Proceedings of ICEM'03:
The 9th International Conference on Environmental Remediation and Radioactive Waste Management
September 21 – 25, 2003, Examination Schools, Oxford, England

ICEM03-4659

A VERY HIGH EFFICIENCY NEUTRON COUNTER FOR THE MEASUREMENT OF PLUTONIUM IN DECOMMISSIONING WASTES

A.C. Tolchard/A.N. Technology Ltd. **M.R. Looman/Consulenze Tecniche**

J.A. Mason/A.N. Technology Ltd.

ABSTRACT

The design of the ANTECH Model 2203 Very High Efficiency Neutron Counter (V-HENC) is a natural progression from the well-proven ANTECH Series 2200 Passive Neutron Drum Monitor used for measuring plutonium in intermediate and low level waste (LLW) 200 litre drums. ANTECH has had considerable experience in the implementation of the base design, originally licensed to ANTECH from the Joint Research Centre at Ispra, Italy.

Three Series 2200 systems have been supplied by ANTECH: two are in operation at AWE Aldermaston for waste monitoring, and a third is implemented at the SMP facility at BNFL Sellafield. Some 15 years cumulative operating experience has been gained, and ANTECH provides technical support as part of continuing maintenance and support arrangements. The design has proven to be inherently reliable, safe, easy to operate and maintain. Recently, both AWE instruments have been fully characterised and calibrated to function in conventional coincidence counting mode with calibrations.

ANTECH has used MCNP to optimise the design of the fast detector packages in order to achieve the lower detection levels required to measure Pu at USA TRU/LLW and UK Nirex LLW levels. With the aid of simulations a typical detection efficiency of 36% with Cd filters deployed and up to 45% with the internal Cd liner removed has been achieved. Statistical data filtering is used to decrease the cosmic ray induced neutron background, the latter also being minimised by the absence of steelwork within the drum measurement chamber. An outer shielding of 270 mm thickness of polyethylene is used to shield the system from external neutrons.

A total of 16 fast detector modules are used, which consist of eight, 7.5 atmospheres ^{3}He tubes (25.4mm diameter and 1033 mm active length) embedded in high density polyethylene and arranged in a double row. Tubes are connected using HN connectors to a junction box at the top of the vertical modules or on the end of the horizontal modules. The junction boxes are hermetically sealed and contain the high voltage distribution and AMPTEK model A-111 charge sensitive amplifiers.

The operation of the V-HENC is based on passive neutron counting of the correlated neutrons from spontaneous fission of the even Pu nuclides, principally ^{240}Pu and is coupled with an ANTECH/Ortec Advanced Multiplicity Shift Register employing the Los Alamos INCC code. Alternatively, the multiple gate ANTECH Time Correlation Analyser (TCA) may be used for enhanced data acquisition for multiplicity counting. The V-HENC can be operated in conventional shift register coincidence counting (Reals) mode (with a calibration function), the absolute multiplicity counting mode (histogram function) or totals counting mode. Plant measured isotopic ratios can be used by the software to convert $^{240}Pu_{effective}$ mass to total Pu mass.

INTRODUCTION

Passive Neutron Coincidence Counting (PNCC) uses the time correlation of spontaneous fission neutrons to count neutrons that arise from the same fission event. Thus, the coincidence rate is proportional to the number of spontaneous fissions within the sample. Since ^{240}Pu is usually the dominant isotope in coincidence counting, an effective ^{240}Pu mass is calculated that gives the same coincidence response as all of the even isotopes of plutonium. As in all neutron methods, the isotopic ratios must be known in order to calculate the total amount of plutonium. For a given background and moderate counting time, a typical detection sensitivity for a 200 litre waste drum PNCC at ~20% efficiency is ~160 mg of low-burn up plutonium in a 100-kg matrix (i.e. 0.5g/cc). In order to screen waste at the USA TRU/LLW and the UK LLW (4000Bq/g) in moderate density waste, and in the absence of an active neutron measurement system, a very high efficiency neutron detection system (30–40% efficiency), is required.

This paper describes the Model 2200 Passive Neutron drum Monitor and briefly reviews the methods used to determine its characteristics. The Model 2200 has been used for the measurement of decommissioning waste in the UK for a number of years. The Model 2200 design forms the basis for the up-rated Model 2203 Very High Efficiency Neutron Counter which is capable of screening waste at the USA TRU/LLW and UK LLW limit.

PASSIVE NEUTRON COINCIDENCE COUNTING

The even mass Pu nuclides (^{238}Pu, ^{240}Pu & ^{242}Pu) generate correlated neutrons from spontaneous fission events, random (single) neutrons from (α,n) reactions and a further source of correlated neutrons from induced fission events caused by primary neutrons. Passive neutron coincidence counting discriminates the correlated pairs of neutrons arising from the spontaneous fission from random background (single) neutrons from (α,n) reactions, allowing thus the determination of the $^{240}Pu_{effective}$ mass, and, from the known isotopic content, the total Pu mass.

