

SPENT FUEL CHARACTERISATION FOR BURNUP CREDIT, SAFEGUARDS AND WASTE DISPOSAL APPLICATIONS

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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, (iii) the determination of radionuclide inventories for direct disposal, and (iv) process control within a reprocessing plant to ensure adherence to flowsheet criteria. 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 radionuclide inventories to be determined.

The successful use of characterisation measurements depend 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 or enrichment. A discussion of such requirements is presented with emphasis on the licensing of burnup credit in the United States, and the regulatory controls and requirements set out in the UK with regard to waste disposal.

A range of modular spent fuel monitoring configurations for fuel characterisation are described including the advantages and disadvantages of different mechanical arrangements. In conclusion the radiometric techniques they employ are presented along with the important topics of system calibration and validation procedures necessary to ensure reliable and accurate operation.

Introduction

Nuclear power generation is operated under comprehensive regulation to ensure safety. This has led to the adoption of very conservative design criteria in, for example, power reactors and fuel handling procedures. Nevertheless, this approach has been justified by its very good safety record, as compared to other energy industries; for a period of more than 40 years. Although this has been at a higher financial cost than originally envisaged.

The continued success of nuclear power, however, depends on being economically competitive with other forms of energy as well as maintaining, and improving where possible, the safety record. As a result of the improved understanding of nuclear fuel cycle processes, coupled with advances in many areas of related technologies, the task of reducing costs is considered to be an achievable challenge and is currently being

addressed by the nuclear industry. One area resulting from advances in technology, and the subject of this paper, is in the use of advanced radiometric instruments for fuel characterisation measurements. Fuel characterisation refers to the determination of spent fuel parameters that are associated with irradiation history, as well as the fuel's residual fissile and radionuclide content. The history parameters include; burnup (or irradiation), cooling time, initial wt.% ^{235}U enrichment, and final wt.% ^{235}U equivalent enrichment (or residual fissile content).

Economic or safety benefits may be offered by fuel characterisation measurements through their application to burnup credit and safeguards, and in waste treatment, waste disposal and process control fuel handling operations. Although to satisfy rigorous regulatory controls the availability of a measurement technology is not on its own sufficient for its application to fuel handling activities. Other conditions to allow its use need

to be in place. These relate to adequate and proven experience in the technologies and techniques as well as licensing approval of appropriate methodologies which can take many years of development and validation.

As a result of work by National Laboratories, particularly in the US, and by the development of special radiometric instrumentation for the reprocessing facility at Sellafield in the UK, a number of proven methodologies and techniques have been established^{1,2,3}. In addition, through the efforts of the US Department of Energy (USDOE), licensing of actinide only burnup credit* by the Nuclear Regulatory Commission (NRC) may occur in 1997 or 1998.

Benefits Of Characterisation Measurements

Burnup Credit Measurements. Taking account of the reduction in the neutron multiplication reactivity of spent fuel which occurs during irradiation is known as burnup credit. The reduced reactivity is caused by the net loss of fissile and fissionable nuclides together with the generation of fission product poisons. The nuclides of major criticality importance are identified in an International Study on Burnup Credit⁴. These are the fissile and fissionable nuclides; uranium 235, 236, and 238, and plutonium 239, 240 and 241. The major fission products are also listed, these are; ⁹⁵Mo, ⁹⁹Tc, ¹⁰¹Ru, ¹⁰³Rh, ¹⁰⁹Ag, ¹³³Cs, ¹⁴⁷Sm, ¹⁴⁹Sm, ¹⁵⁰Sm, ¹⁵¹Sm and ¹⁵²Sm.

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 the 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 considering designs based on the reduced reactivity offered by burnup credit

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

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

change the licensing policy to one in which burnup credit can be used is being pursued by the USDOE.

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 consideration⁵. The topical report proposes a methodology for the application of burnup credit. This is encompassed in the five major steps outlined in the report :

