Burnup credit measurements for cask loading compliance – a review of techniques and calibration philosophies.

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ABSTRACT

Many techniques and technologies have been developed and field tested for the purposes of spent nuclear fuel burnup credit measurements. The collection of data from such measurements is generally regarded as a prerequisite to demonstrate compliance with fuel cask loading curves during fuel loading operations. However, there is still debate not only on whether, in all cases, measurement is compulsory but also on the merits and performance capabilities of the range of techniques and calibration philosophies that may be applied.

Examples of the principal techniques are presented and discussed along with their calibration methodologies. The advantages and disadvantages of dependent and independent calibrations are considered in the light of the accuracies that may be achieved and the magnitude of the work needed to authenticate the calibration data. For a particular measurement technique the calibration development may include Monte Carlo and/or analytical modelling as well as experimental validation. The accuracies of radiation modelling codes and the burnup computer codes ORIGEN and FISPIN will also need to be considered if an independent calibration approach is adopted.

The question of whether a more direct measure of a k_{eff} indicator by means of fissile content assay could or should be adopted rather than the use of the indirect k_{eff} inference from the burnup measurement approach is also discussed and evaluated.

Based on a consideration of the above factors, recommendations are made on the most practical and efficient approaches to ensure criticality safety for spent fuel cask loading, storage and transport applications.

INTRODUCTION

The reduction of neutron reactivity (multiplication) that occurs during nuclear fuel irradiation is from the net loss of fissile and fissionable nuclides together with the generation of fission product poisons. The 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 stable (or longer lived) neutron poison fission products were also listed, these are; Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Sm-147, Sm-149, Sm-150, Sm-151 and Sm-152. Xe-135, although an effective neutron poison, is excluded as it has only a short half life < 9.1h.

Burnup credit (BUC) is a process under which advantage or credit may be taken for this reduced neutron reactivity. As a consequence this offers a means to increase cask loading or reduce the required shielding requirements and hence offers cost savings for fuel loaded into burnup credit casks. The cask design principles are consequently different from the, non burnup credit, conservative approach of using the un-irradiated or fresh fuel reactivity for spent fuel in criticality calculations, known as the “fresh fuel assumption”. This latter approach naturally leads to unnecessarily over-engineered and expensive cask designs of limited packing density.

Cask designs and their accompanying spent nuclear fuel (SNF) cask loading criteria are sanctioned or approved by the Nuclear Regulatory Commission (NRC) under Title 10 to the Code of Federal Regulations (CFR), Part 72 (Storage), Part 71 (Transportation), and Part 60 (Disposal) and supplemented by Guidance notes such as ISG-8 [2] and NUREG publications.

Once a cask criticality design has been approved, the principal control mechanism during spent fuel cask loading is ensured by adherence to the related cask loading curve; an example curve is shown in figure 1. This segregates the spent fuel assemblies into "specified" or acceptable assemblies that meet the acceptance criteria for loading and "non-specified" assemblies that do not meet the criteria. The criteria are based on a combination of fuel burnup and wt.% U-235 initial enrichment.

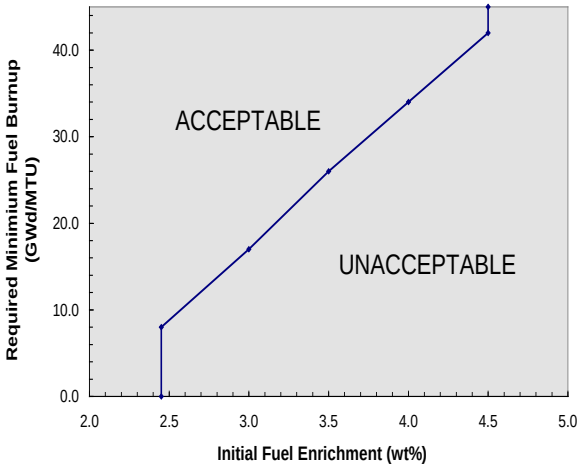

Fig. 1 Typical spent fuel cask fuel loading curve

Since the use of burnup credit may be seen as an erosion of the criticality safety margins, built into the curves are biases or conservatisms in the analysis and modelling parameters to account for uncertainties in the data that relate burnup to the reactivity of the spent fuel [3]. Additionally, to ensure that the correct burnup data associated with fuel assemblies selected for loading is used, there is a requirement, recommended in NUREG/CR-6998 [4] to carry out "out-of-core" burnup verification measurements. In the NUREG document, significant attention is drawn to the need to apply appropriate consideration, in terms of assessment and application, to the associated measurement uncertainties and their impact on the fuel loading criteria.