The neutrons emitted by the fission events are detected in ^{3}He filled proportional counter tubes. The tubes are especially sensitive to thermal neutrons, so they are embedded in a polyethylene moderator assembly to slow down fast fission neutrons. The small analogue signals from the proportional counters are first amplified by a charge sensitive amplifier and then converted to short digital (50 ns, TTL) pulses by a discriminator (SCA). Several tubes may be connected to a single amplifier/discriminator. The output pulses from the discriminators are digitally combined by an OR-chain or a more sophisticated digital pulse mixer, like ANTECH's DBMC (De-randomising Buffer Mixer Counter), the latter minimises dead-time losses. The result is a single pulse train, which is then analysed by various instruments.

The instruments measure one or a combination of the following count rates, called Singles (S), Doubles (D) or Triples (T), and which are respectively given by (point model approximation):

$$S = \varepsilon F_s \nu_{S(1)} M (1+\alpha) \qquad (1)$$

$$D = \varepsilon^2 F_S \frac{f}{2} M^2 \left[\nu_{S(2)} + (M-1)(1+\alpha)\frac{\nu_{S(1)}\nu_{I(2)}}{\nu_{I(1)}-1} \right] \qquad (2)$$

$$T = \frac{\varepsilon^3 F_S f^2 M^3}{6} \left[\begin{array}{l} \nu_{S(3)} + \\ (M-1)\dfrac{3\nu_{S(2)}\nu_{I(2)} + \nu_{S(1)}\nu_{I(3)}(1+\alpha)}{\nu_{I(1)}-1} + \\ 3(M-1)^2 \nu_{S(1)}(1+\alpha)\nu_{I(2)}^2 \end{array} \right] \qquad (3)$$

where:

F_S is the spontaneous fission source strength (in fissions per second),

ε is the neutron detection probability (efficiency),

α is the ratio of total neutrons from (α,n) to those from spontaneous fission,

M is the fast fission neutron multiplication factor, and,

$\nu_{S(j)}$ $\nu_{I(j)}$ are, respectively, the *j*-th factorial moments of the ^{240}Pu spontaneous fission distribution (S), and the 2 MeV ^{239}Pu induced fission distribution (I).

The gate fraction *f* is given by:

$$f = e^{-PD/\tau}(1-e^{-G/\tau}) \qquad (4)$$

where:

PD is the pre-delay (s),

τ is the neutron die-away time (s), and,

G is the shift register gate width (s).

In conventional passive neutron coincidence counting [1], the pulse train is input to a Shift Register Analyser, which counts the multiple neutron events during the characteristic neutron die-away time of the detector assembly when there is a high probability of detecting coincident events. Two experimentally measured quantities are obtained: the total (uncorrelated) neutron count rate (or Singles) and the coincidence count rate (Reals or Doubles). When we regard equations (1) and (2), we conclude that there are 3 sample dependent unknowns: the multiplication, *M*, the α-ratio and the spontaneous fission source strength F_S. The other parameters are either nuclear constants ($\nu_{S(j)}$ and $\nu_{I(j)}$), or determined by calibration (like ε and τ), or characteristic of the system (*PD* and *G*).

In order to solve the set of equations to determine the spontaneous fission rate and hence the $^{240}Pu_{effective}$ mass, i.e. the specific spontaneous fission rate of ^{240}Pu is equal to 479 fissions per second per gram, we need to know the value of one of the unknowns: the multiplication, *M*, or the α-ratio. The value of the α-ratio can be calculated if the isotopic composition of the sample is known and there is some measure of the major (α, n) generating light elements present (i.e. that it is from plutonium oxide only). If it is impossible to determine the isotopic composition, one can assume for small quantities of Pu that the multiplication is unity (M=1.0). For significant quantities of Pu, the conventional passive neutron coincidence counting method requires corrections to be made due to induced fission within the sample. The shift register is unable to distinguish spontaneous fission signature from that due to induced fission. The effect of multiplication is to enhance the coincidence count rate and the resulting quantity of Pu recorded is biased high.

To overcome the problems described above, equation (3) is also used in more advanced techniques called multiplicity counting and time correlation analysis. A third experimental parameter is recorded, namely the triples rate. These techniques measure the numbers of signal multiplets (the signal frequency distribution) in the detected neutron pulse train. When one of the variables is known (or can be inferred), the data analysis can employ the point source model developed by Dr. W. Hage, see [2] to [4] for a detailed description of the technique, to determine the number of neutron totals, the number of correlated neutron pairs and the number of correlated neutron triples in a pulse train. From the factorial moments of the neutron probability distribution it is possible to perform a four parameter analysis based on the frequency distribution to solve for the spontaneous fission rate, F_S (and hence the $^{240}Pu_{effective}$ mass), the detection probability (ε), the α-ratio or the neutron multiplication, M, as appropriate.

DESCRIPTION OF THE MODEL 2200 INSTRUMENT

The ANTECH Model 2200 Passive Neutron Drum Monitor is of modular construction comprising of a decagon shaped measurement chamber for the neutron counting of 200 litre waste drums. The decagon shaped measurement chamber approximates the cylindrical geometry of the drum thus eliminating the need to rotate the drum during measurement. The 16 polyethylene moderated ^3He detector modules surround the drum. There are 10 vertically placed modules constituting the decagon with three modules above and three modules below the drum. Each detector module is lined with a removable aluminium clad cadmium filter for fast neutron measurement. The detector packages are housed inside an outer shielding of 210 mm thickness of polyethylene. The ^3He tubes are connected to a junction box at the top or end of each module which contains the high voltage distribution circuitry and the AMPTEK A-111 charge sensitive amplifiers.

