Improvements to the Operational Efficiency of the Portsmouth "Blue Box" Assay System -15162

Alan Simpson *, Martin Clapham *, Alex Couture*, Dean Lobdell**, David Dreher**
* Pajarito Scientific Corporation, 2976 Rodeo Park Drive East, Santa Fe, NM 87505, USA
** Fluor-B&W Portsmouth LLC, PO Box 548, Building X-157B, 3930 US Route 23 South, Piketon, OH 45661, USA
asimpson@pscnda.com, mclapham@pscnda.com, acouture@pscnda.com, david.dreher@fbports.com, dean.lobdell@fbports.com

ABSTRACT

The "Blue Box" in Building 326 at Portsmouth comprises a dual cargo container lined with high density polyethylene with 5% (elemental) boron that acts as a neutron shield. This construction provides a low environmental neutron background to support the capability to attain low values of minimal detectable activity (MDA).

A Non-destructive Assay (NDA) measurement platform known as the Fluor B&W Portsmouth Slab Counter (FBPSC) is housed inside the Blue Box. This comprises four neutron slab detectors supported with an electronic system supplying power and associated signal processing. The system is operated by a laptop computer which has data acquisition and analysis capabilities. Each slab contains five ^{3}He proportional counters embedded in polyethylene.

The system is designed to measure the mass of U-235 in coolers from the Portsmouth site using passive neutron counting. The system was calibrated by taking measurements with ^{252}Cf sources attached to various positions on the outside surfaces of a surrogate cooler. In addition, Monte-Carlo N-Particle (MCNP) simulations were performed to extend the baseline calibration for coolers of various sizes used in the gaseous-diffusion process. The basic principle of operation is to measure the total neutron emission from the item and determine the mass of ^{235}U using the system efficiency (determined from MCNP calculations combined with the calibration measurements) combined with known isotopic information (^{235}U enrichment and U-235/U-234 ratio), assumed chemical composition, and gross cooler mass.

The FBPSC analysis program calculates the mass of U-235 within the cooler being assayed, together with the associated Total Measurement Uncertainty (TMU). A background correction is performed that accounts for the measured environmental background with a cosmic-ray spallation correction (based on the cooler's gross weight). The software suite developed for the system also includes the capability to track and trend measurement control data in order to quickly identify conditions where the system has drifted outside of its quality control envelope.

The MDA of the system has been determined to be less the 15g of U-235 for count times that are typically less than 5 minutes (depending on the uranium isotopics). This is primarily the result of the low neutron background rate inside the Blue Box (approximately 3 times lower than at nearby locations outside the box). Additionally, various software features have been designed to yield optimum counting performance for the system including count time adjustment and interval filtering to reject statistical outliers that are usually associated with cosmic spallation events.

The operation of the system has proven to be very efficient, with the short measurement time providing a high throughput rate of coolers through the Blue Box NDA system. Significant operational benefits and improved data traceability have been yielded compared to the previous measurement process, which involved operators taking measurement with counter/scalers and manually entering data into spreadsheets. As of August 2014, this system successfully measured all coolers required to support completion of a

contractually-required Performance Milestone. There are approximately 1900 coolers available for measurement.

INTRODUCTION

The Portsmouth Gaseous Diffusion Plant is a DOE-owned facility located near Piketon, Ohio. The plant was previously engaged in uranium enrichment for defense purposes but was deactivated in 1981. Fluor-B&W Portsmouth (FBP) is responsible for the decontamination and decommissioning (D&D) activities including removal of a large quantity of heavy process equipment such as converters, compressors and coolers. These items require characterization in order to be appropriately categorized for treatment / disposal.

Method selection studies were performed using Monte-Carlo N-Particle (MCNP) [1] simulations in order to determine the most appropriate Non-destructive Assay (NDA) method for each type of plant equipment. For the coolers, it was determined that total neutron counting (using an array of slab counters) would meet the required Minimum Detectable Activity (MDA) of less than 15g U-235 without the problem of large Total Measurement Uncertainty (TMU) associated with high resolution gamma spectroscopy.

In order to provide a low neutron background environment for neutron assay activities (i.e. to achieve low MDA in a short count time), FBP have constructed a neutron-shielded enclosure known as the “Blue Box” located in Building 326. This comprises a dual cargo container lined with layers of high-density polyethylene and borated polyethylene containing 5% elemental boron.

The FBP equipment inventory includes a large number of slab counters that have previously been operated with hand-held counter / scaler electronics. Data were acquired manually and then entered into spreadsheets for analysis. It was recognized that this operation modality was inefficient, labor intensive, and prone to human error. Therefore a new system, known as the Fluor B&W Portsmouth Slab Counter (FBPSC), was developed in order to perform rapid, high quality NDA measurements inside the Blue Box. This system was the result of a collaborative effort between FBP and Pajarito Scientific Corporation (PSC).