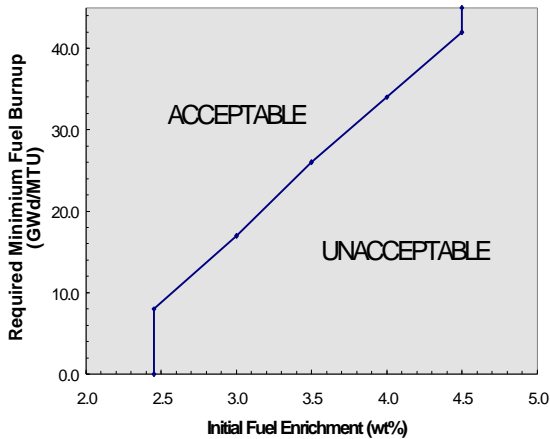
1. Validate a computer code system to calculate isotopic concentrations in the spent nuclear fuel created during burnup in the reactor core and subsequent decay.
2. Validate a computer code system to predict the subcritical multiplication factor, k_{eff} , of a spent nuclear fuel package.
3. Establish bounding conditions for the isotopic concentration from criticality calculations.
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 these steps introduces the need to determine the reactivity of spent fuel; this will be almost certainly achieved by the measurement of burnup. 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 measurement will assist in the confirmation of the identity of each assembly by verifying other fuel history parameters.

The burnup credit loading curves, described in the topical report, 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 fuel burnup and initial ²³⁵U enrichment. An example of a loading curve is presented in Figure 1.

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.

Figure 1 Typical fuel loading curve⁵



Based on the increases in cask capacities⁶ significant commercial and operational advantages are anticipated giving in the region of 25% to 40% reduction in handling costs. 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 credit⁷. 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 by the use of burnup credit, various cost factors have to be considered to determine the total net savings. These factors include; (i) the reduced storage costs associated with the use of cheaper casks designs or the use of fewer casks as alluded to above, (ii) the potential added value of radiometric measurement data acquired prior to shipment. This could eliminate the need to re-open casks at the final repository for measurements that may be required to satisfy waste acceptance criteria, (iii) implementation costs of burnup credit in terms of license approval and administration, (iv) the cost of radiometric measurements and (v) the amount of burnup credit taken, i.e. actinide only or full burnup credit with fission product poisons included. The

poisons reduce the multiplication, k_{eff} , by approximately a further 10%⁴.

Measurements in Support of Safeguards As the global quantity of spent nuclear fuel steadily grows the need for rigorous control of the large quantities of fissile nuclides, predominantly ²³⁵U, ²³⁹Pu, and ²⁴¹Pu, within the fuel becomes very important. Plutonium content represents about 1% of spent fuel assembly mass. In the United States alone, the current stocks of around 100,000 spent fuel assemblies contain about 300 tonnes of plutonium. The amount accumulated through the lifetime of the currently operating US reactors is estimated to be over 1000 tonnes, although these figures would be affected if, for example, reprocessing and recycling in the form of mixed oxide (MOX) fuels were to take place.

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 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, this may improve the prospects for conversion of DOE owned fissile material into commercial fuel for burning as MOX. Alternatively, improved faith in safeguards could encourage the earlier transfer of material from DOE sites and power utilities to a repository.

Waste Monitoring Under present policies a significant proportion of the world's commercial spent fuel is viewed as waste. In the United States there are three main options for dealing with spent fuel⁸. 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.

Even though the waste in the UK is generally not spent fuel, but industrial radionuclides and residues from reprocessing, similar requirements for radionuclide

content assessment seem likely for spent fuel disposal. The measurement of burnup and associated irradiation history parameters such as cooling time could be used to provide the required radionuclide inventory data.

Table I 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

Characterisation Measurement Techniques.

For burnup credit, safeguards or waste handling characterisation measurement methodologies, the system performance achievable using the selected techniques must comply with the appropriate performance specifications. 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 throughput in a plant or process. Multiple systems operating in parallel may be necessary. (iv) The acceptable "degree of blindness". In the context of burnup credit, the degree of blindness, described by N B McLeod⁹, indicates the extent to which utility operator declared data is used to assist in the processing of the measurement data in the determination of burnup. Minimal assistance would be when only the generic fuel

type, i.e. PWR or BWR is used as input data to a calibration or measurement procedure. The other extreme is where there is a high level of assistance or reliance placed on operator data to supplement the measurement. Measurements of this type may act only to confirm reactor data by checking for outliers in the fit between a radiation count rate and burnup. Alternatively for further reliability the primary burnup measurement, can be supplemented by the measurement of cooling time and initial wt.% ²³⁵U enrichment. These can be used with inventory code data to provide information on a wide range of radionuclides.

Burnup Measurement Techniques. Burnup is "measured" indirectly by a process of direct measurement of fuel parameters and the correlation of those parameters 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; ¹³⁴Cs, ¹³⁷Cs, and ¹⁵⁴Eu. Other nuclides such as ¹⁴⁴Ce and ¹⁰⁶Ru 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. The neutron emission from ²⁴⁴Cm is also a useful burnup indicator.

