Use of Burn-Up Credit in the Assessment of Criticality Risk

Issue Number: 1.0

Executive Summary

In the context of criticality safety "Burn-Up Credit" (BUC) relates to the concept of taking credit for the reduction in reactivity (usually expressed in terms of the neutron multiplication factor, k-effective) associated with the reduction in the amount of fissile material present in fuel during operation of a nuclear reactor. The burn-up of fuel changes the proportion of fissile nuclides present and the production of the actinides and fission products which absorb neutrons.

Historically in the UK, no account has been taken of fuel burn-up when assessing criticality risk for handling, storage, transport and disposal of spent fuel. This is a conservative approach, as there would be a significant reduction in the modelled reactivity of the system if burn-up were included in a criticality safety assessment. However, it is expected that the use of BUC in criticality safety cases will become necessary in order to make plant and transport safety justifications for certain types of fuel, and ultimately to support future disposal strategies for spent nuclear fuel.

It is difficult to quantify accurately the effect that BUC will have on criticality safety because of the many factors that affect the reactivity of spent fuel. Parameters that need to be considered include: reactor type, fuel composition, irradiation history, fuel positions in the reactor, cooling time and the amount of moderator intermingled with the fuel. Due to the level of complexity involved it can be difficult to undertake the precise modelling of irradiated fuel for criticality safety purposes and therefore simplifying assumptions must be utilised.

A further difficulty is the accurate measurement of burn-up. It is likely to be necessary for in-core and/or out-of-core burn-up monitoring to be used to demonstrate compliance with the safety limits and conditions that will be derived in the safety cases utilising BUC. Analyses to link the measured radiation from spent nuclear fuel with the degree of burn-up are similarly complicated, again due to the same factors identified above, as well as additional measurement-based complications.

These difficulties demonstrate that BUC is a complex technical area and indeed is one that is still undergoing significant development around the world. In order to support ONR inspectors receiving applications utilising BUC in the UK, this report has been produced to provide an overview of the methodologies and data involved in BUC currently. This report provides information covering:

     i.   regulatory practices around the world (largely concentrating on the US, as the major adopter of BUC);

     ii.   the methods which can be used to model burn-up in spent fuel for criticality safety assessment;

     iii.   the validation methods and data available for the codes and nuclear data used to calculate the models (for both depletion and criticality);

     iv.   the available technology for the calculation and confirmatory measurement of burn-up or reactivity, and the reliability of such technology; and

     v.   areas of potential difficulty, where appropriate.

In order to highlight specific pieces of key information this report includes a series of "Regulator Questions". These questions are intended as triggers for an ONR inspector when reviewing a safety case to ensure areas of known concern are addressed. No attempt has been made to provide "an appropriate answer" to each of the fifty-three questions presented. Appropriate answers may be application specific and would require suitable justification by a licensee, coupled with subsequent consideration by the regulator to confirm that they satisfy all necessary requirements. Information has been included in this report to aid in the formation of such justifications and for regulatory considerations to be made.

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2.   Historically in the UK, no account has been taken of fuel burn-up when assessing criticality risk for handling, storage, transport and disposal of spent fuel. This is a conservative approach, as there would be a significant reduction in the modelled reactivity of the system if burn-up were included in a criticality safety assessment. A number of circumstances do exist in the UK where burn-up credit has been utilised. These are:

   i.   A reprocessing plant where the safety case was able to use a simplified approach to BUC because it dealt with dissolved fuel, and not individual fuel assemblies; and

   ii.   For natural or low enrichment uranium metal fuel, where the maximum reactivity through life (MRL) does not occur under fresh fuel considerations.

3.   However, it is expected that the use of BUC in criticality safety cases will become necessary in order to make plant and transport safety justifications for certain types of fuel, and ultimately to support future disposal strategies for spent nuclear fuel.

4.   It is difficult to quantify accurately the effect that BUC will have on criticality safety because of the many factors that affect the reactivity of spent fuel. The parameters that need to be considered include: reactor type, fuel composition, irradiation history, fuel positions in the reactor, cooling time and the amount of moderator intermingled with the fuel. Due to the level of complexity involved it can be difficult to undertake the precise modelling of irradiated fuel for criticality safety purposes and therefore simplifying assumptions must be utilised.

5.   A further difficulty is the accurate measurement of burn-up. It is likely to be necessary for in-core and/or out-of-core burn-up monitoring to be used to demonstrate compliance with the safety limits and conditions that will be derived in the safety cases utilising BUC. Analyses to link the measured radiation from spent nuclear fuel with the degree of burn-up are similarly complicated, again due to the same factors identified above, as well as additional measurement-based complications.

6.   These difficulties demonstrate that BUC is a complex technical area and indeed is one that is still undergoing significant development around the world. In order to support ONR inspectors receiving applications utilising BUC in the UK, this report has been produced to provide an overview of the methodologies and data involved in BUC currently. This report provides information covering:

   i.   regulatory practices around the world (largely concentrating on the US, as the major adopter of BUC);

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   iv.   the available technology for the calculation and confirmatory measurement of burn-up or reactivity, and the reliability of such technology; and

v.   areas of potential difficulty, where appropriate.

7.   Throughout this document the aim has been to highlight specific pieces of key information by formulation of "Regulator Questions". These questions are intended as triggers for an ONR inspector when reviewing a safety case to ensure areas of known concern are addressed. The questions span the range from broad brush statements like "What specific steps have been undertaken to validate a computer code system to predict the subcritical multiplication factor of SNF in this particular application?" to the more specific "Has the licensee demonstrated that resonance self-shielding has been adequately modelled?". No attempt has been made to provide "an appropriate answer" to each of the fifty one questions derived herein. Appropriate answers may be application specific and would require suitable justification by a licensee, coupled with subsequent consideration by the regulator to confirm that they satisfy all necessary requirements. Information has been included in this report to aid in the formation of such justifications and for regulatory considerations to be made.

2 Application of Burn-up Credit

2.1 Description of the use of Burn-up Credit

2.1.1 Outline of the reasons for burn-up credit

8. Criticality safety assessments are performed incorporating a number of assumptions and approximations. The criticality safety practitioner makes these assumptions and approximations to account for uncertainties and/or unknowns in their particular application. For each assumption/approximation the assessor must ensure that a pessimistic approach is adopted such that the result of any approximation is bounding of the real situation in terms of reactivity.

9. For applications involving spent nuclear fuel (SNF) perhaps the largest of these assumptions/approximations is the so called "fresh fuel" assumption. This simplification ignores the, in most cases, reduction in reactivity associated with the irradiation of the fuel. In simple terms, the irradiation process generally results in the reduction of fissile material mass, and the addition of many neutron absorbing fission products. There would be a significant reduction in the modelled reactivity of the system if this process were included in a criticality safety assessment, using an approach known as burn-up credit (BUC). However, there are so many variables that contribute to the irradiation process that to demonstrate that the adopted irradiated fuel composition bounds the worst case, specific application composition enforces a significant calculation and verification burden on the assessment process. Similarly the fresh fuel assumption is less onerous from a code validation perspective. The vast majority of criticality benchmark experiments have been performed utilising uranium, at various enrichments, mixed oxide, or plutonium and very few incorporating more exotic actinides or fission products. Of course, an assessment utilising the fresh fuel assumption also may not have to consider the misloading scenario, in which a fuel assembly of higher reactivity is included in the system. In addition, a BUC case may impose significant operational requirements, such as verification of fuel irradiation and administrative control practices on operations, that the simplicity of the fresh fuel argument may be more appealing from an operational point of view.