The system has been calibrated and validated using a combination of on-site measurements and MCNP modelling. The Monte Carlo analysis has proven to be a valuable tool to allow the operational envelope to be extended and to determine the system’s performance parameters including MDA and TMU.

SYSTEM DESCRIPTION

The FBPSC comprises an array of up to four neutron slab detectors supported with an electronic system supplying power and associated signal processing. The system is operated using custom software installed on a laptop computer that controls data acquisition and performs all data analysis. Each slab contains five ^{3}He proportional counters (at 4 atmosphere pressure) embedded in polyethylene.

The basic principle of operation is to measure the total neutron emission from the item and determine the mass of U-235 using isotopic information provided by Fluor B&W Portsmouth (FBP), assumed chemical composition, gross item mass, and a detector efficiency based on MCNP calculations validated by the calibration measurements.

The system software provides a real-time visual indication of the mass of U-235 within the cooler being as well as the TMU and MDA. A correction is performed that accounts for the measured neutron background adjusted for cosmic-ray spallation (based on the cooler’s gross weight). An interval filtering method is used for elimination of statistical outliers from the acquired data intervals. Electronic data reports are generated after each measurement that provide the full analysis results. The software suite also provides the capability

to track and trend measurement control data in order to quickly identify conditions where the system has drifted outside of its quality control envelope.

FBPSC ANALYSIS METHOD

The mass of U-235 in each object, M_{U-235}, is determined by the FBPSC using Equation 1.

$$M_{U-235} = \frac{R_N}{SNE(^{235}U)\varepsilon} \qquad \text{(Eq. 1)}$$

where R_N is the measured net count rate, ε is the detection efficiency and $SNE(^{235}U)$ is the specific neutron emission rate per gram of U-235. The latter is calculated (per Equation 2) from the uranium isotopic information provided by FBP for each of the different cooler locations (defined by the U-235 enrichment, and U-235:U-234 mass ratio) as well as spontaneous fission and (alpha,n) neutron emission rates for U-234, U-235, and U-238 with the assumption that the hold-up will be in the chemical form $UO_2F_2{\cdot}2H_20$.

$$SNE(^{235}U) = \frac{1}{f_{235}} \sum_{x=4,5,8} f_{23x} \left(SNE_{23x}^{SF} + SNE_{23x}^{(\alpha,n)} \right) \qquad \text{(Eq. 2)}$$

where f_{23x} are the mass fractions of the uranium isotopes. The specific neutron emission rates for spontaneous fission (SF) and (α,n) are given in Table I (the reference document from which data are obtained is indicated in brackets).

Table I. Specific Neutron Emission Rates used in FBPSC Assay Calculations

Isotope	**SNE^{SF} (n/s/g)** Spontaneous Fission	**$SNE^{(\alpha,n)}$ (n/s/g)** $UO_2F_2{\cdot}2H_20$ (α, n)
U-234	5.020E-03 [2]	1.728E+02 [3]
U-235	2.990E-04 [2]	2.963E-02 [4]
U-238	1.360E-02 [2]	3.690E-03 [3]

A library was developed of the specific neutron emission rates expected for various plant locations based upon the enrichment and U-235:U-234 ratio associated with these locations. For each cooler type, the software provides a set of applicable calibration coefficients for the operator to select from based on the cooler size and its plant location.

MCNP MODEL

The slab counters, “blue box” and coolers were simulated in an MCNP model shown in Figure 1. The coolers are usually placed on wooden pallet that is, in turn, placed on a metal pallet. The stand-off distance from the front of each slab to the front surface of the cooler was set to 56cm which is equal to half of the reference cooler (Type 30EA) length. To determine the optimal assay configuration, the slabs were modelled in two arrangements: (i) an orthogonal layout with the slabs aligned along the primary axes of the cooler (as shown in Figure 1), (ii) a diagonal layout with the slabs aligned at 45 degrees relative to the Process Gas (PG) axis of the cooler (as shown in Figure 2).

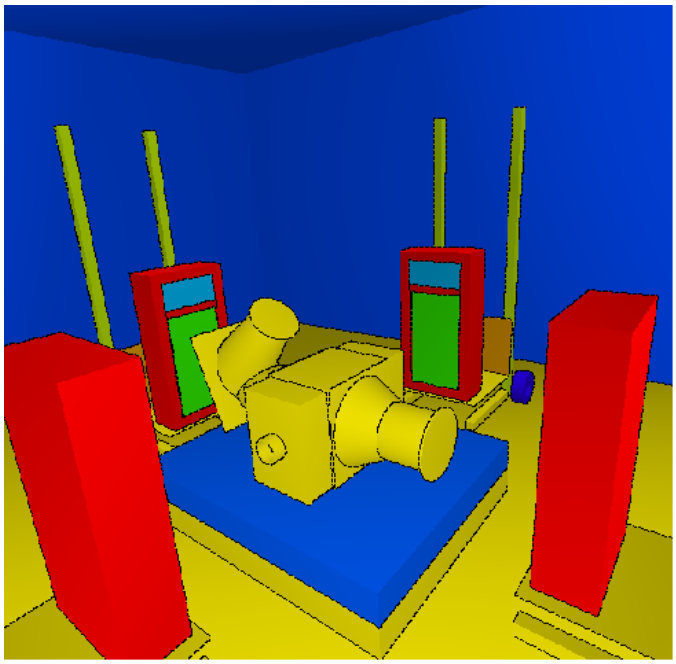

Fig. 1 MCNP Plot Showing Cooler and Neutron Slab Counters inside Shielded Enclosure