The primary techniques for the measurement of the burnup indicators are based on low resolution gamma spectrometry (LRGS), high resolution gamma spectrometry (HRGS) and passive neutron counting. Principal radiation measurements, together with their correlation relationships to give burnup, are:

(i) *The absolute count rate of the 662 keV gamma ray from ¹³⁷Cs.* This is attractive because of the simple linear relationship which exists between ¹³⁷Cs in spent fuel and burnup. The relationship is linear because ¹³⁷Cs is a direct fission product and has an almost equal fission yield from uranium and plutonium. In addition ¹³⁷Cs 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 there must be a well defined and reproducible geometry between the detectors in the monitoring system and the fuel assembly being measured.

(ii) *The activity ratio ¹³⁴Cs/¹³⁷Cs by gamma measurement.* This, a 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 needs a significant cooling time correction, (b) its correlation with burnup is influenced by the initial wt.% ^{235}U 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, ^{134}Cs .

(iii) *The activity ratio $^{106}\text{Ru} \times ^{137}\text{Cs}/(^{134}\text{Cs})^2$ by gamma measurement.* This has the advantages of being a 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 errors associated with its cooling time correction. However, due to decay of the short half life component ^{106}Ru , 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 ^{244}Cm which undergoes spontaneous fission. However, the measured neutron flux not only depends on the ^{244}Cm content but on the quantities of fission product neutron poisons and on the magnitude of 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 ^{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. 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) ^{244}Cm 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 ^{244}Cm produced during irradiation is strongly dependent on the wt.% ^{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 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 since it complicates the interpretation of the measured signal to give burnup.

Calibration of Burnup Measurement Systems

Traditionally monitoring systems that 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, particularly with regard to burnup credit, 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 fuel inventory codes such as ORIGEN and FISPIN¹⁰. These codes, established for many years and validated by comparison with experimental destructive analysis data¹¹, provide inventories of a wide range 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. Though others such as ^{134}Cs , ^{106}Ru , ^{154}Eu and ^{244}Cm do show differences. For example the paper by Takeo Adachi¹² 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, following the

resolution of the different estimates of ^{134}Cs content between FISPIN and ORIGEN, is the complex activity ratio $^{106}\text{Ru} \times ^{137}\text{Cs}/(^{134}\text{Cs})^2$ and its correlation with burnup. Although its drawback is that it 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. 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. Initial tests of this approach at a US utility gave a good correlation between measured and calculated ^{137}Cs count rate.

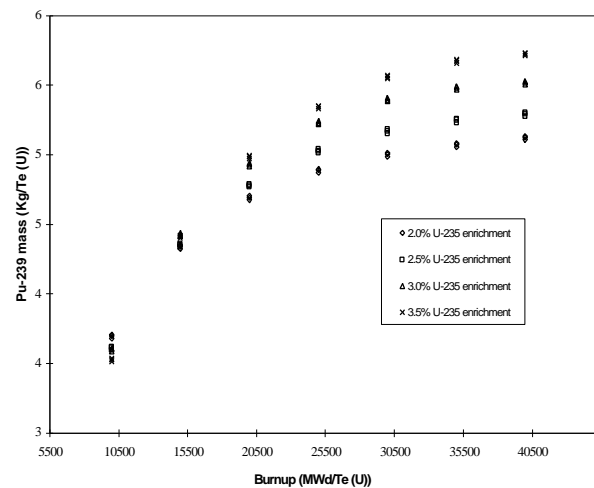
To provide diversity and increased confidence in a burnup instrument, a combination of the empirical/operator declared and independent computer code approaches to calibration could be used. Consistency between the two calibrations would provide mutual validation.

Safeguards and Waste Monitoring The needs of spent fuel safeguards and waste monitoring systems are similar to those for burnup credit measurements. Here the primary parameter to be measured is again burnup with other nuclides being inferred using the computer burnup codes, FISPIN or ORIGEN. This measurement is easier for intact fuel assemblies than for fuel residue wastes because of the known fuel pin geometry. Using the relationships between burnup and the magnitudes per unit fuel uranium mass of 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 fission products as well as actinides and burnup that are utilised in the burnup codes give the nuclide specific activities as a function of fuel type, burnup, reactor power rating, cooling time and initial wt.% ^{235}U enrichment. Of these the power rating is relatively unimportant except for the evaluation of the shorter half life nuclides, of half life ≤ 2 years. The parameters, cooling time and burnup, are both measurable with appropriate gamma monitoring systems. For the initial enrichment, either the operator declared

value can be used as this is an un-changing quality controlled parameter from fuel manufacture or it can be measured using a neutron interrogation technique.