The measurement chamber is constructed in such a way as to minimise or eliminate the need for high Z materials of construction. The design is suitable for either conventional coincidence counting or multiplicity counting when used with the ANTECH/AMETEK Advanced Multiplicity Shift Register (AMSR-150) or the ANTECH 1000 series Time Correlation Analyser (TCA).

The basic design constitutes a platform for selecting the detection efficiency depending on the configuration of ^3He tubes in the fast neutron detector modules. The well proven Model 2200 contains 4 x 4 atmosphere 25.4mm x 1.0 metre ^3He tubes per module (64 in total) and has a detection efficiency for Cf neutrons of ~20%. It has been in service at establishment such as AWE for many years.

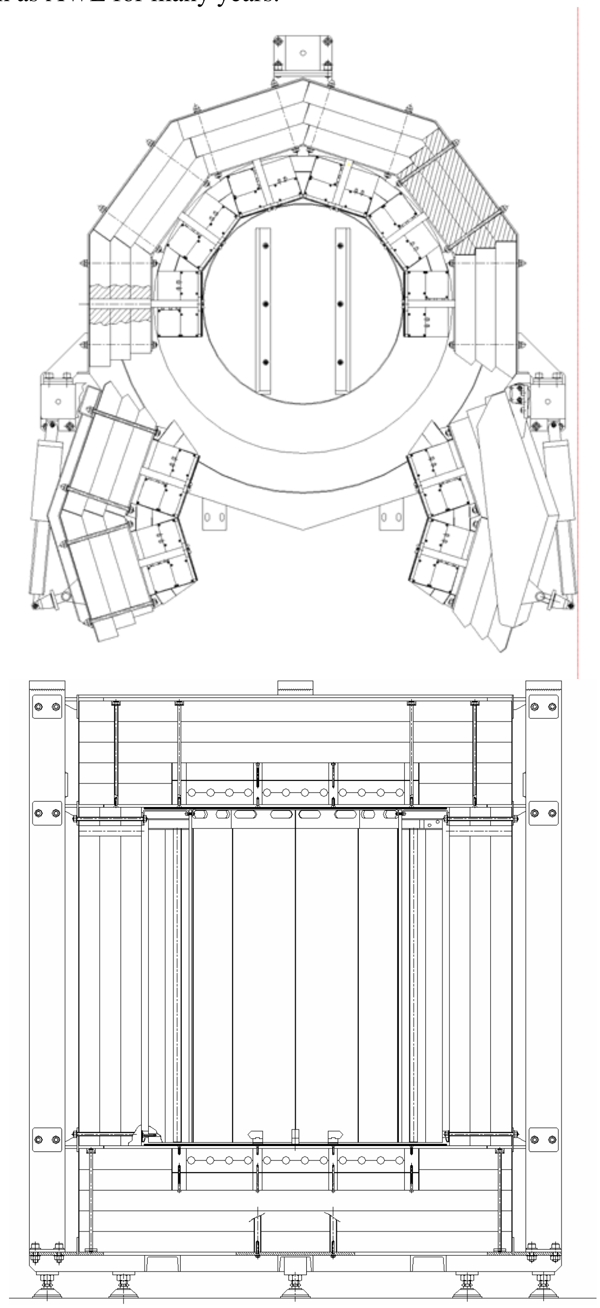

Fig. 1. Model 2200 in Plan and Side View

CHARACTERISATION OF THE MODEL 2200

This section briefly reviews the extensive work carried out in characterising the Model 2200 Passive Neutron Drum Monitor.

Amplifier/Discriminator Adjustment

The A-111 is a hybrid charge-sensitive pre-amplifier, discriminator and pulse shaper developed by AMPTEK. The pre-amplifier gain on each junction box was set to match the other units. Moreover, gamma ray pulse pile up was addressed by measuring a neutron source with the system and re-measuring following introduction of a gamma ray emitting source. The gain was adjusted so that the (fixed) discriminator level corresponded to the valley between the γ-ray and neutron portions of the neutron pulse-height spectrum. This resulted in the same count reading in both cases.

High Voltage Bias Curve

The optimum operating voltage for the ^{3}He proportional counters was determined using a small ^{252}Cf source placed at the centre of the measurement chamber. The automated Bias Curve Routine in the TCA software sets the High Voltage and after a required settling time it starts a Totals count rate measurement. The procedure is repeated at a series of High Voltages. The count rate was plotted as a function of detector voltage to identify the plateau, see Fig. 2. An operating voltage of 1650 V was selected, suitably above the knee of the high voltage plateau to allow for slight variations in the voltage.