10. It is worth noting that the magnitude of the reactivity reduction seen by utilising BUC would depend on the form of BUC taken. There are two types that are currently considered; these are known as "actinide-only" BUC and "limited actinide and fission product" BUC. The first approach uses a limited set of the actinides associated with the fission process, see Table 1, and the second incorporates additional minor actinides and a limited set of fission products that account for a significant proportion of the parasitic neutron absorption in the fuel, see Table 2. It is noted that the nuclides given in Table 1 and Table 2 have been chosen by the international community [Reference 1] for very specific reasons. The nuclides are chosen firstly because of their importance in criticality calculations, due to large neutron capture or fission cross-sections with large reactivity worth. Table 1 provides the actinides that contribute to a significant positive reactivity within the fuel. Table 2 provides the fission products that are either non-volatile, non-gaseous, stable or nuclides with very long half-lives and provide a significant proportion of the large negative reactivity (neutron absorption) contribution within the fuel. The second reason for choosing these nuclides is the availability of sufficient experimental data for validating the computer code systems used to:

> ► calculate isotopic concentrations in SNF created during burn-up in the reactor core and

subsequent decay; and

► predict the subcritical multiplication factor, k-effective, of SNF in the application.

11. It should be noted that there is a general lack of validation data available for fission products and the availability of such data will be discussed later in this report.

12. In addition to the two forms of BUC described above there is a third form of BUC, which is known as peak reactivity burn-up credit. This approach is used internationally for boiling water reactor (BWR) fuel elements containing integrated Gd_2O_3 within fuel elements, and in the UK for Magnox flask criticality assessments. For different reasons (namely the presence of neutron poison in the BWR fuel and the low (or natural) uranium enrichment in Magnox fuel), for these two reactor types peak fuel reactivity occurs at a burn-up some way through a reactor cycle. Therefore assessment of transport safety considers the reactivity difference, between fuel from the peak reactivity reactor situation and fresh fuel, to modify the criticality safety criterion. Reactor conditions, radial and axial burn-up effects, as will be discussed later, are taken into account to ensure that the reactivity difference included is bounding of any fuel in the core, regardless of whether there is a mechanism for including all of this worst case fuel in a single transport flask. This approach means that the criticality calculations only require validation of the fresh fuel isotopes (i.e. low enriched UO_2). A requirement to validate the depletion code still exists, of course. This peak reactivity burn-up credit is perhaps an example of inverse burn-up credit but it is noteworthy in that from the Magnox perspective it is an approach that currently has UK regulatory approval.

13. While limited use of BUC has been made in the UK (both the peak reactivity BUC approach for Magnox transport and actinide only credit approach for the THORP head-end plant), it has been adopted safely elsewhere in the world, namely in the US and Europe. The actual requirements of applying burn-up credit are fairly easily stated [from Reference 2]:

► validate a computer code system to calculate isotopic concentrations in SNF created during burn-up in the reactor core and subsequent decay;

► validate a computer code system to predict the subcritical multiplication factor, k-effective, of SNF in the particular application;

► establish bounding conditions for the isotopic concentration and criticality calculations;

► use the validated codes and bounding conditions to generate application-specific acceptance criteria (BUC loading curves); and

► verify that SNF assemblies meet the acceptable criteria and confirm proper fuel assembly selection prior to use in the particular application.

14. These outline objectives were defined by the US Department of Energy for the application of BUC to pressurised water reactor (PWR) SNF transport and storage packages, but they are an equally valid statement of requirements for general application of the technique. Therefore, these requirements present the first five questions that any regulator must ensure the licensee has answered in any criticality safety submission utilising burn-up credit:

BUC1. What specific steps have been undertaken to validate the computer code system used to calculate isotopic concentrations in SNF created during burn-up in the reactor core and subsequent decay in this application?

BUC2. What specific steps have been undertaken to validate a computer code system to predict the subcritical multiplication factor, k-effective, of SNF in this particular application?

BUC3. Have suitable measures been applied to ensure the bounding conditions for the isotopic concentration and criticality calculations have been established?

BUC4. Have validated codes and bounding conditions been used to adequately generate application specific acceptance criteria (e.g. BUC loading curves)?

BUC5. What measures have been identified that allows verification that SNF assemblies meet the acceptable criteria and confirm that proper fuel assembly selection has occurred prior to use in the particular application?

15. The above list of regulatory questions, along with others raised throughout the text of this document, are included in Table 3.

16. The specifics of how these particular questions have been addressed from the US perspective, for PWR SNF storage and transport systems, can be found in Reference 3. This document is referred to as 'ISG8 Revision 3' and presents the US Nuclear Regulatory Commission (USNRC) current guidance to their own staff on the requirements that need to be fulfilled to satisfy US regulations (10 CFR Parts 71 and 72). The issues identified above are addressed using a series of recommendations which are supported by reference to particular pieces of research work. This research is a combination of studies performed both within the United States and contributions from other organisations around the world. Although not part of a regulatory framework another document of interest is Reference 4. This is a compendium produced by the OECD/NEA Nuclear Science Committee (NSC) Expert Group on Burn-up Credit Criticality (EGBUC). This document presents similar information as will be discussed in the next section, with supporting evidence from a series of burn-up credit benchmark studies coordinated by the EGBUC. However, since Reference 3 is part of a Regulatory framework the following commentary will focus on that source information.

2.2 Recommendations from ISG8 Revision 3

17. In 2001 the USNRC released a series of guidelines, referred to as Interim Staff Guidance (ISG)-8, Revision 1, which provided recommendations to USNRC staff members for accepting, on a design-specific basis, a BUC approach in the criticality safety analysis of PWR SNF storage and transportation systems. Since that time this guidance document has been revised and, as of 2012, is at Revision 3 [Reference 3]. ISG-8 may be seen as something similar to an Office for Nuclear Regulation (ONR) Technical Assessment Guide (TAG), but with more specific prescriptive guidance included. An example of the prescriptive nature of the guidance would be that ENDF/B-VII specific bias and uncertainty data are provided as a function of burn-up, for a specific range of applicability and with caveats stating what else would be required if the application needs to extend beyond this range. Clearly the TAG documents do not go so far as to provide such nuclear data specific values that licensees are expected to use to satisfy regulatory requirements but they do provide guidance in terms of what technical issues will need to be addressed to satisfy regulatory inspection.

18. The issues identified in the US Department of Energy (USDOE) bulleted list [Reference 2] above have been addressed using a series of recommendations which are supported by reference to particular pieces of research work. The supporting work is documented in NUREG reports and these documents provide advice and supporting evidence of how to address the challenges identified.

19. The specific recommendations given are discussed herein.

2.2.1   Limits for the Licensing Basis

20.   Reference 3 indicates that the USNRC accepts that data are available to support allowance for burn-up credit based on either "actinide-only" or "limited actinide and fission product" BUC for the following specific conditions:

   ► Fuel Type: Uranium dioxide (UO_2) fuel irradiated in a PWR;

   ► Fuel Enrichment: Up to 5.0 weight percent enrichment in ^{235}U;

   ► Assembly-Average Burn-up: Up to 60 GW-days per metric ton uranium (GWd/MTU);
      and

   ► Out-of-reactor Cooling Time: 1 to 40 years.