Figure 2 shows an example of the correlation, derived from FISPIN, between the spent fuel parameters, burnup, initial wt.% ^{235}U enrichment and rating, and ^{239}Pu . The accuracy of such correlations are typically better than 5%. Similar relationships exist for about 1000 fission products and actinides so that they too can be determined via this process of measurement in conjunction with computer burnup code data.

Figure 2 FISPIN plot of ^{239}Pu as a function of burnup



Validation Of Techniques

THORP Integrated Monitoring Systems. Many of the techniques described above have been validated by use in fuel monitoring systems developed for spent fuel reprocessing plants at Sellafield in the UK.

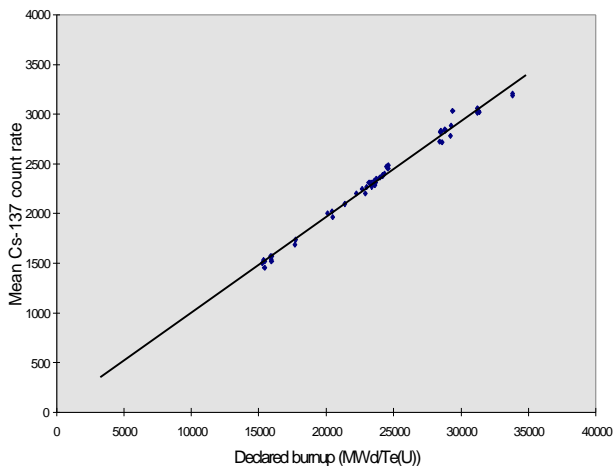
For example, the Sellafield Thermal Oxide Reprocessing Plant (THORP) has two spent fuel instruments called the Feed Pond Fuel Monitors (FPFM). These operate in parallel in order to meet the throughput requirements and measure a number of fuel parameters to ensure that only those fuel assemblies within prescribed flowsheet limits are reprocessed. The limiting parameters relate to the minimum cooling time and maximum burnup and initial ^{235}U enrichment for both light water reactor (LWR) and advanced gas cooled reactor (AGR) fuel. Confidence in the effectiveness of the measurement techniques and equipment has been gained during monitor development by measurement of spent fuel in an oxide fuel storage pond and from the subsequent operational measurements in THORP.

The measurement techniques employed are HRGS and passive and active neutron counting. The active neutron counting uses an external ^{252}Cf interrogating source to induce fissions in the residual fissile material for the determination of the initial enrichment. HRGS is used to give cooling time and burnup. Burnup is also determined from passive neutron counting. In this way the FPFM determines burnup in diverse ways with the minimal use of operator declared data. Only the fuel type is used.

Several thousand AGR and LWR assemblies have been measured since THORP commissioning in 1993, although few measurements have made use of the burnup activity ratio technique because of the relatively long cooling time of the campaigned fuel. Results from the absolute ^{137}Cs and passive neutron techniques only will be presented, therefore, to illustrate burnup measurement performance. These are from a recent campaign of 75 PWR assemblies.

Absolute ^{137}Cs Measurements. Figure 3 shows assembly average ^{137}Cs gamma count rate plotted against the declared assembly burnup. With respect to a nominal burnup of 25 000 MWd/Te(U) the correlation gives a standard error in the burnup of less than 2%. This error is consistent with the accepted accuracy of utility declared burnup which is also about 2%.

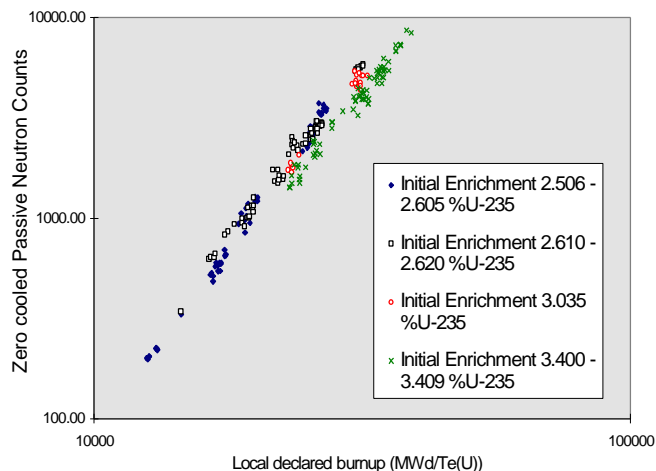
Figure 3 Zero cooled assembly average ^{137}Cs 662 keV count rate vs. declared burnup (MWd/Te(U)).