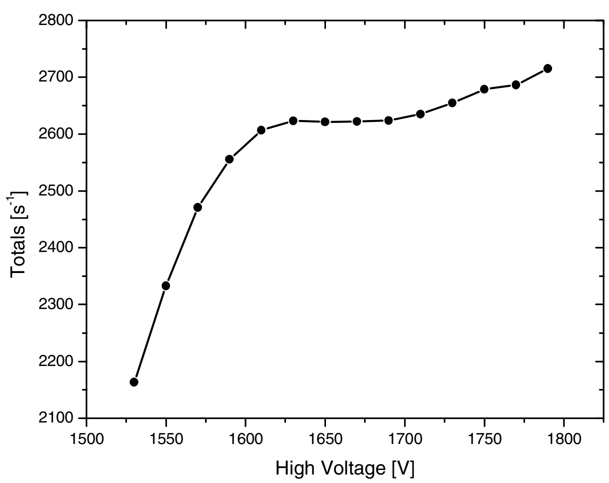

Fig. 2. Typical High Voltage Bias Curve for Model 2200.

Efficiency

The neutron detection efficiency of the Model 2200 was measured using a calibrated ^{252}Cf spontaneous fission source. Table I gives the results for operation modes with and without Cd liner on the surface of the measurement chamber.

Table I. Measurement results for the detection efficiency of a ^{252}Cf spontaneous fission source for two configurations. In the first configuration the source is placed inside the empty chamber. In the second configuration the source is placed inside a waste drum with an internal 7mm thick polyethylene liner.

Configuration	With Cd liner	No Cd liner
Empty chamber	19.4 %	22.5 %
In waste drum	18.6 %	19.5 %

Thermal Neutron Die-Away Time

The thermal neutron die-away time of the detector system was measured using a small ^{252}Cf source in the measurement chamber and measuring the neutron coincidence rate as the gate width was incremented. A special measurement routine for the TCA acquires the Reals count rates at gate widths of 25.6 µs to 409.6µs, in 16 multiple steps of 25.6 µs. The sequence was fully automatic. The results of the measurements are shown in Fig. 3, together with the results of a fit with a single exponential decay function.

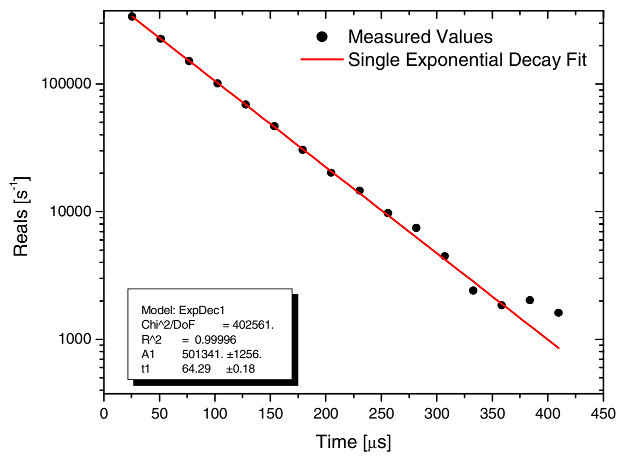

Fig. 3. Typical Die-away Curve for Model 2200

The determination was repeated with the ^{252}Cf source positioned in the centre of a 200 litre drum of various waste matrices. The detector design is such that the die-away time is largely unaffected by the presence of the waste matrix. A maximum variation of ~3% in the measured overall die-away time, τ, of ~64 µs was observed when measured in variety of moderating and non-moderating matrices.

Gate Width

An optimum gate setting on the AMSR of 1.2 times the thermal neutron die–away time is normally used. A die-away time 64 µs results thus in a gate width, G, of 76.8 µs.

Pre-Delay

During the dead time following the trigger pulse, no subsequent pulse can be counted and therefore this shortens the prompt signal gate by an interval equal to the dead time; the net result being a negative bias on the real coincidence count rate. This effect is eliminated by starting the prompt gate after a fixed pre-delay period.

The amount of pre delay to produce equality between prompt and delayed gates can be checked by varying the pre-delay while measuring the totals count rate from a strong, pure random neutron source (e.g. Am/B or Am/Li). A *PD* value of 4 μs proved to be sufficient.

Gate Fraction

When we insert the above found values of *PD*, *G* and τ in equation (4) we obtain the following result for the gate fraction, $f = 0.656$.

Dead time

Two calibrated ^{252}Cf sources were used to measure the system dead time using the 'super-positioning' of sources method. Two adjacent positions were defined at the centre of the Model 2200 measurement chamber. The first ^{252}Cf source was measured alone and the background subtracted Totals count was determined for a fixed counting cycle. The second ^{252}Cf source was placed, without disturbing the first source, next to the first source and the counts acquired for the two sources together under the exact same conditions. The first source was then removed without disturbing the second source and counts acquired for the second source under the exact same conditions.

The Totals dead time for the Model 2200 was calculated as 40.77 ns. The Reals dead time was calculated as 1.064 μs. In the TCA, the analytic dead time only requires the input of the Totals dead time as calculated above.

For dead time correction using the AMSR and INCC-B32 software, the following equations apply for the corrected Reals, R_c, and Totals, T_c:

$$R_c = R_m e^{aT_m + bT_m^2} \tag{5}$$

$$T_c = T_m e^{\frac{aT_m + bT_m^2}{4}} \tag{6}$$

where a and b are empirical dead time coefficients, and R_m and T_m are, respectively, the measured Reals and Totals. The measured values of the dead time coefficients are: $a = 0.387$ μs and $b = 0.0$ s^2.