The unavailability of a US federal permanent solution for SNF storage, is forcing nuclear power plant (NPP) operators into making arrangements for temporary storage of spent fuel in Independent Spent Fuel Storage Installations (ISFSIs). The first US commercial ISFSI was licensed by the NRC in 1986 at the Surry Nuclear Plant in Virginia. As of June 2009, according the NRC website (http://www.nrc.gov/waste/spent-fuel-storage/locations.html) , the number of licensed ISFSI sites had risen to 54 comprising 39 general license ISFSIs and 15 site-specific license ISFSIs.

Because of the need to ultimately transport the ISFSI spent fuel to a federal repository[1] (or reprocessing facility) at some indeterminate date, it is considered prudent, under the terms of the current 10CFR72 that the SNF contained in the dry casks be compliant with both storage and interstate transport regulatory requirements. Under ISG-8, SNF assemblies loaded into "burnup credit" dry casks that are to be used for transportation, require a process of burnup verification. Further it is anticipated that an empirically based verification system will be demanded by the regulators consistent with the regulatory policies set out in

[1] It is believed by some commentators [5] that a US Federal Repository for nuclear spent fuel may never come into existence on the basis that the DOE has failed to deliver this over the last decade or more. The main reason given is that the "State veto power over siting a storage facility makes approval of a facility essentially a national referendum on nuclear power, given that a veto must be overridden by the Senate and the House. Also, the extremely long period of time required to develop any storage facility would certainly span presidential administrations of both political parties, making any project like Yucca Mountain susceptible to closure when the political winds change". The reference suggests that reprocessing should be the preferred solution to the "disposal" of spent nuclear fuel.

NUREG/CR-6998; reference ISG-8 [2] “”administrative procedures should include a measurement that confirms the reactor record for each assembly”.

The US policies and approaches to the use of burnup credit for storage and transport are generally consistent with those of the International Atomic Energy Agency (IAEA). In its TECDOC 1547 [6] the key steps in the application of BUC to a spent fuel management system have been independently defined as:

- *Safety assessment of the system including*
  - *prediction of the spent fuel composition under bounding depletion conditions,*
  - *criticality calculation and evaluation of the fuel loading criterion for the system.*
- *Application of the fuel loading criterion; this step consists in*
  - *quantification and verification of the numerical value which the fuel to be loaded in the system has for the safety parameter chosen to present the loading criterion (e.g. average burnup, see above),*
  - *implementation of a fuel loading procedure assuring compliance with the loading criterion.*

The same document defines the data requirements needed to predict the isotopic inventory of the spent fuel by means of depletion calculations:

- *definition of the fuel characteristics*
- *knowledge of the irradiation history of the fuel*
- *choice of the cooling time.*

BURNUP OUT OF CORE MEASUREMENT

Any assay system selected to demonstrate compliance with an approved burnup credit loading curve should satisfy the requirements of NUREG/CR-6998 [4] for burnup credit empirical verification systems. The loading curve would normally form part of the suite of technical requirements information contained in the Certificate of Compliance (CoC) for a particular storage or transport cask.

The principal NRC topics explored in the NUREG document for the use of BUC measurement systems relate to; (i) the relationship between the measurable radiation and burnup, (ii) the system measurement accuracy and precision, (iii) the impact of the fuel assembly axial (and radial) irradiation profile, (iv) the potential for operator error, and (iv) costs and risks associated with measurement implementation. To alleviate the latter concern ISG-8, Rev. 2 [2] allows for a measurement regime based on sampling as long as the operator can provide adequate proof that a database has been compiled to justify the procedure in comparison with the measurement of all fuel assemblies.

