SPENT FUEL MEASUREMENTS TO IMPROVE STORAGE AND TRANSPORT EFFICIENCY

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Abstract

Spent nuclear fuel characterisation measurements can offer significant benefits to a range of fuel handling activities. These activities include (i) the use of burnup credit in storage, transport and disposal, (ii) safeguards operations in which in situ verification is required, and (iii) the determination of radionuclide inventories for direct disposal. The benefits from the measurements may be economic, or they may be required to satisfy regulatory control requirements for the operation of a utility, reprocessing plant or a fuel management system.

To satisfy these diverse measurement requirements, a range of radiometric measurement techniques have been investigated by research organisations and industry in the United States, Europe and the Far East. The techniques include high resolution gamma spectrometry, passive neutron counting and active neutron counting. When used in conjunction with nuclide inventory computer codes, such as ORIGEN or FISPIN, the radiation measurements allow burnup, cooling time, initial wt.% ^{235}U enrichment, residual wt.% ^{235}U equivalent enrichment and radio-nuclide inventories to be determined.

The successful use of characterisation measurements depends on the development of appropriate techniques together with the availability and acceptance of methodologies that cover the measurement process and the related calibration procedures. These are necessary to correlate the measurable radiation emissions with the required spent fuel parameters, such as burnup or enrichment. Consideration of the range of techniques have led to the development of an effective and robust transportable measurement system that may be used for on site spent fuel burnup credit measurements.

1. INTRODUCTION.

To encourage the continued use, and possibly the growth, of nuclear power as an energy provider, it is important to head off criticisms frequently aimed at the nuclear fuel cycle. These are often concerned with (i) the enhanced risk of nuclear weapons proliferation resulting from the increased amounts of plutonium generated from burning low enriched uranium (LEU) fuels and (ii) the lack of a closed cycle in terms of a satisfactory disposal route for the radioactive waste products. Here, as in other areas of the nuclear fuel cycle, the use of better radiometric instrumentation may contribute to the integrity of safeguards operations and the reduction of waste disposal costs while providing a traceable path for materials through the cycle.

Specifically, fuel characterisation measurements can provide a vital supporting role in a range of fuel handling activities including; (i) the use of burnup credit in storage, transport and disposal operations, (ii) the in situ verification of burnup and fissile content particularly for mixed oxide (MOX) fuels for safeguards and, (iii) the determination of radionuclide inventories for direct disposal of spent fuel or wastes resulting from reprocessing.

Better instrumentation may result from improvements in the way technology is applied to practical measurements, through to achieving greater measurement sensitivity and accuracy by using improved detectors and data processing technology. The focus of this paper is however, a discussion of the way measurements may be applied to achieve the most useful data in support of burnup credit applications along with supplementary examples of measurements associated with the other two application areas namely safeguards and waste disposal.

2. FUEL CHARACTERISATION MEASUREMENT APPLICATIONS

2.1 **Burnup Credit**

Taking account of the reduction in the neutron reactivity (multiplication) of spent fuel compared to that of fresh fuel in criticality assessments, is known as burnup credit. The reduced reactivity being of course a consequence of the net loss of fissile and fissionable nuclides together with the generation of fission product poisons that occur during the fuel's irradiation in a nuclear reactor.

The fuel nuclides of major criticality importance were identified in an International Study on Burnup Credit [1]. These are the fissile and fissionable nuclides; uranium 235, 236, and 238, and plutonium 239, 240 and 241. The major fission products were also listed as; ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm and ^{152}Sm.

The "unused" difference in reactivity between fresh and spent fuel expressed as burnup credit offers the nuclear industry a means of increasing the packing density of spent nuclear fuel in storage racks as well as in transport and disposal casks. Alternatively, it can allow a reduction in the amount of expensive neutron absorbers required in those containers. The present, very conservative, method of using the un-irradiated or fresh fuel reactivity for spent fuel in criticality cask design calculations, known as the "fresh fuel assumption", leads to unnecessarily over-engineered and expensive cask designs of limited packing density. In anticipation of licensing approval of a burnup credit methodology, cask vendors are producing cask designs based on the reduced reactivity values offered by burnup credit.