Axial and Radial Response

The axial and radial dependence of the position of a source was measured in empty waste drums and in drums filled with variety of typical waste matrices. This was achieved using radially spaced re-entrant source tubes down which Cf or Pu sources could be axially positioned. Typical waste compositions were placed in tinplate containers and stacked in layers within the drum liner to simulate typical waste matrices that may be experienced in practice. The apparatus was built into a typical UK waste drum/liner combination (see Fig. 4).

The apparatus described above enabled placement of a calibrated ^{252}Cf source at various axial positions and radial positions within the measurement chamber. The total and correlated response of the source was measured at each position and the efficiency calculated from the calibrated output of the source.

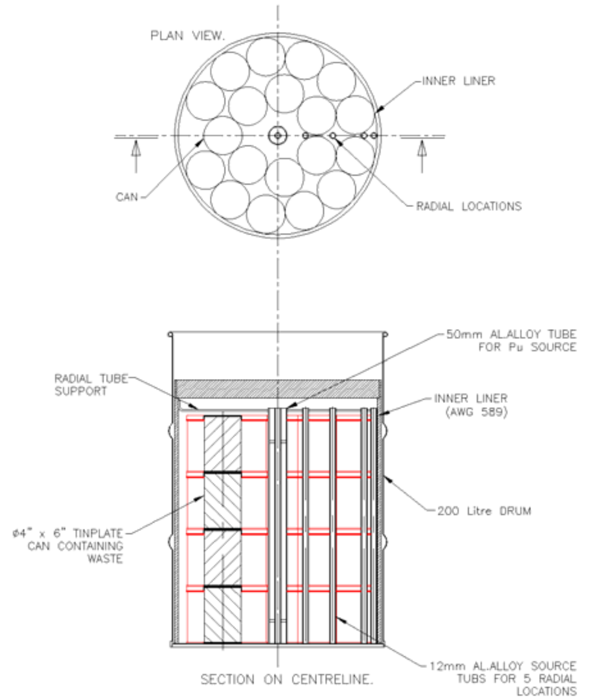

Fig. 4. Apparatus used to measure axial and radial response

Figure 5 shows the axial and radial dependency of the measurement chamber for a moderating waste matrix. The purpose of measuring the calibrated ^{252}Cf source at various vertical and radial positions was to determine a volume weighted response, defined as the ratio of the Model 2200 detector response per unit activity to a given ^{252}Cf activity uniformly distributed throughout the total interior volume the sample container or measurement volume to that of the same activity as a point source at the centre of the sample container or measurement volume.

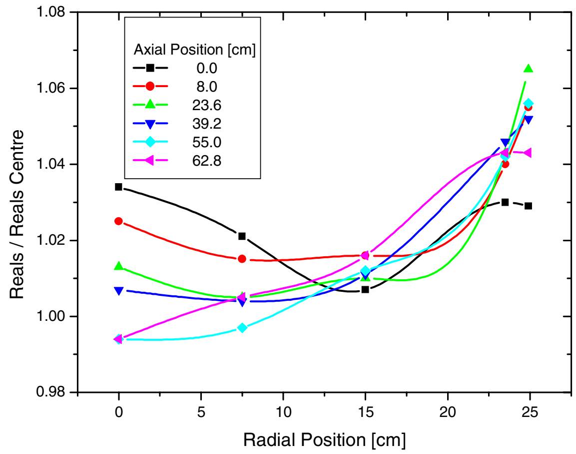

Fig. 5. Radial variation in the coincidence response in a moderating waste matrix of Model 2200, at a number of axial positions.

MODELLING OF THE MODEL 2203

Description of the model

The ANTECH very High Efficiency Neutron Counter (V-HENC) Model 2203 has been based exclusively on the Model 2200 design and is exactly the same construction with 16 detector modules (Fig. 6). In order to increase the efficiency of the system the number of the ^{3}He tubes in the modules has been increased: each of the decagon modules contains 8 counter tubes, while the three modules above and three modules below the measurement chamber each contain 7 tubes. Moreover, the pressure of the ^{3}He gas has been increased to 7.5 atmosphere compared with 4 atmospheres for the Model 2200.

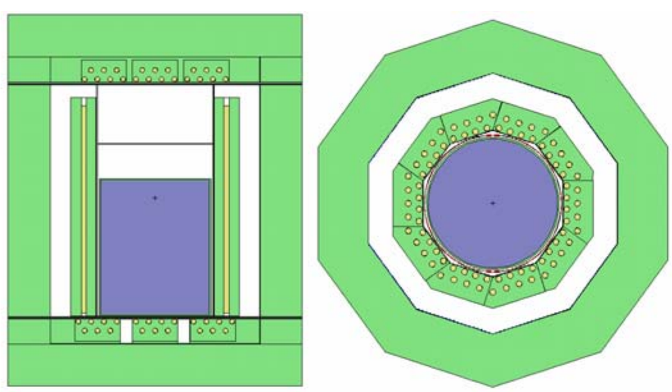

Fig. 6. Model 2203 Neutron Drum Monitor (MCNP Plot), configured for 200 litre waste drum.

Like the Model 2200, the detector modules are surrounded by a polyethylene back shielding of 210mm thickness. As with the Model 2200, a removable aluminium clad cadmium shield can, in principle be fitted to each of the modules. Although for this particular exercise, the 2203 was modelled without cadmium. An interesting feature of the new design is that the detector modules are re-configurable for the measurement of both 200 litre and 400 litre waste drums.

