

RADIOMETRIC INSTRUMENTATION FOR BURNUP CREDIT, SAFEGUARDS AND WASTE CHARACTERIZATION OF SPENT FUEL

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ABSTRACT

The potential applications of spent fuel monitoring include activities such as; the use of burnup credit for the storage, transport and disposal of spent fuel, safeguards inventory verification and radionuclide inventory determination. Two recently published documents consider in detail the issues of burnup credit and the management of plutonium. This paper describes the relevance of spent fuel monitoring, particularly for burnup measurement, to these and to waste characterization activities. The techniques available for burnup measurement and how the data from such measurements may be used are presented. Finally recommendations are given on methods of calibration and the use of measurement systems to provide an acceptable level of confidence in the measurement result to allow these activities to take place safely and with the support of the regulators and the public.

INTRODUCTION

Spent fuel burnup monitoring can be used for burnup credit in spent fuel storage and transport, safeguards inventory verification and inventory determination to meet disposal criteria. Essential requirements for these applications are; (i) there are methodologies, approved by the relevant regulators, that detail the monitoring process, (ii) the relationships or expressions that are used to correlate the measured burnup values with parameters such as reactivity, in the case of burnup credit, fissile content in the case of safeguards, and radionuclide content for waste handling, are approved and validated, and (iii) there is a thorough understanding of the accuracy and precision associated with (i) and (ii).

As a result of the wide range of applications and requirements for spent fuel monitoring, particularly for burnup measurements, BNFL Instruments Ltd is currently developing a series of modular spent fuel monitoring systems. This series is based on instrument systems that have been used to monitor well in excess of one million fuel items in process control applications in the UK's Sellafield reprocessing facility.

BURNUP CREDIT, SAFEGUARDS & WASTE CHARACTERISATION

Burnup Credit The possibility of using the burnup of spent fuel to justify taking credit for the reduction in its reactivity, known as burnup credit, is of benefit to the nuclear industry as a means of increasing the packing density of spent nuclear fuel or for reducing the amount of neutron absorbers

required in storage racks as well as in transport and disposal casks. Based on the possible increases in cask capacities¹ significant commercial and operational advantages are anticipated by the use of burnup credit giving in the region of 25% to 40% reduction in handling costs.

Appropriate design of spent fuel racks and casks will be required to take advantage of the reduced reactivity of the spent fuel compared to the reactivity of un-irradiated fuel. The present very conservative method of using the un-irradiated or fresh fuel reactivity for spent fuel in the criticality cask design calculations, known as the "fresh fuel assumption", does lead to unnecessarily over-engineered and expensive cask designs of limited packing density. A number of vendors are therefore considering cask designs based on the reduced reactivity offered by using burnup credit in anticipation of possible licensing approval of a burnup credit methodology.

In the United States the Nuclear Regulatory Commission (NRC) control 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 (DOE). As the DOE is responsible for managing the disposal of civilian spent nuclear fuel and high level radioactive waste under the Nuclear Waste Policy Act of 1982 (NWPA) they are promoting the use of burnup credit for the benefit of the nuclear industry without compromising the safety of fuel handling procedures. To this end a topical report on actinide only* burnup credit for PWR spent nuclear fuel packages has been submitted to the NRC for their approval².

The topical report proposes a methodology for the application of burnup credit. This is encompassed in the five major steps outlined in the report :

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.

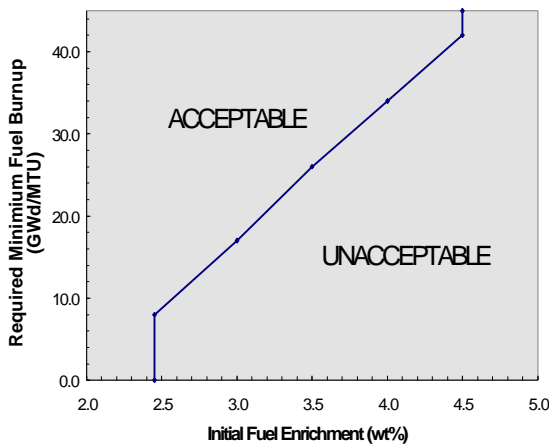
* Consideration of fission products is not included . Only the following actinides, and their effect on neutron reactivity are considered: U-234, U-235, U-236, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242 and Am-241.