In the United States the Nuclear Regulatory Commission (NRC) controls the issue of licenses for spent fuel casks in accordance with the requirements of Title 10 to the Code of Federal Regulations (CFR), Part 72 (Storage), Part 71 (Transportation), and Part 60 (Disposal). A program to change the licensing policy to one in which burnup credit can be used is being pursued by the United States Department of Energy (USDOE) through their series of topical reports on PWR actinide only[1] burnup credit [2,3,4][2]. The reports propose a methodology for the application of burnup credit. This is encompassed in five major steps:

"1. Validate a computer code system to calculate isotopic concentrations in the spent nuclear fuel created during burnup in the reactor core and subsequent decay.
2. Validate a computer code system to predict the subcritical multiplication factor, k_{eff}, of a spent nuclear fuel package.
3. Establish bounding conditions for the isotopic concentration from criticality calculations.
4. Use the validated codes and bounding conditions to generate storage, transportation, and disposal package loading criteria (burnup credit loading curves).
5. Verify that spent nuclear fuel assemblies meet the package loading criteria and confirm proper fuel assembly selection prior to loading."

The last of the steps introduces the need to confirm the reactivity of spent fuel; and will be most likely required to be via the measurement of burnup (the need to measure is in fact stated explicitly in the NRC Interim Staff Guidance IG-8S). Such verification measurements are aimed at enhancing the administrative control to ensure beyond any doubt that fuel loaded into a cask is fully compliant with the prescribed burnup credit loading curves. In addition, the measurements may assist in the confirmation of each assembly's identification by verifying other fuel history parameters.

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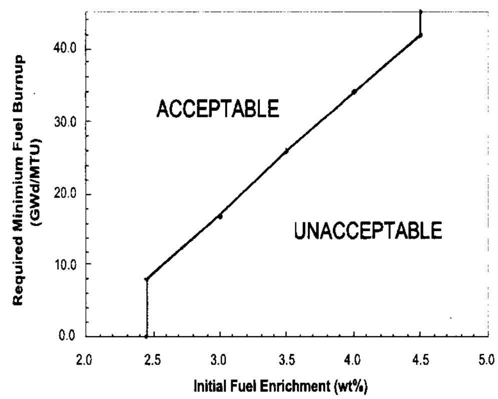
[1] Consideration of fission products as neutron poisons is not included. Only the following actinides, and their effect on neutron reactivity are considered: ^{234}U, ^{235}U, ^{236}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu and ^{241}Am.

[2] Acceptance of limited or partial actinide only burnup credit in criticality safety analyses of PWR spent fuel in transport and storage casks has been accepted by the NRC by issue of the Interim Staff Guidance – 8 titled "Limited Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks", IG-8S, May 1999. In this allowance is given for 50% of the actinide only burnup credit worth based on the procedures set out in the DOE Rev. 2 Topical Report [4].

The burnup credit loading curves, described in the topical reports, provide a means of segregating fuel assemblies into "specified" assemblies, that meet the acceptance criteria for loading into a particular fuel storage rack or transport cask designed to take account of burnup credit; and "non-specified" assemblies that do not meet the criteria. The criteria are based on a combination of an assembly's fuel burnup and wt.% ^{235}U initial enrichment. Figure 1 shows an example of a loading curve.

Built into the curves are biases to account for any uncertainties in the data that relate burnup to the reactivity of the spent fuel. However, before using a cask loading curve to determine the loading status, it is necessary to determine the assembly's minimum assured burnup. The minimum assured burnup being the "actual" burnup minus the uncertainty on this value.

FIG. 1 *Typical fuel loading curve*



Based on the increases in cask capacities from the employment of burnup credit, significant commercial and operational advantages are anticipated giving in the region of 25% to 40% reduction in handling costs [5]. The USDOE estimate spent fuel transport cost savings of 35% using a 4 PWR assembly truck cask with actinide only burnup credit rising to 40% for full (principal isotope) burnup credit [6]. For rail transport, in which the anticipated unit costs are considered to be lower than those for truck due to the possibility of using larger casks of 21, 24 or 32 PWR assembly capacity, cost savings of up to 26% are anticipated. Depending on the mix of transport modes, the overall cost savings for transport of fuel from utility to repository is estimated to be between $200M to $1b if full burnup credit is used. These figures are based on transporting 126,000 PWR assemblies in a mixture of General Atomics GA-4 truck casks and 24 to 32 assembly capacity rail casks.