The Monte Carlo calculations of the Model 2203 have been 'benchmarked' back to the Model 2200 configuration. The radial reals response profile of the Model 2203 is improved compared with the Model 2200.

The V-HENC is capable of screening at the UK LLW level (4GBq/te (4000Bq/g)). For the Model 2203 the high detection efficiency (~40% for ^{252}Cf neutrons with the Cd filters removed)) and cosmic ray background reduction techniques yield a detection limit sufficient to segregate at the 4000 Bq/g UK LLW level. Moreover the ANTECH design eliminates totally any high Z materials within the construction of the measurement chamber thus reducing the background due to cosmic ray spallation events significantly.

As mentioned earlier, the design was based on the well proven Model 2200 design with an increased number of tubes in each module. The Monte Carlo code MCNP4B [6] was used to optimise the positioning of the tubes of the Model 2203 in order to obtain the highest detection efficiency. The optimisation was based only on the detection efficiency. Standard MCNP is not suitable to perform optimisation based on the Doubles or Triples. Multiplicity and coincidence counting can be simulated with MCNP-PTA (Pulse Train Analysis), an enhanced version of the MCNP code that has been developed at JRC Ispra [7].

The positioning of the tubes was based only on symmetry and on the distance of the tubes from the front of the module (see Fig. 7). The pitch between the tubes was not a parameter in the optimisation. The parameters to be optimised were, D1, the distance between the front of the module and the first row of tubes, and, D2, the distance between the front of the module and the second row.

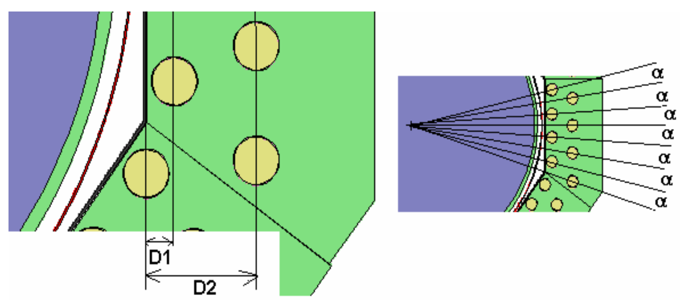

Fig. 7. Model 2203 Decagon Module detail

In the optimisation process the tubes were positioned by selecting first D1 and D2. The vertical position (this is referred to the module shown in right half of Fig. 7) of the tubes was obtained by the intersections of radials originating from the centre of the system and the vertical lines placed at D1 or D2. All angles between the radials, α, are equal to 4.5 degrees, which corresponds to one 80th part of 360 degrees.

The optimum values of D1 and D2 depend on the characteristics of the source, i.e. isotopes, dimensions, and materials inside the cavity. Basically, two configurations have been analysed, one with a ^{252}Cf point source in an empty cavity and one with several types of Pu waste. Once the optimum values for D1 and D2 have been found, the neutron die-away of the system is determined for the optimum. Note that the system was optimised for 200 litre drums.

Optimisation for ^{252}Cf point source

In a preliminary study a broad range of D1 and D2 values has been analysed. This resulted in a smaller range of values close to the maximum efficiency. In Fig. 8 we can see the optimum is found for D1 = 2.0–2.5 cm and D2 = 7.0 cm.

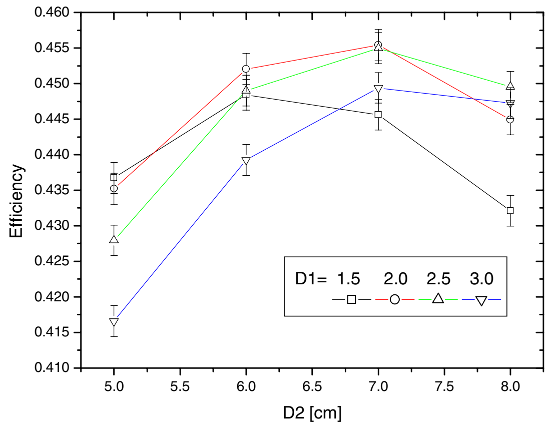

Fig. 8. Simulation results for the detection efficiency of a ^{252}Cf point source placed in the centre of the measurement cavity as a function of D2 for a number of values of D1.

Optimisation for Pu Waste

The Pu material used throughout the modelling work was plutonium dioxide, PuO_2 at two isotopic compositions (i) 'Low Burn-up' (LBU, i.e. ~94% ^{239}Pu and ~6% ^{240}Pu), and (ii) 'High Burn-up' (HBU, i.e. ~76% ^{239}Pu, ~22% ^{240}Pu, with ~1% of each isotope ^{241}Pu and ^{242}Pu).

To access the system sensitivity to the geometry of the Pu source, two configurations have been studied. In the central source configurations the source was concentrated in a small cylinder (radius 1 cm, height 1 cm) placed in the centre (denoted 'CEN') of the drum. In the volume source configuration the Pu material was spread homogenously over the volume of the drum (denoted 'VOL'). Sensitivity to waste matrix was checked by modelling both void drums (denoted 'VOID') and drums filled with a low density (0.3 $g{\cdot}cm^{-3}$) polyethylene matrix (denoted 'PE').