Each cooler comprises an internal steel enclosure that contains copper tubes and aluminum fins. The uranium deposition (source location) within the coolers is expected to be within this enclosure. Various additional process gas nozzles are attached to the coolers, but these are of lesser importance.

The MCNP model calculates the reaction rate (per emitted neutron) for (n, p) events in the He-3 gas region of the slab counter detectors. This rate is equivalent to the counts per emitted source neutron in each neutron detector assuming all events are counted.

The absolute total neutron counting efficiencies for the 4-slab FBPSC determined from the MCNP model (for the "diagonal" arrangement) are summarized in Table II for various types of cooler. In this case, a hydrated uranyl fluoride source spectrum (alpha,n) was uniformly distributed in the fin/tube enclosures.

TABLE II. MCNP Calculated Efficiencies for Various Sizes of Cooler

Cooler Size	Process Gas Axis Length (cm)	Typical Weight (kg)	FBPSC 4-Slab Efficiency (%)
100EC	53.3	454	1.26%
75EB	48.3	454	1.28%
60EA	41.9	380	1.32%
40EA	36.8	330	1.36%
30EA	30.5	283	1.40%
15EA	24.1	246	1.41%

The model was run in a number of different scenarios to evaluate variations from the baseline calibration assumptions including:

- Source material positioned at various locations inside the tube / fin enclosure (study of the deviation from the uniform distribution assumption)

- Fins replaced by a homogenous region of aluminum of equivalent bulk density.
- The entire cooler replaced with air (no matrix).
- The Process Gas Nozzles (PGNs) replaced with air (to simulate less matrix material).
- Mass of material in the PGN regions doubled (to simulate additional matrix material).
- The cooler was moved horizontally by ± 2.5cm from its nominal (center of pallet) position along two axes (left / right and forwards / backwards) to simulate the accuracy to which the operator is expected to be able to position the cooler onto the pallet at the nominal standoff).
- The active center of the slab was moved ± 2.5 cm from its nominal (center) height to simulate the maximum operational error in vertical alignment.

CALIBRATION

The system was calibrated at Portsmouth by taking measurements with a set of Cf-252 sources attached the surface of a surrogate cooler (which had no accessible internal regions) arranged in the "diagonal" position as shown in Figure 2.

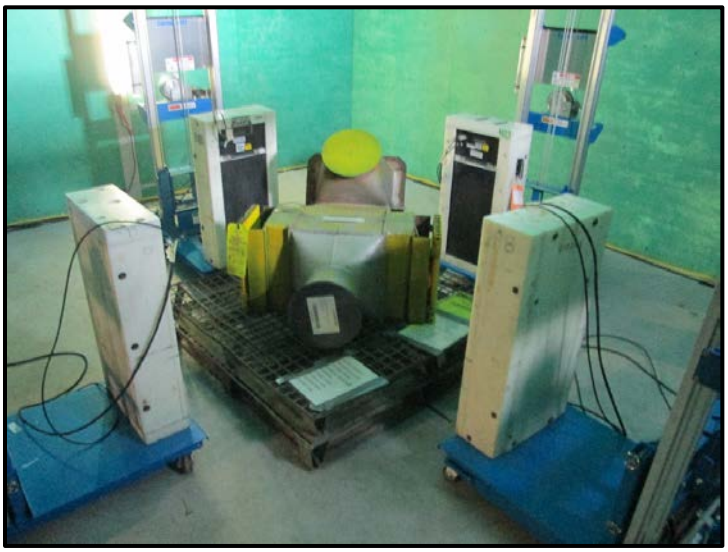

Fig. 2 Measurement Configuration with Surrogate Cooler

In addition, a series of "baseline" efficiencies for each of the slabs were taken with a Cf-252 source located 12 inches in front of each slab surface. For the latter counts, the slabs were moved, one by one, to a corner of the Blue Box to minimize slab to slab reflection effects. As shown in Table III, the average measured efficiency was 1.36% ± 0.11%. The latter uncertainty is dominated by the relative error in the certified neutron emission of the Cf-252 standards (approximately 8%). Under the same conditions, the MCNP predicted efficiency for each slab was 1.38% which demonstrates good agreement in efficiency between the model and the individual slab counter results.