Measurement Techniques

Burnup is “measured” indirectly via the direct measurement of fuel parameters that may be correlated with burnup. The parameters often referred to as “burnup indicators”, are individual or combinations of radiation emissions from radionuclides built up during irradiation. Candidate gamma emitting radionuclides include; Cs-134, Cs-137, and Eu-154. Other nuclides such as Ce-144 and Ru-106 are less useful because of their short half lives and dependence on reactor power rating, although they can be used in conjunction with some of the longer lived nuclides for cooling corrections. With a well collimated gamma detector system, gamma assay also lends itself to the precise measurement of assembly axial burnup profiles[2].

The neutron emission from Cm-244 has also been used successfully as a burnup indicator with active neutron techniques finding their place as a measure of neutron multiplication that can be correlated to residual fissile content [7].

[2] Methods for evaluating gamma detectors and gamma detector instrument response, particularly for use in investigation emissions from spent nuclear fuel rods. Patent number US6664539.

Historically, the primary techniques for the measurement of the burnup indicators have been based on low resolution gamma spectrometry (LRGS), high resolution gamma spectrometry (HRGS)[3] and passive neutron counting. The principal measurements techniques, radiations and radionuclides, together with their correlation relationships, that are used as burnup indicators include:

(i) *The absolute count rate of the 662 keV gamma rays from Cs-137.* This is attractive because of the simple linear relationship which exists between Cs-137 in spent fuel and burnup. The relationship is linear because Cs-137 is a direct fission product and has an almost equal fission yield from uranium and plutonium. In addition Cs-137 has a half life of 30 years, which renders its production insensitive to variations in reactor power rating and dwell time and it has a low sensitivity to errors associated with its required cooling time correction. However, as this involves an absolute measurement technique a well defined and reproducible geometry must be maintained between the detectors in the monitoring system and the fuel assembly being measured.

(ii) *The activity ratio Cs-134/Cs-137 by gamma measurement.* This ratio technique has advantages over an absolute measurement because of its insensitivity to measurement geometry; although correction for relative detection efficiency as a function of energy is still required. Disadvantages are; (a) the ratio has a 2.2 year half life and therefore strongly affected by cooling time, (b) its correlation with burnup is influenced by the initial wt.% U-235 enrichment and the power rating and (c) its application is limited to fuel with cooling times of less than 20 years due to the decay and disappearance of the shorter lived component, Cs-134.

(iii) The activity ratio Ru-106 x Cs-137/(Cs-134)^2 by gamma measurement. This ratio method is useful in that is relatively insensitive to geometry, and unlike the Cs-134/Cs-137 ratio it is virtually independent of enrichment and rating and is therefore subject to lower systematic errors. The half life of the ratio is 22 years, giving it a relatively low sensitivity to errors associated with cooling time correction. However, due to decay of the short half life component Ru-106, this ratio can be used only on fuel of less than 8 or 9 years cooling.

(iv) The measurement of the passive neutron emission. For fuel assemblies of greater than 15GWd/Te(U) burnup and more than 2 years cooling, the primary neutron emitter is Cm-244 which undergoes spontaneous fission. However, the measured neutron flux depends not only on the Cm-244 content but on the quantities of fission product neutron poisons and on the internal neutron multiplication due to the residual fissile content. Even though there is a complicated relationship between burnup and the measured neutron flux, the passive neutron measurement approach offers the following advantages; (a) it is a very sensitive indicator of burnup with the neutron emitting Cm-244 content proportional to the fourth power of burnup, (b) neutrons emitted from a fuel assembly in water cascade through the assembly by induced fission reactions. This effect both amplifies the neutron flux and samples all the pins in the assembly. This is important for safeguards applications as it is sensitive to missing or removed fuel pins. The measurement, therefore, represents the bulk of the fuel assembly in contrast to the outer two or three pins that are “visible” by a gamma measurement; (c) Cm-244 has a relatively long half life of 18.1 years and as a result has a reasonably low sensitivity to cooling time correction errors. The disadvantages are; (a) the quantity of Cm-244 produced during irradiation is strongly dependent on the wt.% U-235 initial enrichment, (b) the measurement of the neutrons is very sensitive to the geometry or water gap between the fuel and detectors and to the presence of any neutron poisons in the pool water or within the fuel itself, (c) the measured neutron flux is influenced by neutron multiplication. This is a disadvantage as well as an advantage (i.e. offers an enhanced neutron signal) since it complicates the interpretation of the measured signal to give burnup.