- Use the validated codes and bounding conditions to generate storage, transportation, and disposal package loading criteria (burnup credit loading curves).
- Verify that spent nuclear fuel assemblies meet the package loading criteria and confirm proper fuel assembly selection prior to loading.”

The last step in the proposed methodology introduces the need to determine the reactivity of spent fuel; this will be almost certainly achieved by the measurement of the fuel burnup. The verification measurement will enhance the administrative control to ensure beyond any doubt that fuel loaded into a cask is fully compliant with the prescribed burnup credit loading curves appropriate to the chosen rack or cask design. In addition the measurement will assist in the confirmation of the identity of each assembly by verifying fuel history parameters.

The concept of “burnup credit loading curves” is described in the topical report. These provide a means of segregating fuel assemblies into “specified” assemblies, that meet the acceptance criteria for loading into a fuel storage rack or transport cask designed to take account of burnup credit. “Non-specified” assemblies are those which do not meet the criteria. The criteria are based on a combination of the fuel burnup and initial ²³⁵U enrichment. An example of a loading curve is presented in Figure 1.

Figure 1 Typical fuel loading curve²



Built into the curves are biases to account for any uncertainties in the data that relate burnup to the reactivity of the spent fuel. In addition the measurement errors associated with monitoring spent fuel need to be taken into account before the burnup can be used with the curves.

Safeguards As the global quantity of spent nuclear fuel steadily grows the need for rigorous control of the large quantities of fissile nuclides, predominantly U²³⁵, Pu²³⁹, and Pu²⁴¹, within the fuel is very important. The plutonium content of spent fuel assemblies represents about 1% by mass. In the United States alone the current stocks of 100 000 spent fuel assemblies contain about 300 tonnes of plutonium. The amount that will be accumulated through the lifetime of the currently operating US reactors is estimated to be over 1000

tonnes (unless for example reprocessing and burning in the form of MOX were to take place).

The measurement and verification of such large quantities of plutonium and fissile uranium within spent fuel assemblies beyond the level of simply item counting may be necessary. If so, the rigorous measurement methodologies required to support burnup credit, as outlined above, are likely to be equally relevant to safeguards measurements for fissile material quantification or verification.

Waste Monitoring As a significant proportion of the world’s spent fuel is not being reprocessed this material may be permanently viewed as waste and treated accordingly. In the United States there are three main options for dealing with spent fuel³, these are; (i) the once-through cycle, with permanent disposal of spent fuel, (ii) the once-through cycle, with retrievable storage of spent fuel pending a decision on how to treat spent fuel, and (iii) the recycle fuel cycle with possible burning of the plutonium in MOX fuel.

With the implementation of either of the first two options some form of monitoring may be required by the waste regulators to confirm or measure the radionuclide content of the spent fuel assemblies. In the UK information is required on a wide range of radionuclides within waste destined for a repository. These nuclides and their threshold recording levels (TRL) are listed in table 1.