The drum (200 l) had the following characteristics:

- wall thickness: 1 mm
- mild steel: 99% Fe, .25% C, .75% Mn, 7.85 $g{\cdot}cm^{-3}$
- outer diameter: 574 mm
- outer height: 860 mm

Inside the drum there was a polyethylene (HDPE) liner with:

- inner diameter: 541 mm
- wall thickness: 7 mm
- inner height: 671 mm

Figures 9-12 show the MCNP simulation results for the detection efficiency for all the configurations mentioned earlier.

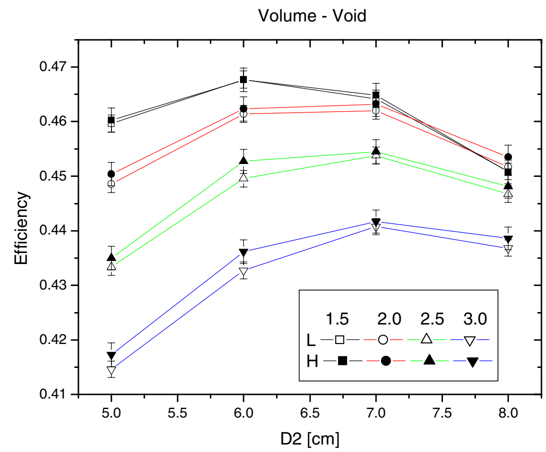

Fig. 9. Simulation results for the detection efficiency for Low (L) and High (H) Burn up Pu waste as a function of D2, for a number of D1 values. The configuration was for a volume source inside an empty drum.

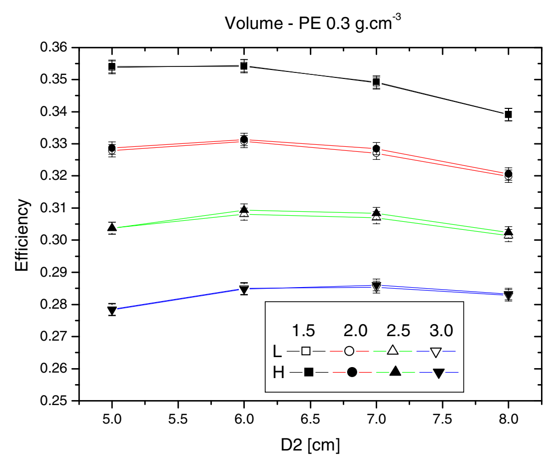

Fig. 10. As Fig. 9 but for a drum filled with 0.3 $g{\cdot}cm^{-3}$ PE.

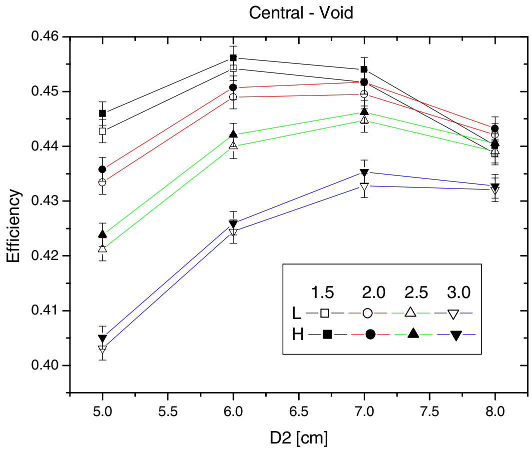

Fig. 11. As Fig. 9, but for a central source.

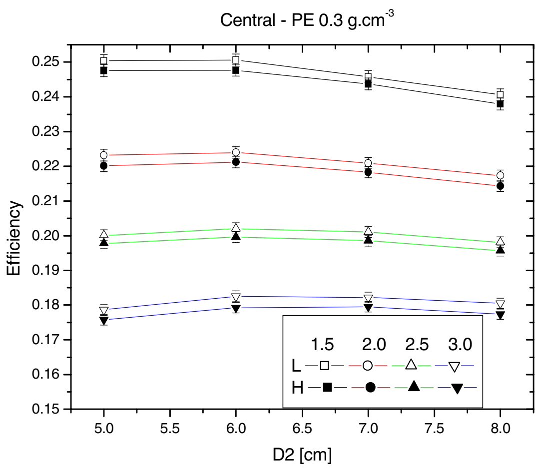

Fig. 12. As Fig. 10, but for a central source.

From the simulation results presented above the optimum values were found to be: D1 = 1.5 cm and D2 = 6.0 cm. This is the optimum for all the cases with a drum, and is slightly different from the ^{252}Cf point source result. This difference can be explained by the 7 mm PE liner found in the drum: less moderation by the modules was required. Since the system was optimised to measure 200 l drums the corresponding optimum values were selected and the efficiency and die away time of the system determined on that basis. The optimum values found for the decagon modules were also applied to the modules above and below the drum.

Efficiency

The results for the detection efficiency for the optimum system configuration are given in Table II below.

Table II. Results for detection efficiency obtained with D1 = 1.5 cm and D2 = 6.0 cm for 200 and 400 l drums. The error is the statistical error with a confidence interval of 95% (≈2σ).