21.   However, the USNRC staff member is instructed to "exercise care" when assessing whether the analytical methods and assumptions have been used appropriately, especially near the limits of the parameter ranges identified above. Further there is a specific stated requirement to ensure that additional data would need to be submitted and assessed to extend the basis beyond the stated range of applicability. This additional information would be measurement data and/or justified extrapolation techniques necessary to extend the isotopic validation needed to quantify or bound the bias and bias uncertainty. From a UK perspective, it would be necessary for the licensee to establish a range of applicability covering their application, and from a regulatory perspective some assessment would be required to establish whether the defined range was acceptable. This approach would also need to be adopted for other UK reactor types, where different justified ranges would need to be appropriately defined.

22.   In the US, for situations where an application has attempted to extend the range of isotopes considered in the BUC approach an additional assurance is required such that isotopes are non-volatile, and non-gaseous, and relatively stable. It is also a requirement that analyses be submitted to support any additional required bias and bias uncertainty.

23.   For some applications, Reference 3 states that a time limit of the validity of the burn-up credit analysis may be necessary. The particular example cited is that of the potential need for longer term dry storage of the SNF. Clearly this issue relates to the varying composition of the fuel due to radioactive decay. As well as being applicable to longer term dry storage, consideration of this issue would be required for the application of BUC to long term storage (permanent disposal) options. To illustrate the issue an example is cited in Reference 3 consisting of a generic flask arrangement containing thirty-two PWR fuel assemblies containing fuel with 4.0 weight percent ^{235}U initial enrichment and 40 GWd/MTU burn-up. The reactivity within the flask is shown as a function of cooling (decay) time for both "actinide-only" or "limited actinide and fission product" BUC. When considering "actinide-only" BUC the following phenomenon is observed. From an initial cooling period of less than 1 year reactivity is fairly constant but from 2 years the effect of the decay of ^{241}Pu (14.4 year half-life ($t_{1/2}$) [Reference 5]) and the in-growth of the neutron absorbing ^{241}Am means that reactivity falls. However, after approximately 100 years the reactivity begins to increase, this time due to the decay of ^{241}Am ($t_{1/2}$=433 years [Reference 5]) and ^{240}Pu ($t_{1/2}$=6,560 years [Reference 5]). The reactivity of the systems peaks at a comparable level to the five-year cooling level after approximately 20,000 years when the decay of ^{241}Am and ^{241}Pu completes and the decay of ^{239}Pu ($t_{1/2}$=24,113 years [Reference 5]) to ^{235}U begins to dominate. A similar trend is observed for "limited actinide and fission product" BUC, although clearly the overall reactivity of the system is lower, due to the presence of the neutron absorbing fission products. The majority of the fission products identified in Table 2 are stable or have substantially long half-lives.

24. From the stated limits of licensing basis the following additional regulator questions are suggested:

BUC6. Has the licensee provided sufficient measurement data and/or justified any extrapolation techniques adopted such that the established bias and bias uncertainty information can be justified as reasonable and bounding?

BUC7. Has the licensee ensured that all isotopes considered in the BUC approach are non-volatile, non-gaseous, relatively stable and that the effect of the presence of the isotopes have been considered in the estimation of bias and bias uncertainty?

BUC8. Has the licensee considered the evolution of the reactivity of the application system over a suitable time period?

25. It is noted that a further additional regulator question was considered here to capture the requirement for the licensee to establish a BUC range of applicability covering their application/reactor type but it was decided that this would largely be a repetition of ***BUC4***.

2.2.2 Licensing-Basis Model Assumptions

26. Reference 3 indicates that the actinide and fission product compositions that have been used to calculate values of k-effective supporting the application should be for fuel design and reactor operating parameter values that appropriately bound the range of design and operating conditions for the SNF in the application. Further, the calculation method should use system models and analysis assumptions that ensure an accurate representation of the physics in the system under consideration.

27. Reference 3 then goes on to identify three technical challenges to which the applicant must pay particular attention. Again, although from a US perspective these are specific to LWR type fuel, the general principles are relevant to other reactor types. The technical challenges given in Reference 3 are:

    i. to account for and effectively model the axial and horizontal variation of the burn-up within a SNF assembly (e.g. the selection of the axial burn-up profiles, number of axial material zones);

    ii. to consider the potential for increased reactivity due to the presence of burnable absorbers or control rods (fully or partially inserted) during irradiation; and

    iii. to account for the irradiation environment factors to which the proposed assembly contents were exposed, including fuel temperature, moderator temperature and density, soluble boron concentration, specific power, and operating history.

28. Appendix A of Reference 3 provides discussions on these technical issues.

2.2.2.1 Axial and Horizontal Variation of the Burn-up

29. Appendix A of Reference 3 indicates that greater consideration is required regarding the effects on reactivity of the axial variation of burn-up within a SNF assembly. It is noted that a uniform axial profile is generally bounding at low burn-up but is increasingly non-conservative (i.e. not pessimistic) at higher burn-up due to the increasing relative worth of the end regions of the fuel elements. Reference 6 describes the phenomenon in detail and has been summarised herein.

30. When fresh fuel is introduced to a reactor environment the assembly will be exposed to an axially near-cosine shaped neutron flux, such that the fuel near the centre will be depleted at

a greater rate than that at the ends. As the reactor continues to operate the cosine shape flattens out due to this fuel depletion and the build-up of fission products at the centre of the assembly. Burn-up of the ends of the fuel is supressed due to neutron leakage from the core. Therefore the majority of PWR SNF assemblies will all have similar axial-burn-up profiles, being relatively flat in the axial mid-section but with significantly less burn-up at the ends. A notional example profile is shown in Figure 1.

31. Reference 7 is a publicly-available database of axial profiles, incorporating information for fuel from US power plants for approximately 4% of all fuel discharged in the US up to 1994. Therefore, Reference 7 contains data for some 3169 PWR axial profiles from approximately 1700 different assemblies, split between three fuel vendors, twenty different reactors, and 106 cycles of operation. The data cover a range of burn-up from 3.086 to 55.289 GWd/MTU and an enrichment range of 1.24 to 4.75 wt.% ^{235}U. From the data in this document, it has been identified that the axial mid-section of the fuel assembly has a peak burn-up of approximately 1.1 times the assembly-averaged burn-up and the ends have approximately half the assembly-averaged burn-up. It is also noted that the burn-up is slightly higher at the bottom of the assembly than at the top. This difference is due to the moderator density being different at the top of the fuel than at the base. The water at the assembly inlet is at a lower temperature, and hence higher density, than higher in the core and this results in higher reactivity and subsequently higher burn-up than is seen in the warmer moderator environment at the assembly outlet.

32. A term often used when considering the axial profile is the "end effect". This term is defined as the change in the value of k-effective between a calculation containing an explicit representation of the axial burn-up distribution and one that assumes an uniform axial burn-up. The end effect has previously been used in a similar way to the peak reactivity BUC approach described previously. Namely, a bounding end effect is calculated, pessimistically set, and if a positive effect then this is added to the results of calculations performed for uniform axial burn-up. This approach is described in Reference 8, which is referred to as the "Kopp memorandum". However, it should be noted that this approach has been discouraged in the US since 2011. It is not the technique necessarily that is questionable however, but rather whether the calculation uncertainty is propagated correctly, with respect to the US regulations (10CFR 50.68 "Criticality Accident Requirements").