Although cost savings are expected from lower cask costs and fewer transport journeys, other cost factors have to be considered to determine the total net savings. These factors include; (i) the value of radiometric measurement data acquired prior to shipment for disposal needs. This could eliminate the need to re-open casks at the final repository for measurements that may be required to satisfy waste acceptance criteria, (ii) implementation costs of burnup credit in terms of license approval and administration, (iii) the cost of radiometric measurements and (iv) the amount of burnup credit that may be used , i.e. actinide only or full burnup credit with the inclusion of fission product poisons. The poisons can reduce the multiplication, k_{eff}, by approximately a further 10% beyond that achieved via actinide only credit.

2.2 Application of Burnup Credit Measurements

Verification of the "candidate" fuel assemblies, i.e. those expected to be above the loading curves based on the reactor operator records, is anticipated to be made by physical measurement. The procedure detailed in Rev. 1, and referenced in the latest topical report, Rev.2 describes the use of a rejection criterion to judge whether the measured burnup of an individual fuel assembly is consistent with that declared in the reactor records. Rejection would result in the assembly being disqualified for loading into a burnup credit cask.

The specification of the rejection criterion can be used as a good illustration of the difficulties involved with expressing and utilising the uncertainties in two data sets when one is intended to verify the other. In this case, the two sets are the burnup values declared by the reactor operator and those derived from measurement.

In the Rev.1 topical report, the proposed criterion is that "the measured burnup must be within 10% of the reactor record burnup" with the minimum assured burnup defined as the reactor record value minus its reactor record declared uncertainty. The role of the measurement is therefore only to confirm the validity of the reactor record with the reactor derived minimum assured burnup then being applied to the loading curves. Using this approach, any disagreement between the measurement and the reactor record, beyond 10% is to be used as an indication that something is wrong and to reject the assembly. The question then arises as to whether an unnoticed error of 10% would lead to an unsafe condition. The answer in the topical report is that approximately half of this difference is accounted for in the reduction of the assembly burnup due to uncertainty in the reactor records, i.e. 5%. However, if the assembly was at the low end of the reactor record uncertainty, the maximum error in burnup could be 10%, comprising perhaps only 1% record and 9% measurement uncertainty. The DOE view this as acceptable (although not yet accepted by the NRC) because there is a significant conservatism in the reactivity of the assemblies from fission products that are not accounted for in the derivation of the loading curves in the actinide only burnup credit proposal.

As part of this process the calibration of the measurement system needs to be considered. A calibration derived from the correlation between a measurable parameter, e.g. the activity of the fission product Cs-137 or the neutron emission rate principally from Cm-244, and the declared burnup for a representative set of assemblies is known as a "dependent" calibration. The use of such a calibration is viewed as appropriate for the application of burnup credit to commercial fuels because of the general acceptance that for a group of assemblies representing a reactor core there is very little, if any, systematic bias in the declared burnup. The proposed acceptance criterion to be used to qualify the dependent calibration and determine each assembly's status is as follows:

(i) A calibration curve of the following form is to be derived and used to correlate the measured parameter (or count rate) to the reactor record burnup.

$$y_{counts} = a + bx_{reac}$$

where a and b are constants, y_{counts} is the count rate of the measurable parameter, and x_{reac} is the reactor record burnup value. (When using neutron emission as a burnup indicator a non-linear expression may need to be substituted).

(ii) The validity of the calibration is then tested over the entire range of x by applying a 10% limit to the count rate Prediction Band Width [7], i.e.:

Prediction Band Width (converted to units of burnup) / Assembly Burnup < 0.1

where the Prediction Band Width (count rate) =

$$\left\{ \sqrt{(n+1)/n + \frac{(x-\bar{x})^2}{S_{xx}}} \cdot \sqrt{\frac{SS_R}{(n-2)}} \cdot t_{\frac{\alpha}{2},n-2} \right\}$$

$$S_{xx} = \sum_{i=1}^{n} (x_i - \bar{x})^2$$

$$SS_R = \sum_{i=1}^{n} (y_i - ax_i - b)^2$$

n is the number of assemblies in the calibration set

and $t_{\alpha/2,n-2}$ is the t-statistic at the 100(1-_)% confidence level for n-2 degrees of freedom (_ = 0.05 for 95% confidence)

The test therefore defines a range of x for which the inequality holds and the calibration is valid. (Procedures for dealing with the ranges of x that do not satisfy the inequality are suggested in the Topical. Splitting the calibration range into smaller groups each with their own calibration is an option).