Table 1 NIREX Threshold Recording Levels

Nuclide	TRL Bq/m3	Nuclide	TRL Bq/m3	Nuclide	TRL Bq/m3
Ac-227	1E+10	Eu-154	1E+8	Ra-228	1E+13
Ag-108m	1E+8	Eu-155	1E+11	Ru-103	1E+9
Ag-110m	1E+7	Fe-55	1E+14	Ru-106	1E+9
Am-241	1E+11	H-3	1E+10	S-35	1E+13
Am-243	1E+10	I-125	1E+13	Sb-125	1E+9
Am242m	1E+11	I-129	1E+8	Se-79	1E+6
Be-10	1E+7	I-131	1E+9	Sm-151	1E+13
C-14	1E+13	Mn-54	1E+8	Sn-121m	1E+12
Ca-41	1E+11	Mo-93	1E+13	Sn-126	1E+8
Ca-45	1E+13	Nb-93m	1E+13	Sr-89	1E+12
Cd-113m	1E+12	Nb-94	1E+8	Sr-90	1E+12
Ce-144	1E+9	Nb-95	1E+8	Ta-182	1E+8
Cl-36	1E+6	Ni-59	1E+9	Tc-99	1E+10
Cm-242	1E+11	Ni-63	1E+12	Th-228	1E+11
Cm-243	1E+10	Np-237	1E+7	Th-229	1E+10
Cm-244	1E+11	Pa-231	1E+9	Th-230	1E+6
Cm-245	1E+11	Pb-210	1E+13	Th-232	1E+7
Cm-246	1E+11	Pd-107	1E+14	U-233	1E+9
Co-57	1E+11	Pm-147	1E+13	U-234	1E+6
Co-58	1E+8	Po-210	1E+11	U-235	1E+5
Co-60	1E+7	Pu-238	1E+11	U-236	1E+7
Cr-51	1E+11	Pu-239	1E+9	U-238	1E+5
Cs-134	1E+8	Pu-240	1E+10	Y-91	1E+10
Cs-135	1E+7	Pu-241	1E+13	Zn-65	1E+8
Cs-137	1E+8	Pu-242	1E+8	Zr-93	1E+10
Eu-152	1E+8	Ra-226	1E+6	Zr-95	1E+8

Although the waste in the UK is generally not spent fuel assemblies but industrial radionuclides and residues from

reprocessing, similar requirements for radionuclide content seem likely for spent fuel disposal. In such a case the measurement of burnup and associated irradiation history parameters such as cooling time could be used to provide the required radionuclide inventory data.

BURNUP MEASUREMENT TECHNIQUES.

There are various techniques available for the measurement of burnup, although direct measurement of burnup can only take place when the fuel is resident in a nuclear reactor. Direct measurement is achieved by the use of in-core gamma flux probes or neutron flux wires. When used with interpolation codes to represent the power density distribution across the reactor core the burnup for individual fuel assemblies can be provided to $\pm 2\%$.

The measurement of spent fuel burnup after discharge from a reactor is generally achieved by an indirect measurement. The time since discharge, or the cooling time, at which measurements are likely to be made is typically 20 years as this is the average cooling time for current US fuels, with the majority falling into the range 5 to 40 years.

Indirect determination of burnup is by the measurement of spent fuel parameters which can be correlated with burnup. These 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; ^{134}Cs , ^{137}Cs , and ^{154}Eu . Other nuclides such as ^{144}Ce and ^{106}Ru are less useful because of their short half lives and dependence on reactor power rating but can be used in conjunction with some of the longer lived nuclides. The neutron emission from ^{244}Cm is also of use as a burnup indicator.

The primary measurement techniques for the above burnup indicators are based on low resolution gamma spectrometry (LRGS), high resolution gamma spectrometry (HRGS) and passive neutron counting.

Assuming the acceptance of a burnup credit methodology or other methodologies relevant to safeguards and waste measurements the performance achievable by the use of selected techniques must comply with or exceed the requirements set out in those methodologies. The performance may be expressed in terms of: (i) the required level of confidence in the measurement results. Greater confidence may be achieved by incorporating more than one measurement technique in a system to give diversity. (ii) the measurement accuracy. This may influence the type or number of radiation detectors in a system and the sophistication of the mechanical arrangement, (iii) the acceptable measurement time. This will depend on the planned rate of fuel throughput in a plant or process and multiple systems operating in parallel may be necessary, (iv) the acceptable "degree of blindness". The degree of blindness, described by N B McLeod⁴, indicates the extent to which utility operator declared data can be used to assist in the processing of the measurement data to determine burnup. Minimal assistance would be when only the generic

fuel type, i.e. PWR or BWR would be used as input data for a calibration and measurement procedure. The other extreme is where there is a high level of assistance or reliance placed on operator data. In the case of passive neutron measurement the measured neutron count rate is corrected for its dependence on the operator declared cooling time, initial ^{235}U enrichment and burnup before correlation of this count rate with the declared burnup. The "measurement" in this procedure acts as a confirmation of the reactor record data by checking for outliers in the fit between the count rate and the burnup.