Passive Neutron Measurements - The passive neutron count rate from the fuel plotted against the local declared burnup is shown in Figure 4. As indicated there is some dependence on initial enrichment. The variation of neutron count rate with burnup gives an average exponent “j” of 3.5 in the expression $n_0 = i \cdot \text{BU}^j$ which describes the correlation between the cooling corrected neutron count rate n_0 and the burnup BU. This is consistent with other reported values¹³.

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 4% with an enrichment correction.

Figure 4 Zero cooled passive neutron count rate vs. local declared burnup(MWd/Te(U)).



Mobile Stand Alone Monitoring Systems.

In response to the wide range of applications for fuel characterisation measurements a series of modular spent fuel monitoring systems have been developed. Members of the series, based on instrument systems developed at Sellafield, are consistent with a general purpose conceptual design that can be installed in a pool as a wall fitted configuration, or as a mobile configuration to be operated over a fuel rack. These systems can accommodate equipment for any of the techniques listed above. The common system features are (i) re-entrant tubes for insertion of detectors into their respective monitoring positions at the monitoring height, (ii) horizontal gamma collimation between the fuel position and the gamma detector in a background shielded housing, (iii) a monitoring collar around which neutron detectors and if required a neutron interrogation source can be placed close to a fuel assembly.

Measurements of several hundred PWR assemblies were made in a US utility pool during 1996-97 using an HRGS based pool wall configuration system as described below.

US Utility Fuel Measurements The measured assemblies were a mixture of Babcock & Wilcox (B&W) 15 x 15 and Combustion Engineering (CE) 16 x 16 assemblies. The irradiation history ranges were; B&W - burnup 12502 to 37797 MWd/Te(U), cooling 2615 to 7139 days, and initial enrichment 1.921 to 3.902 wt.% ^{235}U . The objectives of the measurements were; (i) to demonstrate the use of HRGS to measure burnup with minimal use of

operator declared data, (ii) to produce records, based on measurement, of burnup for possible burnup credit use, (iii) to promote the feasibility of performing measurements within a fuel transfer procedure in which fuel is moved from a storage pool to dry storage casks, and (iv) to make available data to aid the current burnup credit methodology review by the NRC

Assemblies selected from those measured were to be loaded into VSC-24 type dry storage casks. This cask type gained license approval for storage on April 7, 1993 (58 FR 17948). With the adoption of a burnup credit methodology it is hoped that VSC-24 casks can gain approval for use as transport casks. Subject to NRC approval the measurement procedures as demonstrated, or procedures similar to those demonstrated, could play a part in an established burnup credit methodology.

A plan view photograph of the gamma collimator and fuel handling machine is shown in figures 5. The monitor system comprised; (i) a vertical re-entrant tube fixed to the pool wall for the insertion of a gamma detector within a carriage, (ii) a shielded enclosure to minimise the magnitude of background radiation reaching the detector, (iii) a horizontal gamma collimator with field of view defined by lead apertures and (iv) a v-shaped fuel location fixture. This permitted simultaneous views of two faces of the fuel assemblies. The daily measurement procedure included; (i) prior to fuel measurements, detector operational tests are carried out, (ii) background radiation measurements are taken, (iii) for each selected assembly both axial burnup profile and point gamma spectrometry measurements are made at several positions along the length of the assembly.

Figure 5 Plan View of gamma collimator and fuel handling machine.



Measurements from each position along the assembly were used to determine assembly average values of cooling time and burnup. The standard measurement time was approximately 30 minutes per assembly.

Results As the majority of the fuel assemblies were cooled to greater than 9 years, the principal burnup indicator used was ^{137}Cs from its 662keV gamma ray. The typical standard error (1σ) for the burnup based on the measurement of the 662 keV gamma ray count rate was 5% at a burnup of 16000 MWd/Te(U) improving to 3% at a burnup of 32 000 MWd/Te(U). These errors were derived from the correlation between measured and declared burnup with no error component attributed to the declared values. The measured cooling time obtained from isotopic ratios had a typical standard error of 100 days.

Conclusions and Recommendations

Fuel characterisation measurements offer a range of benefits. These include the support of the use of burnup credit, provision of data to aid in safeguards verification, and radionuclide quantification of fuel wastes destined for repository disposal.

In each area of application the characterisation measurement techniques and equipment must be proven in terms of performance criteria and track record to satisfy licensed methodologies. Vital qualifying criteria include calibration and error analysis with emphasis on the use of calibrations independent of operator or utility declared fuel history data.

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