(iii) For an assembly to be accepted for loading, the difference between the measured burnup derived via the measurement with the validated calibration and the declared burnup must be less than 10% of the declared burnup.

The above procedure for measurement and verification of the reactor records appears to be well founded and workable. The implicit assumptions are that the reactor records data has for each assembly approximately a 5% uncertainty in burnup and that the measurement may also contribute a 5% uncertainty. The difficulty, however, occurs when the uncertainty values, taken to be at a 95% confidence, do not meet the arbitrary value of 5%. In particular the reactor records, though accepted to be of a good accuracy with an average uncertainty at 1 sigma of 2% across a reactor core, may for individual assemblies be somewhat greater than 5%. The net result is that either the calibration may not meet the test criterion or some of the individual assemblies may fail the 10% test. If either of these occur, the identified assemblies may be disqualified from being loaded into a burnup credit cask, even though as may be the case, the burnup is well below that prescribed by the applicable loading curve.

An alternative procedure, proposed for discussion in this paper, recommends that measurements should play a greater role in the process of determining the minimum assured burnup for each assembly. This procedure is as follows:

1. Calibrate the measured burnup indicator against the declared burnup records. Compared to the above procedure the linear expression is inverted to give;

   $$y_{reac} = a + bx_{count}$$

   in which y_{reac} is the declared burnup and x_{count} is the count rate of the measured burnup indicator.

   The calibration set is recommended to be consistent, in number of assemblies, with a reactor core load of fuel comprising approximately 200 or more assemblies. This calibration should be carried out before commencing fuel loading.

2. Check the calibration data set for outlier assemblies. In this case an outlier assembly is defined as one for which the difference between the declared and the measured burnup is greater than a pre-defined percentage[3]. This is to eliminate assemblies that are clearly badly measured or incorrectly declared.

3. If any assembly is identified as outlier it should be removed from the calibration data set. The assembly reference number of the rejected assembly should be recorded pending an investigation into the causes of its outlier status. The investigation may include further measurements and other checking procedures. Failure to evaluate and rectify the cause of an outlying assembly will make that assembly ineligible for burnup credit.

4. If any assembly is rejected during step 2 and 3 then a new reduced calibration data set will be used to recalibrate the burnup indicator.

5. Steps 1 through to 4 are repeated until there are no rejections identified at step 3.

6. Determine the assembly burnup, *y* using the measured burnup indicator in conjunction with the established calibration curve for each of the assemblies that remain in the calibration data set and where appropriate the burnup of other assemblies in a larger measurement campaign.

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[3] Identification of outliers can be based either on data points that fall outside a specified confidence interval, or, as in this case outside a specified percentage range. The choice of a fixed percentage is suggested to ensure that the probability of assembly's rejection is lower for good quality calibration data sets in which the amount of scatter is small. With this approach it is possible that there are no rejected assemblies. If on the other hand a confidence limit, derived from the scatter in the calibration set, is chosen, there will always be a fixed proportion of the set rejected regardless of the quality of the data.

7. Determine the uncertainty on each of the measured burnup values by propagating the uncertainty in the calibration and the uncertainty in the individual measurement of the burnup indicator.

The component of the uncertainty in y from the scatter in the calibration data set, is:

$$\left\{ \sqrt{\frac{1}{n} + \frac{(x-\bar{x})^2}{S_{xx}}} \cdot \sqrt{\frac{SS_R}{(n-2)}} \cdot t_{\alpha,n-2} \right\} \qquad (1)$$

where

$$S_{xx} = \sum_{i=1}^{n} (x_i - \bar{x})^2$$

$$SS_R = \sum_{i=1}^{n} (y_i - ax_i - b)^2$$

and $t_{\alpha,n-2}$ is the t-statistic at the 100(1-_)% confidence level for n-2 degrees of freedom.