Principal algorithms that may be used to correlate the measurable parameters with burnup make use of:

- (i) *The absolute count rate of the 662 keV gamma ray from ^{137}Cs .* This technique is attractive because of the simple linear relationship between the activity of ^{137}Cs in spent fuel and burnup. This is because ^{137}Cs is a direct fission product and has an almost equal fission yield from uranium and plutonium. Also ^{137}Cs has a half life of 30 years, which renders its production insensitive to variations in reactor power rating, dwell time and less sensitive to errors in its cooling time correction needed to account for its decay between the time of discharge from a reactor and the time of measurement. However as it is an absolute measurement technique there must be a well defined and reproducible geometry between the detectors and the fuel assembly. The linear relationship is as follows;

$$^{137}\text{Cs} = a + b \cdot \text{BU}$$

where ^{137}Cs is the count rate of the 662 keV gamma ray corrected to zero cooling time,

and a and b are constants in the linear correlation with burnup BU.

- (ii) *The gamma measurement of the activity ratio $^{134}\text{Cs} / ^{137}\text{Cs}$.* The ratio technique has an advantage over an absolute measurement because of its insensitivity to measurement geometry. Some correction for relative detection efficiency as a function of energy is required. Disadvantages are (a) the ratio has a 2.2 year half life and needs a significant cooling time correction, (b) its correlation with burnup is influenced by the initial ^{235}U enrichment and power rating and (c) its application is limited to fuel with cooling times of about 20 years or less due to the decay and disappearance of the shorter lived component, ^{134}Cs . The relationship with burnup can be approximated to a linear function;

$$^{134}\text{Cs} / ^{137}\text{Cs} = c(\text{en}, r) + d(\text{en}, r) \cdot \text{BU}$$

where $^{134}\text{Cs} / ^{137}\text{Cs}$ is the zero cooled activity ratio,

and c and d are coefficients which depend on enrichment "en" and power rating "r".

- (iii) *The gamma measurement of the activity ratio $^{106}\text{Ru} \times ^{137}\text{Cs} / (^{134}\text{Cs})^2$.* This has the advantage of being an activity ratio method, (i.e. insensitive to geometry) but unlike the

$^{134}\text{Cs}/^{137}\text{Cs}$ 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 cooling time correction errors. However due to decay of the short half life component ^{106}Ru this ratio can be used only on fuel that has a cooling time of less than about 8 or 9 years. The relationship with burnup has the following form;

$$\text{Ln}(R_0) = e + f.\text{Ln}(\text{BU})$$

where $\text{Ln}(R_0)$ is the natural log of the cooling corrected ratio $^{106}\text{Ru} \cdot ^{137}\text{Cs}/(^{134}\text{Cs})^2$, and e and f are calibration coefficients.

(iv) *The measurement of the passive neutron emission.* For fuel of greater than 15GWd/Te(U) burnup and more than 2 years cooling the primary spontaneous fission neutron emitter is ^{244}Cm . However, the measured neutron flux also depends on the fuel assembly's fission product neutron poison content and its neutron multiplication due to its residual fissile content. The advantages of this passive neutron measurement approach are; (a) it is a very sensitive indicator of burnup with the neutron emitting ^{244}Cm 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 to reach the external detectors. This is true for neutrons from central pins in the assembly as well as those from pins near the outside of the assembly. 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) ^{244}Cm has a relatively long half life of 18.1 years with its associated low sensitivity to cooling time correction errors. The disadvantages are (a) the quantity of ^{244}Cm produced during irradiation is strongly dependent on its ^{235}U 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 neutrons poisons in the pool water or within the fuel itself, (c) the measured neutron flux is influenced by neutron multiplication. The relationship between neutron count rate and burnup has the form;

$$\text{Ln}(n_0) = g(\text{en}) + h(\text{en}).\text{Ln}(\text{BU})$$

or $n_0 = i \cdot \text{BU}^j$

where n_0 is the cooling corrected passive neutron count rate,

and g and h or i and j are the coefficients which depend on initial enrichment.