TABLE III. Total Neutron Counting Efficiency for each slab with a Cf-252 Source at 30cm

Slab Counter Serial Number	Total Neutron Efficiency (%)
248	1.35%
379	1.37%
380	1.35%
423	1.37%

Figure 3 shows the measured calibration results with the 4-slab system (compared to the equivalent MCNP value) for the Cf-252 source located at reference positions on the surfaces of the cooler. The error bars in the measured values are at approximately 8% due to the uncertainty in the source certified activity. Again, this demonstrates consistency between the MCNP and measured results to within the expected precision.

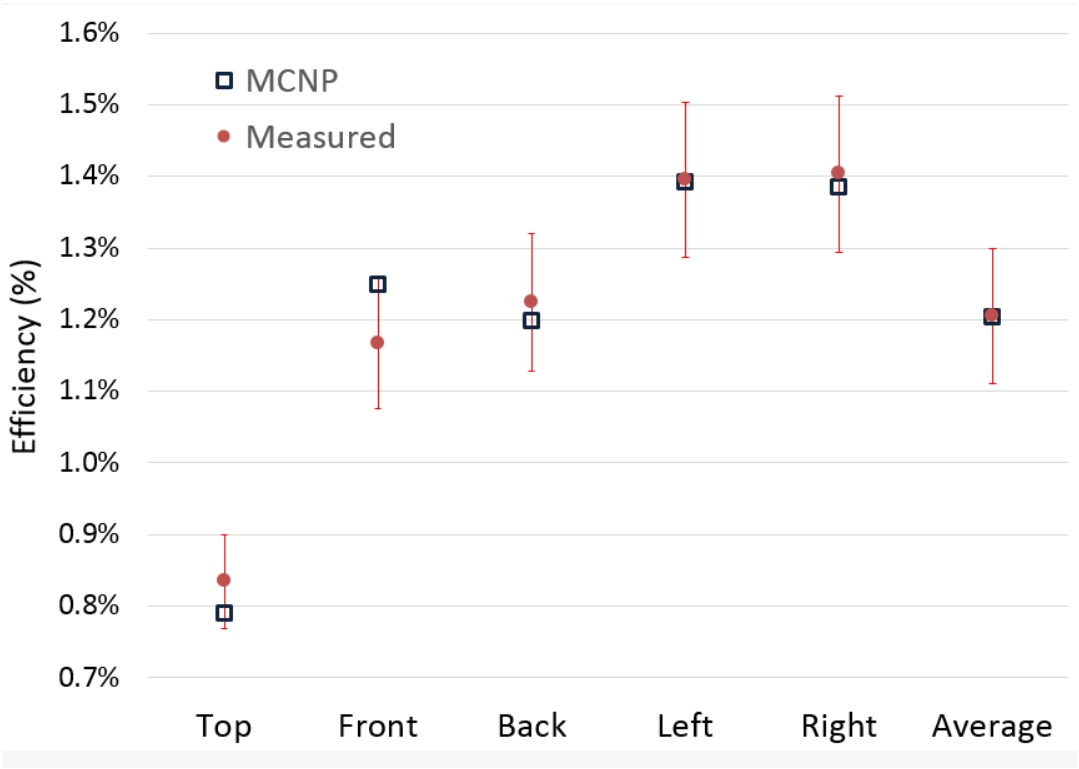

Fig. 3 MCNP and Measured Efficiencies (4-slab system) with Cf-252 Sources on the Surrogate Cooler

After this successful benchmarking, the MCNP model was used to extend the system calibration for a variety of cooler sizes that are encountered at Portsmouth. The absolute efficiency ε of the system for a given cooler size for a given slab arrangement and stand-off is given by Equation 3.

$$\varepsilon = \varepsilon_{ExtSurr}^{Cf-252} CF \qquad \text{(Eq. 3)}$$

where $\varepsilon_{ExtSurr}^{Cf-252}$ is the efficiency measured during calibration using a Cf-252 source averaged over 5 locations on the exterior surface of the surrogate cooler. *CF* is a correction factor (determined in MCNP) to adjust for the differences in energy spectrum and source distribution as given in Equation 4.

$$CF = \frac{\varepsilon_{Uni}^{UO_2F_2 \cdot 2H_2O}}{\varepsilon_{ExtSur}^{Cf-252}} \qquad \text{(Eq. 4)}$$

where $\varepsilon_{Uni}^{UO_2F_2 \cdot 2H_2O}$ is the MCNP efficiency with a $UO_2F_2 \cdot 2H_20$ (α,n) neutron energy spectrum located in a uniform distribution within the tube / fin enclosure, and $\varepsilon_{ExtSur}^{Cf-252}$ is the MCNP efficiency with a Cf-252 neutron energy spectrum averaged over the equivalent 5 locations on the exterior surface on the model of the surrogate cooler.