33. Assemblies exposed to control rods during their operating history deviate from the typical profile. However, these constitute only a small portion of discharged SNF. As the control rods are typically deployed at the early stages of an assembly's life the effect of these rods tends to be burned-out by the time the assembly is discharged. Nevertheless, this discussion shows that the axial-burn-up profile is dependent on a number of reactor operating characteristics, and it is clear that variations between individual assemblies will exist.

34. Additional axial profile databases exist around the world, with Reference 6 commenting on 120 measured profiles from two German reactors, using Siemens fuel, and a further approximately 3400 axial burn-up profiles made available as part of the Yucca Mountain disposal facility project. These latter profiles were for reactors using Framatome-COGEMA Fuels and Westinghouse PWR fuel assemblies. In addition a presentation given at Reference 9, indicated that some years ago the French were able to analyse 200 PWR-UO$_2$ fuel assembly axial profiles from different reactors covering a burn-up range of $20 < BU < 50$ GWd/t. These profiles were taken from a proprietary CEA database utilising gamma spectrometry measurement of ^{137}Cs at the La Hague fuel reprocessing plant.

35. It is assumed that UK licensees are likely to have access to their own proprietary information with regards axial burn-up profiles in the various reactor types employed in this country.

However, this information is not freely available so cannot be reviewed for consistency with other sources as part of this work.

36.   It has been shown that the burn-up profile has a strong influence on reactivity. Hence it is important to consider the effect these variations in axial burn-up have on criticality safety. Consider, for instance, a transport package flooding scenario where the initial accidental introduction of water would cause moderation in the lower burn-up, and hence a more reactive fuel, portion of the fuel assembly. Although, it should also be borne in mind that the reactivity of the system due to this effect is not restricted to accident scenarios. As will be explained below it is in fact the top end of the fuel assembly that is likely to have the lowest burn-up. Therefore even in normal condition scenarios this top area will be more reactive than the remainder of the fuel assembly.

37.   However, it must also be noted that axial variation in burn-up may not always be a significant safety issue. Detailed consideration of this effect may not be necessary, for instance, in a fuel dissolution application. For such an application, where the fuel is essentially homogenised, an average burn-up value may well be an adequate parameter. However, some argument would be required to ensure preferential accumulation of low burn-up fuel could not occur. Caution is of course noted in that it could also be considered that for advanced gas cooled reactor (AGR) fuel, where the fuel elements are of stringer type, so comprise a stack of seven or eight sub-elements, a similar argument may be appropriate. Consider that following defueling the fuel is stored and transported as packages of individual sub-elements, not an intact stack. The transport package may therefore contain a more homogenised arrangement of fuel and hence burn-up. It might be concluded that unless there is some identified mechanism where lower burn-up area sub-elements can be preferentially loaded into a storage or transport flask then an argument could potentially be made to dismiss the concern of the axial profile. However, the licensee would still have to consider the effect of the lowest burn-up fuel being positioned in the most reactive location within the storage position, to ensure the most onerous loading condition has been assessed.

38.   It will be the licensee's responsibility to demonstrate that the potential effects of axial burn-up have been adequately considered such that the real physics effects have been addressed for their particular application.

39.   The following additional regulator question is proposed:

BUC9. Has the licensee taken into account the potential effects of axial variation in burn-up and ensured that sufficient evidence has been provided to justify that a suitably representative and bounding axial burn-up profile has been used?

40.   With regards to the effect of horizontal variation of burn-up within a SNF assembly, Reference 3 indicates that this is much less of an issue. Specifically it is stated that consideration of pin-by-pin burn-up (and associated variations in SNF composition) does not appear to be necessary for analysis of the integral value of k-effective in a SNF transport package. The basis for this statement is that it is indicated that for PWR reactors the fuel is managed such that the vast majority of the fuel assemblies experience uniform burn-up horizontally across the assembly during an operational cycle. It is acknowledged that this is not necessarily the case for fuel assemblies that were situated in the core periphery. The radial shape of the neutron flux profile is approximately a zero order Bessel function, and hence fuel assemblies at the core periphery may be exposed to a significant neutron flux across their horizontal profile. However, it is judged that it is unlikely that multiple fuel assemblies that have been exposed to this type of environment would be accumulated in a storage, or transport package, arrangement. It would therefore be incumbent on the licensee to justify why this could not occur, coupled with definition of suitable safeguards, or it would be necessary for explicit

consideration of the maximum reactivity change that may occur as a result of non-uniform horizontal burn-up.

41. Power profiling is used in other types of reactor use in the UK. Therefore, consideration of this issue would clearly be appropriate irrespective of reactor design.

42. The following additional regulator question is therefore proposed:

BUC10. Has the licensee adequately considered the potential effects of horizontal variation in burn-up?

2.2.2.2 Potential for increased reactivity due to the presence of burnable absorbers or control rods during irradiation

43. Appendix A of Reference 3 indicates that consideration is required regarding the effects on reactivity of the presence of fixed and removable neutron absorbers that can affect SNF assemblies during irradiation. Fixed neutron absorbers are generally referred to as integral burnable absorbers (IBAs), whereas removable neutron absorbers are generally referred to as burnable poison rod assemblies (BPRs). These BPRs are different from control rods (CRs), in respect of their design intent. BPRs are rods containing neutron absorbing material that are inserted for the dual purpose of reactivity control and enhancing fuel utilisation, whereas control rods' sole function is as a safety feature controlling the overall reaction rate of the reactor at start-up and shutdown. Outside these periods, when the reactor is at full operating power, control rods are not inserted within the core. Consideration of the effects of CRs is also required and this will be discussed later. The effect of both IBAs and BPRs is to harden the neutron spectrum, which will result in increased fissile plutonium nuclide production and a reduction in the depletion of ^{235}U. In addition, the insertion of BPRs may also harden the spectrum further if this results in the displacement of moderator.

44. Many variants of IBA types have been used in US reactors, dependent on fuel manufacturer. The types available are described in Reference 10. That document indicates that there are two broad types, namely IBAs and integral fuel burnable absorbers (IFBAs). The former type generally consist of solid rods containing alumina pellets with uniformly-dispersed boron carbide particles (Al_2O_3-B_4C), with the rods clad in zircaloy. Variability is present in the weight percent of boron carbide and the number of rods per assembly for different operational reactors. IFBAs exist in more varied types. The major varieties are:

    i. Westinghouse ZrB_2, where rods contain enriched uranium dioxide pellets with a thin coating of ZrB_2 on the outer surface;

    ii. A number of nuclear fuel manufacturers (including Consumers Energy, Framatome ANP (formerly Babcock & Wilcox), and Siemens) have manufactured gadolinia-uranium (UO_2-Gd_2O_3) IFBA rods. These gadolinia rods, are fuel rods with gadolinia (Gd_2O_3) as an integral part of the fuel matrix and are also used extensively in BWRs; and

    iii. Consumers Energy erbia rods, similar to UO_2-Gd_2O_3 IFBA rods but using Er_2O_3 as an integral part of the fuel matrix.

45. As well as the material differences, variability exists in terms of the weight percent of absorber material present and the number of rods per assembly.