The overall uncertainty σ_y (shown at 95% confidence level) is calculated at a stated confidence level based on the uncertainty in the measured count rate, σ_x and the scatter in the calibration data set:

$$\sigma_y = \sqrt{(a.1.65.\sigma_x)^2 + \left\{ \sqrt{\frac{1}{n} + \frac{(x-\bar{x})^2}{S_{xx}}} \sqrt{\frac{SS_R}{(n-2)}} t_{\alpha,n-2} \right\}^2} \qquad (2)$$

8. Calculate the minimum assured burnup for each fuel assembly by decreasing the measured burnup by its total uncertainty to a specified confidence level. The confidence level, to be defined by the regulators, will ensure that the reduced burnup value gives a minimum assured burnup at the required level of confidence. (Expression 2 above gives the total uncertainty to 95% confidence using a t statistic with $\alpha = 0.05$ and a multiple of 1.65 for the uncertainty on the individual measurement. Note the value of 1.65 assumes that the uncertainties are well described by a Gaussian distribution). From equations (1) and (2) the minimum assured burnup, M_{BU} at the specified confidence level is given by

$$M_{BU} = y - \sigma_y$$

9. Compare the minimum assured burnup, as defined by the measured burnup and its associated uncertainty, with the cask loading curve for each assembly to establish its loading qualification.

This proposed methodology has several beneficial features that are not apparent in methods that use the burnup measurement purely as a verification of the declared burnup. Firstly, it is a very simple method that does not rely on any arbitrary assumptions about the scatter of the declared data set used during the production of the calibration (other than the initial rejection criteria of badly measured or declared data). Secondly, it is capable of providing a determination of the uncertainty in burnup for each individual fuel assembly.

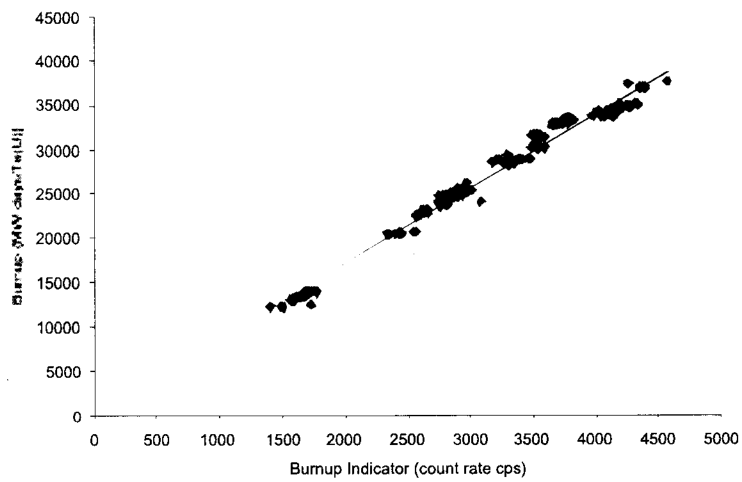
Though as with any dependent calibration this approach relies on the accepted position that the operator declared values for burnup have, when taken en-masse, negligible systematic error. This is commonly viewed as a key strength of the declared data that enables an unbiased dependent calibration to be defined. The weakness in the reactor records is that the uncertainty in the burnup associated with individual assemblies is often undetermined. This weakness is overcome by the use of the declared data with the measured data as outlined by this alternative proposed methodology. The principal improvement stems from the use of a verifiable measure of the burnup and its associated uncertainty for each fuel assembly.

It should also be noted that this approach takes credit both for the quality of the declared burnup records and for the precision of the measurements. The better these are, the greater will be the minimum assured burnup for each assembly. This, in turn, means that the number of assemblies that qualify for burnup credit loading may be increased.

In summary, it is suggested that this alternative measurement based approach offers a more realistic determination of minimum assured burnup for each assembly. Their values are likely to be higher, and hence of greater economic value, than those derived from a method that utilises an assumed operator declared uncertainty for each assembly. It is anticipated that this latter value would have to be fairly pessimistic to ensure that the maximum uncertainties in the records are assumed for each assembly. This would result in lower minimum assured burnup values at the required level of confidence.