PARTICLE SIZE EFFECTS

The calibration assumes a dilute uranyl fluoride source comprising small particles where there is no self-shielding or self-multiplication. A study was performed to quantify these effects in source particles of various sizes and compositions located either (i) within the cooler, or (ii) at 30cm from a neutron slab counter. The latter scenario enables the particle-size effects to be isolated from other effects in the surrounding matrix. Enrichments were varied from 50% to 90% U-235 by mass.

Induced fission events occur in the uranium lumps from the emitted (α,n) neutrons in both U-235 (primarily thermal fission) and U-238 (fast fission). For a 2cm radius sphere, U-235 fission neutrons represent 99.2% of all fission neutrons produced in the 90% enriched material. For 50% enriched material this percentage is 93.4% with the remainder of neutrons made up from U-238 fast fission (although the fission neutrons represent less than 1% of the total neutron emission). The results indicate that fission is the dominant factor in determining the change in relative efficiency for each scenario modeled (in the lump versus non-lump comparison). Other factors such as the moderating / absorbing effect of the water in the hydrated uranyl fluoride have only a minor effect on the relative efficiency (compared to an equivalent geometrical source distribution in an air matrix).

For all hydrated uranyl fluoride particles less than 1cm in diameter, the neutron efficiency variation due to particle size is negligible (<1%). This size is equivalent to 0.86 – 1.56g of U-235 depending on enrichment. For larger particles up to 100g U-235 (i.e. a 4cm diameter sphere at 90% enrichment) the change in efficiency is less than 3%. Particles of the latter size are not expected to be encountered in the Portsmouth waste. Thus for a range of actual particle sizes consistent with the Portsmouth D&D waste, the particle size effect can be considered to be negligible in total neutron counting.

Note that, by contrast, the self-shielding effect would be a significant component of TMU for a gamma assay that relies on the low energy photons emitted by U-235.

TMU ESTIMATES

The Total Measurement Uncertainty (TMU) for the assay of the cooler will comprise various components including:

- **Statistical.** The statistical or random uncertainty term will be dependent on the quantity of uranium source material, the counting efficiency, and count time used. Quantification of the statistical term is performed by the FBPSC software for each assay.
- **Matrix**. The effect of variable matrix contents is expected to be small as the MCNP results have indicated that the matrix has little effect on counting efficiency. In the extreme case with the cooler matrix not present (i.e replaced by air) relative to the “cooler matrix present” reference case, the variation in efficiency is <± 7% (for uniform sources). For point sources with the cooler matrix

present / not present the average bias term is ± 9.4%. The grade of steel and impurities in the copper / aluminum are not expected to have a significant impact on efficiency. With a homogeneous mass of aluminum replacing the fins, the impact on efficiency was <± 0.3%, so the distribution of fins is considered to be of no significance in neutron mode. The changes to the process gas nozzles results in a bias of <± 9% in all scenarios considered, i.e, removal of the PG nozzles or addition of extra steel in the PG nozzles. Consequently taking the bounding case (point sources with cooler matrix removed), the matrix uncertainty term is estimated as ±9.4%.

- **Source Geometry.** The uncertainty term due to variation in source geometry has been quantified by evaluation of the relative efficiency in point source efficiencies. The variations observed across the PG axis and across the coolant axis were both smaller than the variation from the point source distribution. As a result, the source geometry uncertainty is estimated to be ± 3.9%.
- **Self-Shielding / Multiplication.** It has been demonstrated that self-shielding / multiplication effects are negligible for the particle size ranges expected in the Portsmouth waste.
- **Calibration.** The calibration uncertainty term will comprise the uncertainty in the Cf-252 calibration source certified activity (8%) and any energy related terms, e.g., due to differences in the neutron energy spectra between the model and reality and errors introduced during the MCNP neutron transport calculations. It is expected that the majority of neutron emissions from uranium in the cooler will arise from (α,n) emissions from U-234 in the form of hydrated uranyl fluoride. However, the energy spectra used in the model only represents an approximation of the actual neutron emissions. Consequently the MCNP calibration uncertainty term is quantified by evaluating the efficiency with the (α,n) spectrum compared to an identically distributed Cf-252 spontaneous fission spectrum. The resulting efficiency difference is approximately 8%. Combining the energy and source activity terms together in quadrature, the estimate of calibration uncertainty is ± 11.3%.
- **Cooler Size.** The efficiency term is dependent on the size of cooler and is predicated by the size of the fin/tube enclosure. The average difference in the efficiency for the uniform source case between neighboring cooler sizes is ± 2.5%.
- **Cooler Position Variability**. An uncertainty estimate of ± 2.0% accounts for variations in positioning of the cooler and slab with respect to each other with a ± 2.5 cm tolerance.
- **Cooler Orientation Variability.** Even with careful alignment, there will be some degree of error in the orientation angle of the cooler on the pallet. A value of ± 5.0% provides a reasonable estimate for this term based on analysis of orthogonal and diagonal data.
- **Specific Neutron Emission (SNE) rate.** An uncertainty of ± 5.4% is applicable to the U-234 SNE rate [3].
- **Uranium Isotopics.** An uncertainty of ± 25% has been estimated by FBP for declared uranium isotopics (i.e. for the U-235:U-234 ratio).