46. As stated above IBAs, IFBAs and BPRs are utilised for the dual purpose of reactivity control and enhancing fuel utilisation. Without these absorbers in a standard PWR fuel reactivity decreases nearly linearly with burn-up. With a significant number of IBAs within the reactor, fuel reactivity initially increases as fuel burn-up proceeds. A maximum reactivity is reached

only when the IBA is nearly depleted and then reactivity reduces as a near linear function. If fewer IBAs are used then the fuel reactivity reduces slowly again up to the point that the IBA is nearly depleted and then the near linear burn-up function dominates. Enhanced fuel management is therefore achieved by optimising the number of IBAs present. The reason for this IBA behaviour is that the neutron absorber hardens the neutron spectrum, as described above, resulting in increased fissile plutonium nuclide production and a reduction in the depletion of ^{235}U. The enhanced plutonium production and the concurrent reduced ^{235}U fission rate (due to increased ^{239}Pu fission) can increase the reactivity of the fuel at discharge and during the subsequent cooling period.

47. BPRs have similar effects, however they are inserted or removed from guide tubes within the fuel assemblies during reactor operation. Hence as well as hardening the neutron spectrum due to absorption, they also affect the spectrum by removing moderator by displacement. Full details of typical BPR types and reactivity effects are provided in Reference 11.

48. Either designated as a BPR or an IFBA, UO_2-Gd_2O_3 rods have been used in the UK PWR at Sizewell, and would likely be used in future reactor designs, so this is clearly an issue that would need consideration in the application of BUC for UK SNF. The following additional regulator question is therefore proposed:

BUC11. Has the licensee adequately considered the potential for increased reactivity due to the presence of burnable absorbers or control rods during irradiation?

49. With regards to control rods, either fully or partially inserted into the core, these will have effects similar to the BPRs described above. Namely, near the location of insertion the neutron spectrum will be hardened due to thermal absorption and this will lead to increased production of fissile plutonium nuclides. In addition, as the CRs are inserted at an end of the fuel assembly their insertion will affect the axial burn-up profile of any assembly exposed to such rods. However, as described above reactors only generally use CRs at start-up and shutdown, so their effects are not expected to be significant unless they were inserted for a significant fraction of the total irradiation time of an assembly. Appendix A of Reference 3 indicates that if atypical CR insertion has occurred (e.g. full insertion for one full reactor operation cycle) then specific consideration of the reactivity effects of such an insertion would need to be assessed. However, these effects may be accounted for in some cases by consideration of the available axial profile information. *BUC9* highlighted the requirement to consider axial burn-up profiles. It would be expected that some of the assemblies within the available axial profile database included those exposed to CRs. It could therefore be concluded that any assessment that had used a bounding axial profile, should have incorporated these CR insertion effects. It is prudent to include two further proposed regulator questions:

BUC12. Has the licensee adequately considered the potential effects of atypical control rod insertions during irradiation?

BUC13. Has the licensee ensured that the bounding axial burn-up profile incorporates assemblies exposed to control rod insertions?

2.2.2.3 Irradiation environment factors

50. Appendix A of Reference 3 indicates that extensive studies of the irradiation environment factors to which the proposed assembly contents were exposed have been performed. The environment factors include fuel temperature, moderator temperature and density, soluble boron concentration, specific power, and operating history. Details of the studies performed are included in Reference 1. From this document the following has been concluded.

2.2.2.3.1 Fuel temperature

51. For studies performed for BWR and PWR fuel a clear trend was observed, such that as the modelled reactor fuel temperature was increased in the depletion calculation the value of k-effective in a generic cask model also increased. This increase in reactivity was attributed to the increased temperature increasing resonance absorption in ^{238}U, due to Doppler broadening of the cross-sections. This increased resonance absorption resulted in a hardening of the neutron spectrum and increased plutonium production. Since there is an increase in the concentration of fissile plutonium nuclides there is a decrease in ^{235}U burn out. Reference 1 indicated that a temperature of 1000 K would seem to be an appropriately-conservative estimate to bound normal PWR reactor operations but recommended that a reference industry report should establish a defensible value for PWR and BWR operations. From this then a further regulator question is proposed:

BUC14. Has the licensee adequately considered the potential effects of fuel temperature during irradiation and used a suitable bounding value in the depletion calculations?

2.2.2.3.2 Moderator temperature and density

52. For moderator temperature and density, Reference 1 indicated that the most conservative nuclide compositions were produced during depletion calculations if an upper bound moderator temperature was used. This upper bound temperature was identified as that associated with the core outlet temperature. Reference 1 indicated that although the mechanisms involved in this process are different to the fuel temperature effect the resultant effect in system behaviour was the same. Namely, in a PWR environment as the moderator temperature increases, and the moderator density decreases, there is a reduction in moderation. This again causes a hardening of the neutron spectrum, with the commensurate increase in production of plutonium and reduction in ^{235}U utilisation. It was also noted that this effect is more pronounced for higher burn-up fuel, but that the variation of moderator temperature and density is relatively small. Reference 1 indicated that a value of 600 K would seem to be an appropriately conservative estimate to bound normal PWR reactor operations but again recommended that a reference industry report should establish a defensible value.

53. Clearly this would be an issue for BWR type fuel as well, but it might be anticipated that the effect would be encompassed within the consideration of a bounding axial burn-up profile.

54. From a UK perspective the majority of SNF from reactors in this country is, and will continue to be, from graphite moderated reactors. Since the effect of moderator temperature and density concerns the effect that the neutron moderator in the reactor has on the neutron spectrum, and plutonium production in uranium fuel, it will also be important to consider these parameters for UK graphite moderated reactor types. The known phenomenon of graphite weight loss, which affects the moderator density in the reactor, would similarly need consideration.

55. The effect of moderator temperature and density needs to be assessed for any reactor type and so a further regulator question is proposed:

BUC15. Has the licensee adequately considered the potential effects of moderator temperature and density during irradiation and used a suitable bounding value in the depletion calculations?

2.2.2.3.3 Soluble boron concentration

56. Within a PWR, soluble boron is used to control excess reactivity. Reference 1 indicates that typical soluble concentrations of 1000-1500 parts per million (ppm) boron is present at the start of a reactor cycle, which decreases to 0-200 ppm at the end of the cycle. It is noted that two strategies are used in depletion calculations. Either a cycle average is included, with the boron material tagged as non-burnable so a fixed amount is present through the cycle, or the code may model the depletion from initial level to discharge. Regardless of the type of approach adopted the effect of the presence of boron is the same. As with the other irradiation environment factors described above, although the mechanisms involved in this process are different, the resultant effect in system behaviour is the same. The neutron spectrum is hardened from the absorption of thermal neutrons in the moderator by the soluble boron. As with the temperature effects the effect of higher boron concentrations is more significant for higher burn-up fuel. Reference 1 indicated that a value of 750 ppm would seem to be an appropriately conservative estimate to bound normal PWR reactor operations. In this case Reference 1 stated that it would be informative to perform a study into any differences due to the modelling strategy in the depletion calculation, namely average soluble boron vs boron depletion.

57. Clearly Reference 1 is very much describing an issue associated with PWR operations. Other UK reactor types do not use soluble boron, so this issue would be easily dismissed. However, the following regulator question is proposed:

BUC16. Has the licensee adequately considered the potential effects of soluble boron concentration during irradiation, where appropriate, and used a suitable bounding value in the depletion calculations?