The proposed methodology has been tested on a campaign of commercial PWR spent fuel. The campaign comprised 203 assemblies measured in the U.S. in 1997. In these the burnup was measured using the burnup indicator Cs-137 corrected for cooling time and axial burnup profile to give the assembly average burnup. Figure 2 shows the measurement data used to determine a calibration for the burnup indicator. Table 1 shows the derived minimum assured burnup along with the operator declared burnup for a set of 31 assemblies chosen randomly from the 203 assemblies in the campaign. The effectiveness of the approach is demonstrated by the relatively small amount that the minimum assured burnup is below the declared burnup. For the 31 assemblies in the table this is 4% ± 3%.

FIG. 2 *Calibration of Cs-137 662 keV gamma ray emission burnup indicator for a campaign of 203 assemblies.*



2.3 Safeguardi

As the global quantity of spent nuclear fuel steadily grows the need for rigorous control of the large quantities of fissile nuclides, predominantly ^{235}U, ^{239}Pu, and ^{241}Pu, within the fuel is becoming increasingly important from a safeguards standpoint. Plutonium content represents about 1% of spent fuel assembly mass. Globally the current stocks of more than 150,000 t HM in spent fuel assemblies contain more than 1000 tonnes of plutonium. The amount accumulated through the lifetime of the currently operating reactors may rise by a factor of 2 or 3 depending on the quantities reprocessed and recycled in the form of mixed oxide (MOX) fuels.

Measurement and verification of such large quantities of plutonium and fissile uranium within spent fuel assemblies, beyond the level of simply item counting, are potential requirements. If so, rigorous measurement methodologies, similar to those proposed for burnup credit, will be equally relevant to safeguards for fissile material quantification or verification. Improved safeguards measurements may not only be of benefit to aid non-proliferation but could enhance the public acceptability of handling and transportation of fissile materials. For example, in the U.S. this may improve the prospects for conversion of DOE owned fissile material into commercial fuel for burning as MOX. Equally, improved faith in safeguards could encourage the earlier transfer of material from DOE sites and power utilities to a long term repository.

TABLE 1 *A random selection of 31 assemblies showing the derived minimum assured burnup compared to the operator declared burnup.*

Assembly	Burnup Indicator		Burnup {MWd/t(U)}		
Reference	Value	Uncertainty	Declared	Measured	Minimum Assured
2	1504	41	12310	12827	12221
5	1773	46	13946	15124	14455
11	2753	69	23509	23487	22505
19	2648	67	23066	22590	21645
27	2744	69	24221	23405	22421
33	1773	46	14173	15127	14458
44	1666	44	13650	14209	13567
50	2342	60	20444	19973	19123
51	2896	73	25001	24700	23663
58	1640	43	13493	13988	13347
61	2640	67	22815	22520	21572
67	2888	73	25250	24635	23595
74	2823	72	24792	24076	23059
79	2850	73	25212	24313	23285
83	2947	74	25350	25136	24087
90	2935	74	25236	25032	23984
96	2828	72	24304	24126	23113
99	2592	66	22572	22111	21177
113	3655	90	33075	31174	29898
119	3778	93	33733	32228	30909
123	3693	91	32736	31498	30214
132	3663	90	32647	31248	29981
142	3412	84	28801	29104	27914
153	3277	81	28434	27954	26802
157	3279	83	28657	27969	26801
170	4104	101	34043	35010	33581
176	4141	102	34247	35322	33879
184	4241	104	35001	36177	34703
192	4216	104	34766	35966	34498
198	4568	113	37797	38963	37362
202	2895	73	24891	24695	23657

The above table contains a subset of data from a campaign of 203 assembly measurements. The minimum assured burnup is calculated in each case from the measured burnup indicator and its associated uncertainty and the uncertainty derived from the dependent calibration. The following are the calculated terms used in the uncertainty analyisis:

The number on assemblies measured during this campaign, $n=203$
The average burnup indicator, $\xi=3073$ The calibration parameters, $a=8.53$, $b=0$
Summations of the residuals, $SS_R=134026293$, $S_{xx}=138643969$
The one tailed t-statistic for n-2 degrees of freedom at the 95% confidence interval, $t_{0.05,\,n-2}=1.65$

2.4 Measurements in Support of Safeguards

The application of measurements to safeguards, in contrast to burnup credit, is likely to require direct measurement of fissile content. For burnup credit, though fissile content is the key parameter for the purpose of criticality calculations, the use of a burnup measurement (burnup verification) in combination with a given initial U-235 enrichment is generally accepted for commercial power generation spent fuel as a means to assess fissile content. Such an arrangement for safeguards would not guarantee diversion detection.