A summary of the pros and cons of the above traditional non destruction burnup measurement techniques is presented in table 1.

The above developed techniques have been applied in a number of practical research and commercial devices including; (i) the Fork and Fork+ which contain neutron thermal and epithermal detectors

[3] Room temperature semiconductors are however gaining favour as intermediate or medium resolution gamma spectrometry devices because of their lower cost, low maintenance and tolerance to high dose rate environments.

combined with a cadmium zinc telluride (CZT) room temperature semiconductor gamma detector [8], (ii) the PYTHON device also incorporating passive neutron and collimated gamma counting detectors [9], and (iii) in the UK Thermal Oxide Reprocessing Plant (THORP) installed integrated systems measure burnup using high resolution gamma and passive neutron detection [7]. Silicon carbide (SiC (^{6}LiF coated)) detectors have also been employed in the US by Westinghouse for use in harsh high dose rate environments [4].

Table 1 Pros and cons of various burnup measurement techniques

TECHNIQUE	ADVANTAGES	DISADVANTAGES
Absolute count rate of the 662 keV gamma ray from Cs-137.	Simple linear relationship between Cs-137 and burnup. Half life of 30 years. Insensitive to variations in reactor power rating and dwell time.	Absolute measurement requires a well defined and reproducible geometry between the detectors and the fuel assembly. Samples outer pins only.
The nuclide activity ratio: Cs-134/Cs-137.	This gamma ray ratio method is relatively insensitive to geometry.	2.2 year half life requires significant decay correction and can be applied only to fuel with cooling time < 20 years. Burnup correlation is dependent on initial enrichment and power rating. Samples outer pins only.
The nuclide activity ratio: ^{106}Ru x ^{137}Cs/(^{134}Cs)^{2} .	Insensitive to geometry. Independent of enrichment and rating.	Only useful for fuel < 9 years cooling time (Ru-106 has a 372 day half life). Samples outer pins only.
Passive neutron measurement (predominantly from Cm-244).	The neutron signals are received approximately uniformly from all pins in the assembly (gamma measurements are only sensitive to the outer pins). Good for safeguards applications, as it is sensitive to missing or removed fuel pins.	The abundance of Cm-244 is a strong function of initial enrichment. Neutron assay is very geometry sensitive and is also affected by multiplication and neutron poisons in the pool or within the assembly.

CALIBRATION OF BURNUP MEASUREMENT SYSTEMS

Traditionally, systems that determine burnup are calibrated by measuring burnup indicators from a representative sample of fuel assemblies with well defined irradiation histories. This “dependent” method has the benefit that the calibration assemblies have the same geometry as the fuel to be measured. Moreover, other fuel parameters such as cooling time can be determined independently to provide validation of the operator declared parameters for the reference assemblies.

There has been interest, however, in using methods of calibration that are independent of operator declared data, “independent” calibration. One independent approach is to determine the correlation between the burnup indicators and burnup by the use of fuel inventory codes such as ORIGEN and FISPIN [10]. These codes, established for many years and validated by comparison with destructive analysis data [11], provide inventories of a wide range of fission products and actinides.

The key to the success of an independent approach is to select burnup indicators that can be calibrated by the use of the inventory codes and which can be measured reliably. There are several geometry insensitive

gamma activity ratios that are good candidates for this approach, however these ratios have limited applicability for long cooled fuels and often have dependency on initial enrichment.

The most useful burnup indicator for fuel with a broad range of enrichment, burnup and cooling times is the absolute measurement of Cs-137. The activity of this nuclide in spent fuel has been shown to be consistently predicted by the different inventory codes and validated by destructive analysis. If the measurement geometry and detection efficiency are well known and are reproducible[4], Cs-137 can be used to provide a calibration fully independent of operator irradiation history. The key is to ensure that no changes occur between the calibration conditions and the measurement conditions. A measurement procedure that uses this approach should, therefore, include suitable checks to eliminate the possibility of these systematic errors.

To provide diversity and increased confidence in a burnup instrument, a combination of the empirical/operator declared and independent computer code approaches to calibration may be used in a system. Consistency between the two calibrations would provide mutual validation.