2.2.2.3.4 Specific power and operating history

58. Reference 3 states that the impact of specific power and operating history is much more complex but has a very small impact on the value of k-effective for subsequent transport flask systems. The effects are sufficiently complex that the effects are different dependent on the type of burn-up credit approach adopted. Namely, irradiation at higher specific power results in a slightly higher value of k-effective for actinide-only burn-up credit but the opposite is found for burn-up credit that includes actinides and fission products. Reference 3 concludes that although the specific power at the end of irradiation is most important, the assumption of constant full-power is more straightforward and acceptable while having minimal impact on the calculated value of k-effective relative to other assumptions.

59. The basis for this conclusion is discussed in Reference 1, where it is reported that the effect of various operating histories (variations in specific power with time) on the reactivity of spent fuel have been studied for a limited set of hypothetical power histograms. These histograms are not intended to examine real operating histories but aim to be bounding, incorporating key aspects of operating histories that affect the production and decay of nuclides. The key aspects considered include: extended downtime early in life; extended downtime late in cycle; high-power operation early in life; short inter-cycle downtimes; and long inter-cycle downtimes. Reference 1 reports that the results of the studies show wide variability in effects due to the significantly different decay rates and equilibrium concentrations for the nuclides considered. However it was noted that, in general, low-power operation near the end of cycle produce the highest reactivity due to decreased fission product inventory. The opposite was observed when only actinides were considered for burn-up credit, where high-power operation is more conservative at end of life. It was concluded that fission product worth is more sensitive to

specific power than that of actinides; so that when both are present the net effect is driven by fission product behaviour. Hence, Reference 1 indicates that low-power operation toward end of life yields the most conservative estimate of reactivity.

60.   Similarly the effect of specific power used during the depletion calculations has been studied independently of operating history. Although an operating history is a time-varying specific power profile, the power itself can be considered as a separable parameter. i.e. since the profile is a sequence of at power and off power (shutdown) cycles of varying frequency and duration the actual power level when at power can be independent. The study in Reference 1 considered a Westinghouse 17×17 fuel assembly for a range of burn-up and a 5-year cooling time. As with the operating history profile, varied results were obtained. The effect of the actual power used was that the calculated value of reactivity decreased with increasing specific power when fission products were present in the criticality calculation. That is to say that the reactivity worth of the actinide and fission products is highest if the fuel is burned at a lower specific power for a longer period of time. However, reactivity was seen to increase with increasing specific power when fission products were omitted from the criticality calculation, so higher power for a shorter period is worst for actinide-only BUC. It was also found that when fission products were present the reactivity change was strongly tied, with a near linear relationship, to the fuel burn-up, and to a lesser extent the initial enrichment. When fission products are not considered in the criticality calculation, both initial enrichment and burn-up are significant factors affecting the level of conservatism in the calculated spent fuel compositions.

61.   Clearly, although Reference 3 indicates that specific power and operating history will have minimal impact on the calculated value of k-effective relative to other assumptions, it is important for the UK licensee to be aware of the phenomenon. This is particularly the case since the effect seems to have some dependence on the type of BUC applied. Therefore, the following regulator question is proposed:

BUC17. Has the licensee adequately considered the potential effects of specific power and operating history during irradiation and used a suitable bounding approach in the depletion calculations?

2.2.3   Code Validation – Isotopic Depletion

62.   Reference 3 states the requirement that to ensure accurate criticality calculation results, the depletion code used to calculate the concentrations of the isotopes important to burn-up credit should be validated. This is the requirement identified as **BUC1** above. In the US, to satisfy the requirement it is necessary to establish any code bias and uncertainty on that bias for the code used. The mandatory US requirement is to determine this uncertainty on the bias using the 95% probability, 95% confidence level methodology. In the UK, this is not mandated and the licensee is free to adopt any methodology providing the arguments presented can be evidentially justified.

63.   It is noted that assay data are important to many different areas of SNF safety and management. As well as for criticality purposes, SNF compositions are utilised for evaluation of radioactivity, neutron and gamma ray source terms and decay heat. It is therefore necessary for the assay data to be sufficient to validate a variety of design and safety evaluations for the whole nuclear fuel cycle i.e. fuel handling, dry spent fuel storage installations, pool storage, fuel reprocessing facilities and waste repository studies. For this report consideration will only be given to the aspects of criticality safety, but clearly these other aspects need to be considered in a full BUC-based safety case. However, these other aspects must, logically, already have been addressed in non-BUC safety case submissions, where the fresh fuel assumption would not be the bounding situation.

64. There are a variety of potential sources of bias and uncertainty that can influence the accuracy and precision of computer codes. Some are relevant to any computer code but the following list groups potential sources of bias and uncertainty for isotopic depletion codes:

   i. Computational Methods. This is the bias and uncertainty associated with the computational algorithms, methods and numerical approximations. This relates to the computer code routines themselves rather than specific user input choices;

   ii. Nuclear cross-section and decay data. This is the bias and uncertainty associated with the nuclear data used in the depletion calculations. This should include uncertainties in the evaluated neutron cross-sections, fission product yields, branching ratios and decay constants, etc. used by the depletion code. These uncertainties will be discussed in detail in Section 3.2;

   iii. Input parameters. This is the bias and uncertainty associated with the input data used in the calculation. For example, this could include the declared burn-up of the fuel and reactor-operating parameters (such as fission power, fuel temperatures, moderator temperature and density, exposure to control rods, etc.). As discussed previously, for a BUC safety assessment it is necessary to optimise these parameters to ensure that a bounding fuel composition is produced. In terms of validation, it is necessary to have accurate information available for these key parameters to minimise calculation uncertainties when comparing with experimental values;

   iv. Modelling. This is the bias and uncertainty associated with modelling approximations. This category is distinct from computational methods in that it is dependent on modelling choices, aimed at simplifying the problem. For example, the time-dependent fission power rating may be well known, but in order to minimise the number of cycles performed in a depletion calculation these data may be approximated by smaller series of discrete steps that simulate the average power for each irradiation interval. It is important to choose an appropriate time step here; too small a time step can be inefficient in computing time while too large a step can bias the results (see 3.1.1.4). Other sources of uncertainty would be introduced by approximating the model of the fuel assembly geometry, modelling of control rods and other potentially difficult-to-simulate phenomena. An example of this latter case would be modelling the effects from adjacent reactor fuel assemblies where they may cause a significant horizontal flux gradient within an assembly.

   v. Experimental Data. This is the bias and uncertainty associated with the basic experimental data. This is potentially a large source of overall uncertainty in isotopic validation. Radiochemical analysis of irradiated fuel is a complex and difficult task, which can result in significant uncertainties on nuclide measurements. To obtain accurate measurements of isotopic compositions, irradiated fuel samples need to be destructively examined by means of radiochemical analysis methods that involve a series of complex analytical methods for sample preparation, may involve chemical separations of the various elements, and finally isotopic and elemental measurements. When comparing the differences between calculated and measured data it becomes a difficult task to differentiate between experimental uncertainty and code bias and uncertainty.

65. While these general sources of bias and uncertainty contribute to the global requirement of **BUC1** it is prudent to include the following, more specific, requirements to support regulator acceptance of **BUC1**:

BUC18. Has the licensee adequately considered the effect of computational methods on the bias and uncertainty associated with the computer code system used to calculate isotopic concentrations?