Direct measurement of fissile content with an ability to discriminate between fissile uranium and plutonium may also be required. The candidate measurement approaches are therefore likely to comprise a combination of active and passive neutron techniques with gamma spectrometry.

As for burnup credit, the question arises, as to the most appropriate method of calibration of safeguards instruments. Clearly a calibration independent of any operator declared information would be preferred. Though at first this appears quite straightforward, it is difficult to fully achieve because of the strong dependence of a fissile measurement on the fuel geometry as well as the fissile content. The standard approach would be to calibrate the measurement systems by modelling or simulating the given material and measurement geometry. During safeguards measurement it is therefore important that the geometry is as expected, otherwise the measurement could be invalidated or at least inaccurate. This is an important aspect of safeguard monitoring, as good knowledge of measurement uncertainty is key to gaining high confidence in an ability to detect material diversion.

A possible solution could be to combine the radiometric techniques (active and passive neutron counting and gamma spectrometry) with a means of confirming the geometrical arrangement, using for example real time radiography (RTR). This approach could also offer the ability to correct a measurement for the effects of damaged fuel or research fuels for which detailed geometrical information may be lacking. Currently an instrument offering this combination of techniques is not yet available but could be developed in response to the demands of the safeguards regulators.

2.5 **Spent Fuel Waste Disposal**

Under present policies a significant proportion of the world's commercial spent fuel is viewed as waste. Although the waste in the UK is not in general spent fuel, but industrial radionuclides and residues from reprocessing, there is a requirement to measure (or infer from measurement) some 78 radionuclides. Similar requirements for radionuclide content assessment are therefore likely to be demanded for spent fuel disposal. The measurement of burnup and associated irradiation history parameters such as cooling time could be used, as it is for waste under the U.K definition, to provide the required radionuclide inventory data for spent fuel.

2.6 **Measurement Techniques and Methodologies.**

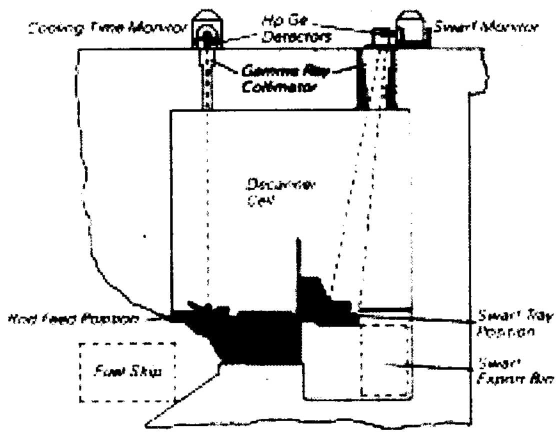
Available techniques include high resolution gamma spectrometry, passive neutron counting and active neutron counting. When used in conjunction with nuclide inventory computer codes, such as ORIGEN or FISPIN, the radiation measurements allow burnup, cooling time, initial wt.% ^{235}U enrichment, residual wt.% ^{235}U equivalent enrichment and radionuclide inventories to be determined for intact fuel assemblies and for dismantled assemblies or fuel debris.

The successful use of the characterisation measurements depend, in a similar way to the other applications outlined above, on development of appropriate techniques together with the availability and acceptance of methodologies that cover the measurement process and the related calibration procedures. These are necessary to correlate the measurable radiation emissions with the required spent fuel parameters, such as burnup, and will be essential to the regulatory control of spent fuel contaminated waste destined for interim or final disposal.

A range of modular spent fuel monitoring systems for fuel characterisation has been developed by BNFL Instruments. In the main these have been developed from instrument systems produced for reprocessing facilities at Sellafield; their primarily roles being related to process control, radionuclide inventory assay and safeguards applications. The systems use a variety of radiometric techniques along with different approaches to calibration and validation procedures necessary to ensure reliable and accurate operation that is appropriate to the customer requirements.