It was demonstrated that the diagonal array had a higher efficiency and lower TMU compared to the orthogonal mode and was therefore selected as the recommended alignment for measurements of the cooler. A summary of the TMU components for the diagonal mode is given in Table IV.

TABLE IV. TMU Estimate for Cooler Assay using the FBPSC in Diagonal Mode

Uncertainty Component	Uncertainty (%)
Calibration	11.3%
Positional	2.0%
Orientation	5.0%
Cooler Size	2.5%
Source Geometry	3.9%
SNE Rate	5.4%
Uranium Isotopics	25.0%
Matrix	9.4%
Total	30.3%

SPALLATION CORRECTION

The FBPSC software performs a gross weight dependent background correction. This accounts for the increase in the background neutron count rate associated with cosmic ray spallation events in the metal cooler.

Spallation correction parameters were determined by performing background measurements with various clean (uranium-free) steel scrap objects of known mass in the measurement area as shown in Figure 4. This included a 159 kg drum filled with scrap iron, three 50 kg weights, a 25 kg weight, and a 61 kg disk.

Fig. 4 Spallation Neutron Background Correction Measurement Configuration.

The spallation correction proceeds by adjusting the measured net count rate using Equation 5. Secondly, the net rate is determined as follows:

$$R_N = R_v - R_{b0}(Z^0 + Z^1 M) \qquad \text{(Eq. 5)}$$

where, R_v is the gross count rate, R_{b0} is the background rate (with no cooler present), M is the gross weight of the cooler (in kg), Z^0 and Z^0 are the scaling coefficients (offset and gradient). The Z coefficients were determined by linear regression to be 0.99 ± 0.03 and 1.62E-04 ± 0.42E-04 kg^{-1}, respectively.

MDA ESTIMATES

Using the Currie method [5], the minimum detectable activity (MDA) is defined as that level of radioactivity that, if present, yields a measured value greater than the critical level with a 95% probability, where the critical level is defined as that value which measurements of the background will exceed with 5% probability. In determining the MDA, an account is made for interferences from different matrix conditions or radiation backgrounds that occur in the waste. For operational reasons, it is necessary that the FBPSC attains an MDA of less than 15 g U-235.

The "Matrix MDA" for each cooler type MDA_{matrix} is defined in Equation 6 using the method described in NUREG-1507 [6] for the scenario where assay time and background time are both equal to a nominal "MDA count time", T_{matrix}^{MDA}.

$$MDA_{matrix} = \frac{3 + 3.29\sqrt{2R_b T_{matrix}^{MDA}}}{SNE(^{235}U)\varepsilon T_{matrix}^{MDA}} \qquad \text{(Eq. 6)}$$

The MDA_{matrix} values were determined for the FBPSC by performing a set of replicate background measurements under typical operating conditions with T_{matrix}^{MDA} set to 600 seconds.

For each real measurement, the "Assay MDA" MDA_{assay} will be given by Equation 7, also found in NUREG-1507 [6], in the case where the assay count time T_v and background count time T_{Bv} may both differ from the reference count time T_{matrix}^{MDA}.

$$MDA_{assay} = \frac{3 + 3.29\sqrt{R_b T_v \left(1 + \frac{T_v}{T_{Bv}}\right)}}{SNE(^{235}U)\varepsilon T_v} \qquad \text{(Eq. 7)}$$

Equations 6 and 7 may be combined to yield the approximation given Equation 8 which expresses the Assay MDA in terms in the (pre-determined) Matrix MDA and the various count times:

$$MDA_{assay} \approx MDA_{matrix}\sqrt{\frac{T_{matrix}^{MDA}}{2}\left(\frac{1}{T_v} + \frac{1}{T_{Bv}}\right)} \qquad \text{(Eq. 8)}$$

Equation 8 is particularly useful because it may be rearranged to determine the assay count time required to achieve the desired MDA (i.e. below 15 g U-235). The FBPSC automatically calculates the assay count time needed for each cooler based on its known isotopic grade and efficiency. This improves system throughput ensuring that the minimum count time is used for each cooler.

NON-POISSON MDA COMPONENTS

In addition to the usual Poisson statistical effects, the MDA also needs to account for background correction effects and fluctuations that may arise between the time of day the background is measured and the time of day when the assay is performed (standard practice is for a background measurement to be taken every 24 hours during operational periods). Fluctuations in the magnitude of the background neutron count rate can arise from daily variation in barometric pressure, geomagnetic field, diurnal cycle, solar activity and weather [7, 8]. This term is expressed as the fraction *F*.

