decay constant. This is given by:

$$\alpha = \frac{1-k_p}{l^*}$$

where k_p is the prompt neutron multiplication and l^* the prompt neutron lifetime. A similar expression applies for auto-correlation where the counts are multiplied by delayed counts from the same detector.

For the DIMPLE measurements, a PDP11 computer used to collect and store the counts from pairs of detectors in a time interval of 10^{-4} seconds. The results were subsequently analysed, again using the PDP11, to extract the exponential and thus α.

The method was first tried in the CAGR skip loading with 112 pins at the centre of each compartment. Here it was found that because of the neutron attenuation in the water gaps and boron steel across the skip, it was only possible to detect cross-correlation between adjacent compartments. This casts doubt on the ability of the technique to identify a relatively high reactivity in a compartment far removed from the detector. The analysis led to a tentative k-value of 0.82±0.04, which is in reasonable agreement with the 0.84±0.01 from the MSM measurements.

The second set of noise measurements were made in a more favourable geometry, the cylindrical array of 7% enriched uranium oxide pins described in the previous section. The results are given in Table 2, where it can be seen that there is reasonable agreement for the noise results and the MSM results with prediction, provided that it is borne in mind that the MSM results relate to an eigen-value distribution, while the noise results relate to a source mode solution.

It should be noted that the analysis of the noise results is preliminary and based on the conventional treatment, where it is assumed that neutron chains decay exponentially. An alternative approach using the Monte Carlo Code ICBEND[8] has been examined at Winfrith in which a description of the build-up and decay of an individual chain in a multiplying system is correctly represented.[9] This work indicates the creation of some extremely long chains which have very much greater probabilities of being produced than would be expected from the conventional analysis and lead, for example, to the conventional analysis introducing an error of about 0.05 in a measured k-value of 0.6. Subject to the technical programme priorities, this interesting new approach will be adopted in the analysis of the DIMPLE measurements.

V FUTURE DEVELOPMENTS

A limitation of the current MSM technique is the need to calibrate against a reference configuration, where the sub-criticality is known precisely. While this does not pose a problem in DIMPLE measurements, it precludes the use of the technique in a plant environment. To broaden the application of the method requires a knowledge of the neutron source strength and a code that could apply this source in a source-mode calculation to predict detector count-rates as a function of k-value. With this information, measurement of the count-rates with absolutely calibrated neutron detectors would yield the k-value.

As shown in the previous Sections, both the WIMS transport theory code and the MONK Monte Carlo code show every indication of providing satisfactory modelling of complex geometries. The validity of neutron source predictions for irradiated fuel are currently under investigation in DIMPLE.[10] Briefly, the measurements are made with gram quantities of irradiated fuel, where the burn-up and cooling-times are well defined. The technique is again based on equation 1. The samples are inserted in the centre of a DIMPLE assembly and the count-rates measured on a series of neutron detectors as a function of water height, which close to critical is proportional to the reactivity of the system. A plot of inverse counting rate against water height yields a straight line with a gradient inversely proportional to the neutron source strength. The results are put on an absolute basis by repeating the measurements with an absolutely calibrated Cf252 source. Small corrections are necessary to allow for the background U238 spontaneous fission source in the assembly and for small differences in worth and spatial distribution of the neutrons from the Cf252 source and the irradiated sample.

Measurements have been completed with four samples of CAGR fuel irradiated to about 20GWD/t and currently PWR samples irradiated from 20GWD/t to 50GWD/t are being studied.

A preliminary analysis of the CAGR results was included in Reference 9, along with a comparison of measured and predicted neutron sources in plutonium fuels. A more recent analysis of the results using the UK burn-up code FISPIN with the appropriate CAGR burn-up dependent libraries is summarised below in Table 3. It can be seen agreement between prediction and experiment is surprisingly good, bearing in mind the complex chain of events leading to the production of the neutron

Table 3

Comparison of Measured and Predicted Neutron Source Strengths from Irradiated CAGR Samples

Sample	Initial Enrichment (%)	Absolute Neutron Source Strength (ns^{-1})		C/E
		Calculation	Experiment	
A	2.5	11572	10306	1.12±5%
B	2.0	22140	19282	1.15±5%
C	2.5	5123	4845	1.06±5%
D	2.5	7115	6091	1.16±5%

Table 4

Neutron Source Strengths in CAGR Sample 'A'

Isotope	FISPIN Prediction for Predominant Contributors (ns^{-1})	
	Spontaneous Fission	(α,n)
Pu238	20	106
Pu239	-	17
Pu240	293	42
Pu242	76	-
Am241	-	35
Cm242	830	157
Cm244	9900	73
Total FISPIN for all isotopes	11572	
Experiment	10306	

sources. The average C/E value obtained is 1.12. The various contributions to the predicted results for a typical sample is shown in Table 4, which demonstrates that the predominant contributor to the neutron source, Cm244, is the most likely cause of the discrepancy between prediction and experiment.

An error of 12% in predicted neutron source strength corresponds to an error of about 1% to 3% for k-values in the range 0.9 to 0.7 when these are measured by MSM. Thus, with the sophisticated modelling capabilities of the WIMS and MONK codes, it is apparent that the opportunity exists in fuel stores and reprocessing plants to translate neutron count-rates, which may already be used for process or safeguards control, into reasonably accurate k-values.

VI CONCLUSIONS

Sub-critical monitoring based on MSM plays a key role in the DIMPLE criticality programme. To extend the technique to measure k-values in a plant environment requires validated codes and a knowledge of the neutron sources present. The DIMPLE programme of sub-critical monitoring development and neutron source strength validation indicates that this stage has already been reached and k-values in plants and stores could be routinely monitored with MSM and Winfrith methods and data.

The next stage, a demonstration of the absolute MSM technique in DIMPLE followed by its application in a plant environment, is currently being developed as a possible collaborative programme with the nuclear industry.

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