Further discussion on the potential requirements and constraints related to dependent calibration, i.e. those that use representative spent fuel assemblies to derive a burnup/measured parameter calibration and independent calibration can be found in USDOE Topical Report on Actinide Only Burnup Credit [3].

MEASUREMENT UNCERTAINTY AND CALIBRATION PHILOSOPHY

A procedure to account for burnup verification measurement uncertainty[5] was detailed in the 1997 Topical Report [3]. This described 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. A development of this basic approach as an enhancement to that in the Topical Report was published in 2000 [12]. This approach, illustrated below, is intended to ensure account is correctly taken for both the uncertainties associated with the verification measurements and those of the reactor record data.

In general a calibration derived from the correlation between a measurable parameter, for example the activity (in arbitrary units) of the fission product Cs-137, and the declared burnup for a representative set of assemblies, will belong to the "dependent" calibration class. The use of such a calibration is viewed as appropriate for the application of burnup credit on 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. For individual assemblies the reactor record data are considered to be better than 5% uncertainty at the 2 sigma 95% confidence level [3, 4].

The developed approach calculates an acceptance criterion to qualify the dependent calibration and to determine the status of each spent nuclear fuel (SNF) assembly. This is implemented by the following procedure using as an example a linear relationship between the measured parameter x_c and burnup y_r:

1. Calibrate the measured burnup indicator against the declared burnup records to give;

$$y_r = a + bx_c$$

in which y_r is the declared burnup and x_c is the count rate of the measured burnup indicator. The parameters a and b represent the calibration constants.

[4] In practice this is a very difficult task that requires the precise linking of the fission product yield and its production as a function of irradiation as well as a detailed model of the fuel assembly and measurement configuration to determine the detection efficiency to gamma rays for each of the assembly pins in conjunction with a precise knowledge of the detector's field of view profile both axially and radially across the assembly. Such parameters would need to be determined through independent Monte Carlo and/or analytical modelling with some element of empirical proving.
[5] Applicable to any measurement system independent of measurement technique.

2. The calibration set, i.e. number of assemblies, is recommended to be consistent with a reactor core load of fuel comprising approximately 200 or more assemblies. This calibration should be carried out before commencing fuel loading.
3. 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[6]. This is to eliminate assemblies that are clearly badly measured or incorrectly declared.
4. If any assemblies are identified as outliers these should be removed from the calibration data set. The assembly reference numbers of the rejected assemblies should be recorded pending an investigation that may include further measurement and other checking procedures. Failure to evaluate and rectify the cause of their outlier positions will make those assemblies ineligible for burnup credit loading.
5. If any assemblies are rejected during step 3 then a new reduced calibration data set will be used to recalibrate the burnup indicator.
6. Steps 2, 3 and 4 are repeated until there are no rejections identified at step 2.
7. 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 other assemblies in the larger measurement campaign.
8. 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 uncertainty in *y* based on the scatter in the calibration data set [13], is:

$$y = a.x + b \pm \left\{ \sqrt{\frac{1}{n} + \frac{(x-\bar{x})^2}{S_{xx}}} \cdot \sqrt{\frac{SS_R}{(n-2)}} \cdot t_{\alpha,n-2} \right\} \qquad \text{(Eq. 1)}$$

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-α)% 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:

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

9. Calculate the minimum assured burnup for each fuel assembly by decreasing the measured burnup by its total uncertainty to a specified confidence level. The specific confidence level may

[6] 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 rejection is lower for calibration data sets in which the amount of scatter is small. In this case 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.

be defined by the regulators or be within the cask Certificate of Compliance. From equations (1) and (2) the minimum assured burnup, M_{BU} at the specified confidence level is:

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

10. 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 methodology is considered to have several beneficial features, compared to methods that use the 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. Secondly, it is capable of providing a determination of the uncertainty in burnup for each individual fuel assembly.

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, which enables an unbiased dependent calibration to be defined. An identified 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 in this proposed methodology. The improvement stems from the use of a verifiable measure of the burnup and its associated uncertainty for each individual fuel assembly.

It should also be noted that this approach takes credit for both 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 will be maximised.

In summary, it is suggested that the proposed approach offers a 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 expected that this latter value would have to be pessimistic to ensure that the worst uncertainties in the records are assumed for each assembly. This would result in lower minimum assured burnup values at the required level of confidence.