The *F* term is dependent on a complex mix of local, global and extra-terrestrial phenomenon and cannot be easily determined from modelling. Studies with neutron counters on other sites in the United States at a similar altitude to Portsmouth (e.g. Oak Ridge and Hanford) has found that *F* varies from site to site up to 3%. A long series of background measurements were performed inside the Blue Box in 2013 and 2014. The results of these runs indicated a best estimate of *F* of 1%.

The FBPSC software's cosmic spallation correction also impacts the MDA due to the uncertainty in the Z parameter and variations in spallation neutron production due to the elemental composition of the cooler. The relative uncertainty in the adjusted background rate R_b is given in Equation 9.

$$\left(\frac{\Delta R_b}{R_b}\right)^2 \approx \left(\frac{\Delta Z^0}{Z^0}\right)^2 + \left(Z^1 M\right)^2 \left(\left(\frac{\Delta Z^1}{Z^1}\right)^2 + \left(\frac{\Delta M}{M}\right)^2 + \Omega_{Sp}^2\right) \qquad \text{(Eq.9)}$$

The term Ω_{Sp} accounts for the variation in spallation neutron production due to the elemental composition of the cooler. The surrogate materials used to determine the spallation neutron production weight were, like the coolers, predominantly steel by weight. On average, the coolers are approximately 86% by weight steel and 14% from aluminum (fins) and copper (tubes). The copper represents the largest percentage of the secondary mass. The spallation rate is proportional to neutron density of the target material [8]. Based on the neutron density of aluminum (8.68E+23 n/cm^3), copper (2.97E+24 n/cm^3) and iron (2.52E+24 n/cm^3), the spallation rate for each element should be approximately, 0.013 n/s/kg-Al, 0.043 n/s/kg-Cu and 0.038 n/s/kg-Fe. The spallation rate uncertainty term Ω_{Sp} due to variation in the cooler's elemental composition has been estimated to be 0.8% representing the percent difference over the bounding cases.

Using ±1% and ±26% as estimates of uncertainty in Z^0 and Z^1 respectively, and Ω_{Sp} of ± 0.8%, the relative uncertainty in R_b is approximately ± 1.8%. Adding the 1% diurnal *F* term (described above) in quadrature with this background correction uncertainty gives a final estimate of systematic background uncertainty of ± 2.0%. This term is propagated by the FBPSC software together with the statistical uncertainty determined for both the background and gross count rate.

Assuming the non-Poisson effects (which alter the magnitude of the true background compared to the corrected estimate) will give rise to a normally distributed variance, then this combined term can be modeled as acting independently of the Poisson related statistical term. The estimate of final MDA for each

cooler type is then derived by adding the Poisson and non-Poisson terms in quadrature in the Currie formula [5] as per Equation 10.

$$MDA_{matrix} = \frac{2.71 + 4.65\sqrt{R_b T_{matrix}^{MDA} + \left(R_b T_{matrix}^{MDA}\right)^2 \left(F^2 + \frac{\Delta Z_0^2}{Z_0^2} + Z_1^2 M^2 \left(\frac{\Delta Z_1^2}{Z_1^2} + \frac{\Delta M^2}{M^2} + \Omega_{Sp}^2 \right) \right)}}{SNE(^{235}U)\varepsilon T_{matrix}^{MDA}} \quad \text{(Eq. 10)}$$

Examples of Matrix MDA (600 s count time) and Assay MDA are provided in Table V for a U-235 enrichment of 42%, U-235:U234 ratio of 116, SNE of 1.544 n/s/g-U-235 with a background count rate (with no cooler) of 1.053 ± 0.029 cps (measured in April 2014). The second column provides the assay count time (rounded up to the nearest 10s interval) that has been calculated to achieve an Assay MDA of less than 15 g U-235.

TABLE V. Estimates of MDA for Cooler Assay using the FBPSC

Cooler Size	Assay Count Time (seconds)	Matrix MDA (g U-235)	Assay MDA (g U-235)
100EC	310	12.70	14.49
75EB	290	12.48	14.59
60EA	250	11.76	14.56
40EA	220	11.20	14.59
30EA	200	10.66	14.44
15EA	190	10.51	14.53

CALIBRATION CONFIRMATION

A series of measurements were performed in July 2014 with a set of well characterized hydrated uranyl fluoride standards to confirm the system calibration. These "Working Reference Materials" (WRMs) each contain approximately 180 grams of U-235 at 5% enrichment [9]. Ten of these sources had their neutron emission rates characterized by Los Alamos National Laboratory (LANL), at about 20 n/s each [4].

The internal regions of the surrogate cooler were not accessible, so the WRMs were placed in source holder tubes attached to the outside of the cooler. Six 20-minute measurements were performed using the FBPSC with the surrogate loaded with WRMs containing a total of 140.8 ± 0.08 g of U-235. Between each measurement the 16 WRMs were randomly shuffled in the source tube holders, their positions were recorded and the 4 neutron slabs were removed and repositioned.