BUC19. Has the licensee adequately considered the effect of nuclear cross-section and decay data on the bias and uncertainty associated with the computer code system used to calculate isotopic concentrations?

BUC20. Has the licensee adequately considered the effect of input parameters on the bias and uncertainty associated with the computer code system used to calculate isotopic concentrations?

BUC21. Has the licensee adequately considered the effect of modelling on the bias and uncertainty associated with the computer code system used to calculate isotopic concentrations?

BUC22. Has the licensee adequately considered the effect of experimental data on the bias and uncertainty associated with the computer code system used to calculate isotopic concentrations?

66. It is stated in Reference 3 that the code and validation approach adopted for fuel depletion analysis is required to ensure that the depletion code and associated cross-section data employed should be capable of accurately modelling the fuel geometry and neutronic characteristics of the environment in which the fuel was irradiated. Clearly, this is re-stating the requirement that all code systems used in BUC assessments should use an accurate representation of the physics in the system(s) under consideration. In addition this requirement would cover the requirements to consider bias and uncertainty on the bias highlighted in bullets i, iii and iv of Paragraph 64.

67. Reference 3 goes on to indicate that in general two-dimensional depletion codes have been used in BUC applications, but one-dimensional codes have also been employed. While the former is seen as an effective means of performing BUC analyses, 1D codes have been seen as suitable for assembly average isotopic predictions for fuel burn-up. However, from a more international perspective, moving forward it is likely that 2D codes will generally be used, with three dimensional Monte Carlo simulations becoming more prevalent. Current developments in Monte Carlo codes are aimed at coupling codes to permit multi-physics approaches, e.g. depletion coupled with thermal hydraulics. However, discussion of modelling depletion will be given later in this document. The choices made in this area would impact consideration of the bias and uncertainty on the bias highlighted in bullet iv (Paragraph 64).

68. Reference 3 also describes the requirements associated with the radiochemical assay (RCA) data used to support depletion code validation. As described in bullet v (Paragraph 64) the uncertainties associated with RCA can be significant. It is therefore essential that in order for validation calculations to be effective detailed information regarding the irradiated fuel samples involved is required. This information should include the fuel rod location within the assembly, axial location of the sample in the rod, any exposure to strong absorbers (control rods, BPRs, etc.), the boron content of the moderator during irradiation periods, moderator temperature, specific power, and any other cycle-specific data for the cycles in which the sample was irradiated.

69. Inevitably some RCA data will prove unsuitable for depletion code validation due to the depletion histories or environments of these samples either being too difficult to accurately define in the code benchmark models, or sufficient detailed irradiation history is unknown. It therefore seems prudent to propose the following regulator requirement:

BUC23. Has the licensee employed appropriate radiochemical assay (RCA) or post-irradiation examination (PIE) data to support depletion code validation?

70. Potential sources of suitable RCA benchmark data will be discussed in Section 3. However, it should not be discounted that licensees will have their own, proprietary, post-irradiation examination (PIE) sample information. If this is the case then sufficient detailed information would need to be submitted with the application to allow the regulator to confidently judge its acceptability for use. The following regulator requirement is therefore proposed:

BUC24. If the licensee has utilised proprietary data to support depletion code validation has the licensee supplied sufficient information to allow regulator verification of the claimed calculation accuracy?

71. Clearly for such an important topic substantial effort has been expended in defining requirements, methodologies and accumulation of suitable information necessary to support depletion code validation. Two US NUREG references [References 12 and 13], describe various methodologies. Internationally the Working Party on Nuclear Criticality Safety (WPNCS), a working group of the OECD/NEA Nuclear Science Committee (NSC), has established an Expert Group on Assay Data of Spent Nuclear Fuel (EGADSNF). Part of the role of this group is to produce a state-of-the art report (SOAR) on assay data of spent nuclear fuel. The current SOAR [Reference 14] was published in 2011 and contains detailed descriptions of available information, measurement techniques and validation techniques. These three documents [Reference 12, 13 and 14] therefore provide an invaluable resource for regulators to assess the suitability of licensee supplied data in this area. For PWRs a recent approach to validating criticality methods for BUC, based on the analysis of EPRI depletion reactivity benchmarks, is discussed in Reference 15.

72. Discussion of the methodologies described in these documents will be given in Section 3.

2.2.4   Code Validation – Determination of k-effective

73. In principle, there is no difference in the techniques required to validate a criticality code for use in a BUC criticality safety assessment than one based on the fresh fuel assumption. In the UK, for criticality calculations, a commonly used form of the safety criterion can be written as:

$$\text{k-effective} + 3\sigma \leq 1 - X - E_{SM} - E_R - E_{PD} \qquad (1)$$

where:

      σ is the standard deviation of the k-effective value obtained by Monte Carlo modelling;

      E_{SM} is any random uncertainty in k-effective arising from uncertainty in the data available to model the system;

      E_R is a bias to allow for operational or accidental changes causing an increase in reactivity;

      E_{PD} is the systematic bias of the criticality code and data library for the type of system being analysed and is known as the error in physical data; and

      X is the subcritical margin, typically set to 5% or 2% in k-effective. More usually the 5% subcritical margin is adopted for both normal and credible accident scenarios.

74. In general, calculations performed in criticality assessments are constructed so that any modelling uncertainties are covered by setting relevant parameters to pessimistic values (i.e. to maximise the value of k-effective). Also, it is usual to optimise model parameters to ensure a calculation of the most reactive credible accident condition is performed (e.g. maximum theoretical fuel densities, maximising fissile mass, minimising mass of absorbing materials, etc.). In this way E_{SM} and E_R are usually taken to be zero and the criterion therefore reduces to:

$$k\text{-effective} + 3\sigma \leq 0.95 - E_{PD} \tag{2}$$

75. In order to calculate a suitable value of E_{PD} selection of applicable benchmark experiments is required. Modern criticality codes incorporate detailed and explicit modelling of neutron transport in continuous energy. This means that uncertainties in the calculated neutron multiplication factor result is almost entirely due to the uncertainties in the basic nuclear data, with some contribution arising from limitations of the code itself. Therefore, the most important selection criterion for validation benchmark experiments is that they should cover the main isotopes and neutron spectra found in the application. Clearly this is an area where technical issues arise with respect to calculation of E_{PD} for a case involving BUC. The vast majority of available validation benchmark experiments cover fresh fuel, be that uranium metal or oxide, mixed oxide, or plutonium metal, and are therefore not applicable to BUC. When considering actinide-only burn-up credit some coverage may be afforded by the mixed oxide lattice or solution experiments. However, the majority of the experiments performed with mixed oxide fuels have been to support applications for use of mixed oxide fuel in nuclear power reactors at the start and end-of-life, including transport. In addition, the mixed oxide would be appropriate to support reprocessing of irradiated uranium, with reasonable cooling periods of, typically, five to ten years. Issues with data applicability can therefore arise for applications such as long term storage or final disposal where the evolution of the plutonium isotopic composition could take the fuel outside the range of applicability of the validation.

76. BUC applications based upon the actinide and major fission product methodology have the additional issue that there are very few (and in some cases no) benchmark experiments that cover minor actinides and fission products.