CALIBRATION OF BURNUP MEASUREMENT SYSTEMS

Traditionally monitoring systems used to determine burnup have been calibrated by measuring burnup indicators from a representative sample of fuel assemblies with well defined operator declared irradiation histories. This approach has been accepted on the basis that operator data is in most cases accurate and that any isolated errors due, for instance, to errors in the paper records would be apparent as outliers and could be corrected for or eliminated from the calibration set.

This approach has the benefit that the calibration assemblies are of the same geometry as the fuel to be measured. Moreover other fuel parameters such as cooling time can be determined independently to provide partial validation of the operator declared parameters for the calibration assemblies.

There is interest, however, in using methods of calibration that are independent of operator declared data. One independent approach is to determine the correlation between the burnup indicators and burnup by the use of computer burnup inventory codes such as ORIGEN and FISPIN⁵. These codes, established for many years and validated by comparison with experimental destructive analysis data^{6,7}, provide inventories of fission products and transuranic nuclides as a function of irradiation history.

The content per unit fuel mass of the primary burnup indicator ^{137}Cs has good agreement between the two codes. There are, however, some differences in the values for ^{134}Cs , ^{106}Ru , ^{154}Eu and ^{244}Cm . For example the paper by Takeo Adachi⁶ indicates that ^{134}Cs and ^{154}Eu as predicted by ORIGEN are overestimated when compared to destructive analysis. This "high value" of ^{134}Cs from ORIGEN produces a low value, compared to FISPIN, of the complex activity ratio $^{106}\text{Ru} \times ^{137}\text{Cs}/(^{134}\text{Cs})^2$.

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 to allow the determination of burnup using a calibration independent of operator declared data. The use of an activity ratio burnup indicator, determined from gamma measurements, would appear to be a strong candidate for this approach as a ratio measurement is not sensitive to measurement geometry as long as the detection efficiency as a function of energy, can be measured. The preferred activity ratio, as mentioned above in the Burnup Measurement Techniques section is the complex activity ratio $^{106}\text{Ru} \times ^{137}\text{Cs}/(^{134}\text{Cs})^2$ and its correlation with burnup. Although, as mentioned earlier, this activity ratio can be used only on fuel cooled to a maximum of 8 or 9 years.

The most useful burnup indicator for fuel with a broad range of burnup and cooling times (to more than 50 years) is therefore the absolute measurement of ^{137}Cs . The magnitude of this burnup indicator in spent fuel has been shown to be consistently predicted by the different inventory codes and validated satisfactorily by destructive analysis. If the measurement geometry and detection efficiency are well known and are reproducible, ^{137}Cs can be used to provide a calibration fully independent of operator irradiation history. In addition the cooling time needed to correct for the decay of ^{137}Cs can also be measured independently using gamma spectrometry. The key to this approach is to ensure that no changes occur between the calibration conditions and the measurement conditions. These could include changes in; (i) detection efficiency, (ii) monitor geometry, and (iii) the position of the fuel assembly relative to the monitor. A measurement procedure that uses this approach should, therefore, include suitable checks to eliminate the possibility of these systematic errors. Initial tests of this approach at a US utility gave a good correlation between measured and