For each of the six calibration confirmation measurements, an assay mass of U-235 was calculated using the known net specific neutron activity of the WRM standards.

The measurement results are required to meet the Portsmouth performance criteria [10] as follows:

$$\hat{\beta} < 0.3\mu + 2.571\sqrt{s^2/n + s_0^2} \quad \text{(Eq. 11)}$$

where $\hat{\beta}$ is the measurement bias i.e. the difference between the known mass μ and the mean of the six

measured results, s_0 is the uncertainty in the known mass of the WRMs and s is the standard deviation of the *n* measurements (*n*=6 in this case). The 0.3 factor is a thirty percent bias defined by FBP. The factor of 2.571 is the t-test parameter associated with a 95% confidence level and five degrees of freedom.

The uncertainty in the WRM certified masses was 0.08 g U-235. However, based on the prior measurements of the surrogate cooler with no sources present, the surrogate cooler is estimated to contain a small amount of uranium hold-up equal to 5.8 ± 2.3 g U-235. Using a value of 5.8 g for s_0, the measurement bias must be less than 57.6 g to pass the calibration confirmation.

The mean of the six measured masses was 153.8 g U-235 with a measured standard deviation of 3.6 g. Thus the measurement bias (13.0 g) was significantly below the site's required upper limit. The standard deviation of the six measurements relative to the known mass was 2.57%, which was also lower than the site's required upper limit in precision of 43.45% [10] to pass the test. The FBPSC therefore successfully passed both the bias and precision requirements for calibration confirmation.

SYSTEM USE TO DATE

The operation of the system has proven to be very efficient, with the short measurement time providing capability to perform a high throughput rate of coolers through the Blue Box. Significant operational benefits and improved data traceability has been yielded compared to the previous measurement processes. As of August 2014, this system has successfully measured all coolers required to support completion of a contract required performance milestone. There are approximately 1900 coolers available for measurement.

FUTURE PLANS FOR UPDATING PORTSMOUTH SYSTEMS

The MDA of the system could be improved by introducing some simple corrections to the background rate. For example, it is known that barometric pressure has a significant influence on the magnitude of neutron background at ground level [7]. Barometric pressure is an indicator of the overhead air mass which acts as a cosmic ray shield. A correction based on measurement of barometric pressure at the time of assay compared to the time of background measurement could be a very simple method to reduce MDA which would, in turn, further improve operational throughput by providing reduced count times.

The MCNP model that has been developed for the FBPSC can be readily adapted to determine calibration and MDA coefficients for new cooler sizes or additional D&D waste items with similar size and elemental composition to the coolers. This allows FBP to rapidly expand the operational range of the system to encompass new waste streams without the need to build expensive new surrogate matrices.

CONCLUSIONS

A new neutron slab counting system, the FBPSC, has been jointly developed by PSC and FBP for use in a specially designed shielded enclosure on the Portsmouth This system performs characterization of D&D waste streams. The system has been calibrated and tested, and is now in full time operation, performing measurements on coolers.

An MCNP model has been developed for the FBPSC that has been successfully benchmarked with a campaign of on-site measurements using Cf-252 sources and a surrogate cooler. The model was used to determine calibration coefficients for a wide range of cooler sizes. The Monte Carlo method has yielded savings in cost and schedule allowing FBP to rapidly configure the system to count new waste streams as D&D of the facility progresses.

The total systematic uncertainty has been estimated to be ± 30.3% for assay of coolers using the FBPSC. This is dominated by the uncertainty in the declared uranium isotopics rather than from systematic effects related to the neutron measurement itself.

The MDA of the system has been determined to be less the 15g of U-235 for count times that are typically less than 5 minutes (depending on the uranium isotopics). This is primarily the result of the low neutron background rate inside the Blue Box (approximately 3 times lower than at nearby locations outside the box). Additionally, various software features have been designed to yield optimum counting performance for the system including dynamic count time adjustment and interval filtering to reject statistical outliers that are usually associated with cosmic spallation events.

The automatic count time adjustment method eliminates the “over-counting” problem whereby systems that use fixed count time will waste time by counting too long. Instead the system pre-determines the amount of time needed to reach required MDA and stops counting when the MDA has been met. This provides significant improvements in system throughput.

The system has successfully passed the calibration confirmation tests which comprised a series of measurements on uranyl fluoride standards. The system’s accuracy (bias) and precision were shown to be in compliance with the FBP requirements.

Previous measurement campaigns had relied on the use of hand held counter / scalers for the data acquisition with data being manually entered into spreadsheets. By contrast, the new FBPSC system has been demonstrated to provide significant improvements in operational efficiency for the Portsmouth site, enabling rapid characterization of decommissioning waste streams with a high degree of quality assurance.

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