Neutron Die-Away Time

The die away curves in Fig. 13, did not display a single exponential decay. This was attributed to the absence of a Cd-liner on the surface of the measurement cavity. The first part of the decay curves was found to be quite steep, indicating fast decay.

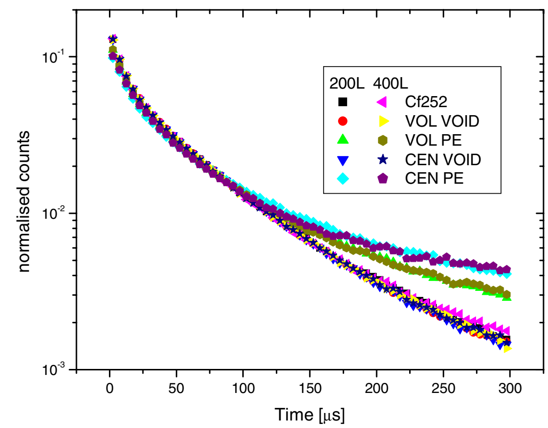

Fig. 13. Normalized counts as a function of time for different cases for the optimum detector system (D1 = 1.5 cm, D2 = 6.0 cm).

It can be seen that the shape of the die away is independent of the drum size. Moreover, the response is also similar when PE is absent (Cf252, VOL VOID and CEN VOID). There remain thus 3 cases for further analyses: Cf252, VOL PE, and CEN PE (200 l drum). When a range from 20 to 150 µs was selected, the decay was fairly well described by a single exponential decay function. The fit results are listed in Table III.

Table III. Fit results for neutron die-away time, τ, obtained with D1 = 1.5 cm and D2 = 6.0 cm

CASE			EFFICIENCY			
			200 l		400 l	
Geometry	**Matrix**	**Pu**	**Mean**	**Error**	**Mean**	**Error**
Point	Void	Cf	0.4482	0.0022	0.3262	0.0019
Volume	Empty	LBU	0.4688	0.0022	0.3423	0.0019
Volume	Empty	HBU	0.4702	0.0022	0.3431	0.0019
Volume	PE	LBU	0.3585	0.0020	0.2433	0.0017
Volume	PE	HBU	0.3589	0.0020	0.2430	0.0017
Central	Empty	LBU	0.4568	0.0022	0.3297	0.0019
Central	Empty	HBU	0.4591	0.0022	0.3323	0.0019
Central	PE	LBU	0.2536	0.0017	0.1401	0.0013
Central	PE	HBU	0.2508	0.0018	0.1377	0.0013

for the Model 2203 (200 l position).

Case	τ [µs]
Cf252	45.3
VOL PE	43.3
VOL CEN	43.1

The die-away times in Table III are much lower than the 64 μs of the Model 2200, this can be attributed to the fact that the system is operated in fast mode (without Cd-liner on the inner surface of the chamber). Moreover, the presence of more ^{3}He in the Model 2203 also speeds it up.

REFERENCES

[1] K. Boehnel "Determination of Pu in Nuclear Fuels using the Neutron Coincidence Method" KFK 2203 and AWRE Translation No. 70, 54/4252 (1978).

[2] W. Hage & D.M. Cifarelli “On the Factorial Moments of the Neutron Multiplicity Distribution of Fission Cascades” Nuclear Instruments and Methods in Physics Research A **236**, 165-177 (1985).

[3] D.M. Cifarelli & W. Hage “Models for a Three-Parameter Analysis of Neutron Signal Correlation Measurements for Fissile Material Assay” Nuclear Instruments and Methods in Physics Research A **251**, 550-563 (1986).

[4] W. Hage & D.M. Cifarelli “Correlation Analysis with Neutron Count Distributions in Randomly or Signal Triggered Time Intervals for Assay of Special Fissile Materials” Nuclear Science and Engineering **89**, 159-176 (1985).

[5] B. Pedersen, W. Hage, J.A. Mason “Neutron Multiple Correlation Analysis Method Applied to the Assay of Radioactive Waste” Proceedings of the 13th Annual Symposium on Safeguards and Nuclear Materials Management, Avignon (1991).

[6] J.F. Briesmeister, Editor "MCNP A General Monte Carlo N-Particle Transport Code - Version 4B" LA-12625-M (1997).

[7] M.R. Looman, P. Peerani, and P. Schillebeeckx, “An Overview of NDA Instruments Modelled with the MCNP-PTA Code at the JRC Ispra”, 23rd ESARDA Symposium on Safeguards and Nuclear Material Management. Report EUR 19944 EN (2001).

[8] J.A. Mason, L. Bondar, W. Hage and B. H. Pedersen, "The Advantages of Neutron Multiple Correlation Analysis", Proc. of the 15th ESARDA Symposium on Safeguards and Nuclear Materials Management, Rome, May 1993, p. 355-358.

[9] J.A. Mason, L. Bondar, W. Hage, B. Pedersen, M. Swinhoe and F.W. Ledebrink, "The Assay of Pu by Neutron Multiplicity Counting Using Periodic and Signal Trigger Methods", Proc. of the 34th Annual Meeting of the Institute of Nuclear Materials Management, Scottsdale, Arizona, USA, July 1993, p. 471-476.