An important example of one of the first industrial nuclear waste monitoring systems is the Swarf Inventory Monitor used in the UK reprocessing plant at Sellafield. This was introduced in the 1980s, and is still in operation, for measurement of both fuel content and a broad range of radio-nuclides, primarily fission products, within the waste material. The waste material in this case being the fuel rod container material made from magnesium alloy (Magnox); and the fuel rod being natural uranium metal used to power the Uk's first generation of graphite moderated power stations. Figure 3 shows a sectional schematic view of the Swarf Inventory Monitor (SIM) along with a parallel instrument for the measurement of cooling time of irradiated fuel rods before decanning (container removal).

FIG. 3 *Mechanical Arrangement of the FHP Cooling Time Monitor and Swarf Inventory Monitor.*



The high-resolution gamma detector views the Magnox swarf on a sorting tray through a roof collimator. The uranium fuel mass and radio-nuclide inventory are calculated from the measured activities of the radio-nuclides listed in table 2. These along with other fission product radio-nuclides entrained in fuel fragments and ^{60}Co activation in steel components such as nimonic springs give rise to the observed gamma spectra. Having corrected the gamma spectra for background activity determined from an empty tray measurement following each swarf measurement, the net photopeaks, within the energy range 500 - 1600keV, are determined.

A relationship between irradiation and Cs-137 specific activity (Bq/g(U)) derived from the nuclide inventory code, FISPIN(8), is used with the measured Cs-137 and irradiation determined from the Ru-106 x Cs-137/(Cs-134)^2 activity ratio to give fuel mass.

The cooling time, which is needed to make a small correction to the irradiation and in the inferential determination of the non measurable fission product radio-nuclides, is calculated from the two activity ratios Zr-95/Ce-144 and Cs-134/Cs-137. The activities of 29 non-measurable radio-nuclides are inferred by the use of look-up curves which correlate their activity with that of Cs-137 and the measured irradiation and cooling time.

TABLE 2 *Radio-nuclides measured by the SIM*

Nuclide	Half-life	Energy of principal gamma rays (keV)	Branching. ratio (%)
^{95}Zr	64.0d	724.2	44.1
		756.7	54.5
^{95}Nb	35.0d	765.8	99.8
$^{106}Ru(Rh)$	1.02y	511.9	20.7
		621.9	9.8
^{134}Cs	2.062y	569.3	15.4
		604.7	97.6
		795.9	85.4
		802.0	8.73
		1365.2	3.04
^{137}Cs	30y	661.7	85.2
$^{144}Ce(Pr)$	285d	696.5	1.34
^{154}Eu	8.8y	723.4	19.7
		1004.8	17.9
		1274.5	35.5

Other features incorporated in the radiometric method are an energy dependent relative gamma detection efficiency correction and a self attenuation correction to correct the Cs-137 photopeak detection efficiency for the effect of different sizes of fuel debris. This latter routine uses a "Newton-Raphson" iterative technique which gives a representation of the distribution of fuel sizes present on the tray by a mix of two particles, one of zero thickness, i.e. fines offering no attenuation to the emitted gamma rays, and the other of a finite thickness up to the approximate diameter of a fuel bar (30mm). The results of a performance assessment have shown that for over a million rods measured to date the accuracy of fuel mass determination as assigned to the exported swarf bins is in the range of ± 8%.

3. CONCLUSIONS

It is clear that radiometric measurements play an important role in a range of fuel handling activities and can offer benefits that may be financial or safety related. Before applying measurements to fuel handling tasks there are a number of key questions that should be answered during an instrument's design. Such questions include; what measurement technique should be used to balance cost against performance, what approach should be taken to calibration, what operator supplied data can be depended on to supplement or aid calibration and measurement procedures.

With respect to burnup credit significant benefits may be gained by choosing a particular approach to calibration and deployment as illustrated by the measurement example in the report. The resulting minimum assured burnup values offer significant potential cost saving through a more efficient application of burnup credit while maintaining a high degree of safety.

4. REFERENCES

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