calculated ^{137}Cs count rate

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

BURNUP MEASUREMENTS IN THORP

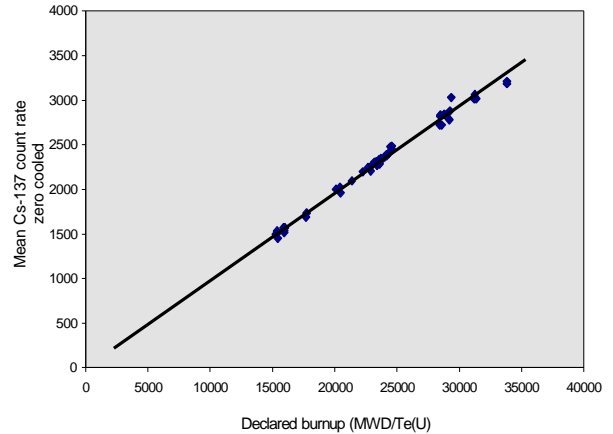
Validations of some of the above techniques are shown by the measurements on oxide fuel in the Thermal Oxide Reprocessing Plant (THORP) at Sellafield. This plant has two spent fuel instruments called the Feed Pond Fuel Monitors (FPFM). These operate in parallel in order to meet the throughput requirements and measure a number of fuel parameters to ensure that only those fuel assemblies within prescribed flowsheet limits enter the Head End of THORP for reprocessing. The flowsheet limits relate to the minimum cooling time and maximum burnup and initial ^{235}U enrichment for both light water reactor fuel (LWR) and advanced gas cooled reactor fuel (AGR). Evidence for the effectiveness of the burnup measurement techniques has been gained during monitor development by measurement of spent fuel in an oxide fuel storage pond and from the measurements in THORP since the plant was commissioned. The measurement techniques employed by the FPFM are high resolution gamma spectrometry (HRGS) and passive and active neutron counting by the use of an external ^{252}Cf interrogating source. The burnup measurements utilise the HRGS and passive neutron counting using algorithms similar to those described above. The required cooling time and initial enrichment corrections for burnup determination use measured values from the HRGS and active neutron measurements respectively. In this way the FPFM determines burnup with the minimal use of operator declared data. Only the fuel type is used to aid the measurement by the selection of an appropriate calibration.

Approximately 1000 AGR cans and 700 LWR assemblies have been measured since the start of commissioning measurements in 1993. A recent campaign of 75 PWR assembly measurements is used to illustrate the burnup measurement performance. The range and average values of irradiation, cooling time and initial enrichment of this campaign are respectively; 15 to 34 GWd/Te(U) - average 25.4 GWd/Te(U), 2600 to 7000 days - average of 5565 days (15 years), and 2.5 to 3.4% - average 2.9%. Very few measurements in this campaign made use of activity ratio techniques because the campaigning programme concentrated on the longer cooled fuel. Results from the absolute ^{137}Cs and passive neutron techniques only will therefore be presented.

Absolute ^{137}Cs Measurements - Measurement of LWR fuel are made at several positions along the assemblies as they are lifted past the monitoring position. These measurements can be correlated with local values of burnup calculated from the declared assembly average and axial profile factors. The factors are determined experimentally during system calibration.. Figure 2 shows assembly average ^{137}Cs gamma count rate against the declared assembly burnup.

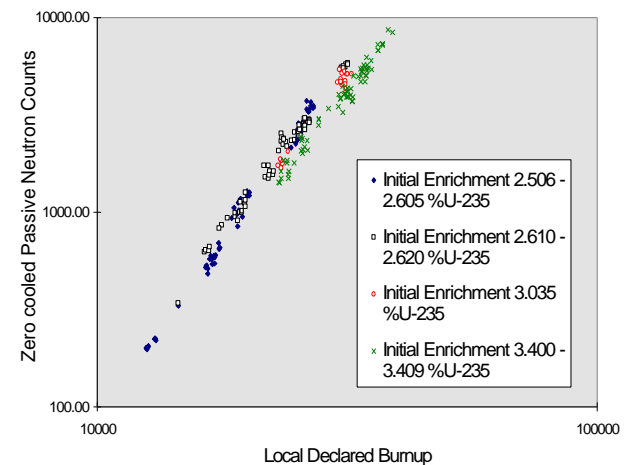
The fit gives a standard error in the burnup of less than 2% for 25 000 MWd/Te(U). This error is close to the accepted accuracy of the declared burnup which is also about 2%.