BURNUP MEASUREMENT AND CALIBRATION DISCUSSION

After debate and discussion within the USDOE [4] the NRC accepted the use of dependent calibrations [2, 3] for qualification of burnup credit measurement/verification systems. This is clearly the an appropriate conclusion because of the inherent difficulties associated with the derivation of independent burnup calibrations for SNF. These difficulties relate to the production of accurate SNF models and their applicability to particular measurement equipment and environments where conditions may differ from the theoretical/calibration model and supporting verification measurements.

If greater confidence is required to ensure that the k_{eff} of the measured assemblies is consistent with the CoC represented through the cask loading curves, a more direct measurement of fissile content or multiplication characteristics could be made. The traditional approach here is to use an external neutron source to interrogate the SNF assembly to obtain a measurement of the induced fission response that is related to the fissile content or assembly k_{eff} [7]. However, a clever technique published in the early 80s [14] suggests the use of a Cd wrapper placed around (3 sides) of an assembly. The neutron flux measured with and without the presence of the wrapper may be used to determine the magnitude of the neutron multiplication. This technique originally titled “Self Multiplication” has also been labelled as the Cd Albedo technique in a more recent technique summary publication [15].

Such a technique has a number of advantages; the most obvious of which is neutron interrogation can be performed without the hazards and management associated with the use of an external neutron interrogation

source, typically Cf-252. Also it would be relatively straightforward to engineer systems to incorporate the set of passive neutron, gamma spectrometry and self interrogation capability.

Another more recent programme of work is also looking very promising with respect to direct measurement of fissile content within fresh or spent nuclear fuel. This work, under the USDOE Next Generation Safeguards Initiative, is directed towards the development of a related technique; in so much as the inherent neutron flux is used as the interrogating source of neutrons. The technique is called Self Interrogation Neutron Resonance Densitometry (SINRD) and utilises the resonant absorption of epithermal neutrons as they make their way through the fissile materials within fuel pins of a fuel assembly [16, 17]. This phenomenon is measured by the responses of a set of fission chamber type neutron detectors placed near to the fuel. The set considered in the study references, comprises 4 detectors with 3 wrapped with different neutron shielding materials in the form of graded cadmium and B_4C, and 1 "bare" detector. In this way each detector is "tuned" to specific neutron energy bands of the fuel's epithermal flux by their particular (n,f) reaction rate. The outputs from the "shielded" detectors are compared to each other and to the "bare" detector to extract the required neutron flux information for correlation with the fissile nuclide content.

The technique is made more sensitive to the resonant neutron absorptions that take place in the fuel by matching the fission chamber neutron detection nuclide to that of the nuclide being measured in the fuel. For example the sensitivity to U235 resonant absorption in the fuel is enhanced or amplified through the (n,f) response if U-235 (normally highly enriched U-235) is used as the "active" nuclide in the fission chamber. In a similar manner the system may be tuned to different fissile nuclides, e.g. Pu239, Pu241, through the selection of these nuclides as the "active" nuclide within the fission chamber.

In general, the technique may be applied to fissile content measurements to support spent fuel management as well as for safeguards tasks, particular the detection or removal of bulk fissile material or individual fuel pins.

Other important benefits of this technique are that it may be applied to both fresh and spent nuclear fuel and for the measurement of fuel in arrays, as in fuel assemblies, or for individual fuel pins. In fresh fuel, the interrogating neutron flux is from the spontaneous fissions of the U-238 component. Clearly this is at a significantly lower magnitude than that from the Cm-244 source of spontaneous fission neutrons present in spent fuel.

CONCLUSIONS

Many burnup measurement systems have been developed and tested to provide confidence in the application of burnup credit to increase cask loading capacities and provide cost reduction benefits.

Appropriate statistical methodologies are available to maximise, with confidence, the SNF assembly compliance with approved BUC loading curves; of which the use of the "minimum assured burnup statistical" methodology seems well suited to the role.

Direct fissile or multiplication techniques, including SINRD, could be further developed to provide standard tools for fissile measurement in SNF designated for cask loading. This could give greater confidence and assurance, if required, that fuel assemblies are compliant with their cask loading curves and Certificates of Conformity by supplementing or indeed replacing burnup measurement procedures.

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