Figure 2 Zero cooled assembly average ^{137}Cs 662 keV count rate vs. declared burnup (MWd/Te(U)).



Passive Neutron Measurements - The passive neutron count rate from the fuel plotted against the local declared burnup is shown in Figure 3.

Figure 3 Zero cooled passive neutron count rate vs. local declared burnup(MWd/Te(U)).



As indicated there is some dependence on initial enrichment but this is not as strong as suggested by fuel burnup codes. This dependence on enrichment, with lower initial enrichment giving more ^{244}Cm per unit irradiation, is considered to be offset by an increase in the production of neutron poisons in the fuel per unit irradiation as well as a decrease in the neutron multiplication. The variation of neutron count rate with burnup gives an average exponent “j” of 3.5 in the expression $n_0 = i \cdot \text{BU}^j$; this is consistent with other reported values⁷.

The standard error of the fit between passive neutron count rate and burnup gives an error of about 5%, with no

enrichment correction, for a burnup of 25 000 MWd/Te(U) and less than 4% with an enrichment correction.

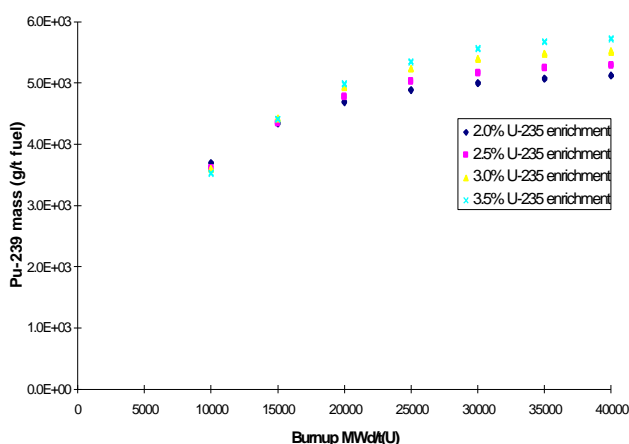
DETERMINATION OF FISSILE CONTENT, REACTIVITY, AND RADIONUCLIDE INVENTORY FROM BURNUP MEASUREMENTS.

As spent fuel assemblies are of known geometry and there are known relationships between burnup and the radionuclides produced during irradiation, it is possible to determine the quantities of a wide range of radionuclides from a measurement of burnup. The known relationships, between these fission products and actinides, are utilised in computer burnup codes such as FISPIN and ORIGEN to give the nuclides as a function of fuel type, burnup, reactor power rating, cooling time and initial ^{235}U enrichment. The power rating is, however, relatively unimportant except for the evaluation of the shorter half life nuclides, of half life ≤ 2 years.

The remaining input parameters required by the burnup codes the cooling time along with the burnup can be measured with an appropriate gamma monitoring system. For the initial enrichment either the operator declared value can be used or it can be measured using a neutron technique.

Figure 4 shows an example of the correlation, derived from FISPIN, between the measurable spent fuel parameters and one of the key fissile nuclides, ^{239}Pu . The accuracy of such correlations are better than 5%. Similar relationships exist for about 1000 fission products and actinides so that they also can be determined via this process of measurement in conjunction with FISPIN or ORIGEN burnup codes,

Figure 4 FISPIN plot of Pu-239 as a function of burnup



CONCLUSIONS

The measurement of spent fuel is likely to play a crucial role in the support of three key fuel handling activities; burnup credit, safeguards verification and waste characterization for disposal of fuel in a repository.

For each of these spent fuel measurement applications the measurement procedures would need to satisfy the regulators. Particular attention to methods of calibration and error analysis are expected with an emphasis on the use of calibrations independent of operator or utility declared fuel history data.

The experience gained by the development and industrial use of spent fuel monitors at the UK Sellafield reprocessing facility and in a recent campaign of demonstration measurements at a US utility have contributed to a development program aimed at meeting actual and anticipated spent fuel measurement requirements.

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