77. Reference 3 provides a specific requirement that a particular experiment should be included in the data to support criticality code validation for both actinide-only and fission product and minor actinide BUC. The need to perform such validation is captured clearly in the requirement identified as *BUC2*. However, it is unlikely that UK regulators will state specifically which experiments should be utilised in any such validation study. In fact, the experiment stated in Reference 3 "Haut Taux de Combustion (HTC) Critical Experiment Data," was actually assessed in the supporting NUREG document [Reference 16] under a confidentiality agreement that means that a detailed and complete description of the experiments could not be presented in the analysis report. The series of experiments described were conducted by the Institut de Radioprotection et de Sûreté Nucléaire (IRSN) at the experimental criticality facility in Valduc, France during the 1980s. Since the original experimental reports are not in the public domain the Reference 3 requirement is to utilise the Reference 16 assessment of the experiment rather than use the experimental data directly. Unless the UK licensee was utilising the same code/data system as in Reference 3 the analysis would not be of direct use.

78. Clearly the scarcity of benchmark experiments involving irradiated fuel means that more sophisticated techniques, such as sensitivity and uncertainty analyses, have to be considered to justify the selection of benchmarks in support of criticality code validation. A US NUREG report [Reference 17] describes various methodologies that have been developed, supporting validation specifically relating to BUC. It is interesting to note that the techniques described in Reference 17 could also be applied to any criticality code validation to improve and perhaps further standardise the approach to the selection of appropriate benchmark experimental data for validation purposes.

79. At this stage of the document no further discussion will be included with regard to the criticality code validation. However, further commentary will be provided in Section 3 when available validation techniques and sources are discussed.

2.2.5 Loading Curves

80. Another requirement specified in Reference 3 is the need to produce "loading curves" which specify the minimum required assembly average burn-up as a function of initial enrichment for the purpose of loading SNF storage or transportation systems. In the UK, this may be referred to as an acceptable domain curve. Reference 3 further states that individual loading curves are required for each set of applicable licensing conditions. An example provided is a separate loading curve for each minimum cooling time to be utilised in the system loading. The range of applicability of the loading curve is also required to bound the various fuel types or burnable absorber loadings and this range of applicability must be justified.

81. A simple example of a loading curve is provided in Figure 2. This figure demonstrates the simplistic concept of a limiting burn-up as a function of the initial uranium enrichment, where any fuel element that does not meet the required burn-up level for its initial enrichment and hence would be below the line shown would not be meet the loading requirements.

82. A more sophisticated version is included as Figure 3. This figure has been taken from Appendix A of Reference 3. Figure 3 shows a representative loading curve, with acceptable domains for actinide only and actinide and fission product BUC superimposed over a notional population of discharged spent nuclear fuel. From the information presented it is possible to judge, for a particular application, which form of BUC would be necessary. It should also be noted that this is still rather a simplification, as a real loading curve would have to take into account the range of validity of the depletion code validation. Since this is based on measured data areas with little or no validation cover it is likely to be judged as unacceptable. Hence, whereas the domain curves cease on Figure 2 and Figure 3 they should probably be more accurately terminated with a vertical line, thus indicating that anything to the right of the line is unacceptable.

83. These curves, particularly in the form of Figure 3, are clearly useful in demonstrating the applicability of the BUC analysis adopted. If it were to be shown that significant numbers of elements would be unsuitable, then clearly a greater degree of scrutiny into the steps to be taken to ensure that unacceptable fuel elements could not be inadvertently introduced to the processing route would be appropriate. This is not to say that such scrutiny would not be appropriate if only a small number of elements were seen to be unacceptable, but rather it is a trigger for additional caution.

84. Clearly the definition of the acceptable domain loading curves should take full account of the source of bias and uncertainty on that bias.

85. From the above discussion on acceptable domain loading curves the following regulatory questions are proposed for analyses involving acceptable domain loading curves:

BUC25. Has the licensee defined suitable acceptable domain loading curves, demonstrating both the range of applicability and the anticipated population of fuel inventory?

BUC26. Has the licensee ensured that all appropriate sources of bias and uncertainty have been incorporated into the definition of the acceptable domain loading curves?

2.2.6 Misload Analyses

86. Following on from the requirement to define acceptable domain loading curves there is a companion requirement for burn-up verification. These are the steps that are to be taken to ensure fuel assemblies meet the loading criterion prior to installation in the store or transport container. The approaches to burn-up verification are detailed in Section 4 of this report, so

will not be discussed further here. However, Reference 3 indicates that an alternative strategy may be performed if a specific verification burn-up measurement is not available. This strategy involves a misload analysis. This is required to demonstrate that the system remains subcritical for such misload scenarios.

87.   It should be noted that misloading of fuel assemblies is not unique to burn-up credit. For example, misloading would also be a concern for fresh fuel scenarios if ^{235}U enrichment restrictions were imposed at different locations within a store or perhaps safety relied upon the presence of integral burnable absorber rods.

88.   Reference 3 states that the applicant is required to demonstrate that the system remains subcritical for misload conditions, including calculation biases, uncertainties and an appropriate administrative margin that is not less than 0.02 Δ(k-effective). In the past, the UK has used this smaller subcritical margin but this would only generally be acceptable for extremely unlikely postulated accident scenarios. Reference 3 does require an adequate justification if a subcritical margin below 0.05 Δ(k-effective) is to be adopted, that includes the level of conservatism in the depletion and criticality calculations, sensitivity of the system to further upset conditions, and the level of rigor in the code validation methods.

89.   Any misload analysis is required to consider not just the single, severely-underburned (or even potentially fresh) assembly, but also multiple, moderately-underburned assemblies. From the US perspective the single misload analysis is required to be selected on the basis of an assembly with reactivity that bounds 95% of the discharged PWR fuel population considered unacceptable for loading in a particular storage or transportation system with 95% confidence. For the multiple assembly misload scenario, the misloaded assemblies should occupy at least 50% of the payload and their reactivity should bound 90% of the total discharged PWR fuel population.

90.   The reason for considering both scenarios can be postulated as follows. Firstly, the single, severely-underburned assembly covers the misload scenario that, despite having rigorous procedures to prevent misload, there is still a small probability, but potentially high consequence, of a non-compliant element being introduced. Secondly, the multiple, moderately-underburned assemblies misload scenario covers a systematic failing in the definition or administration of the loading curve.

91.   Based on the above discussion the following regulator requirement is proposed:

BUC27. Has the licensee considered the implications of misloading a single, severely-underburned (or even potentially fresh) assembly as well as multiple, moderately-underburned assemblies?

92.   Reference 3 indicates that, coupled with the consideration of misloading, it is necessary to define administrative procedures to ensure that the SNF storage or transportation system will be loaded with fuel that is within the specifications of the approved contents. Clearly administrative procedures are the least preferential of safety controls, under the normal hierarchy of controls. However, in the absence of specific verification burn-up measurements they may be required. A regulator would clearly have to assess whether these procedures were sufficiently robust and that the misloading analysis covered the consequences of any likelihood of procedural failure. Therefore the following regulator requirement is proposed:

BUC28. Has the licensee defined robust administrative procedures to ensure that only acceptable SNF will be utilised in the application under consideration?

3 Review of Criticality Assessment Methods for BUC

93. This section provides an overview of the techniques, methods and data used in the application of BUC. The section also provides link back to the regulatory questions to emphasise the areas where both the licensee and regulator will need to apply judgement to ensure criticality safety can be assured when utilising BUC.

3.1 Modelling

94. The modelling requirements for BUC comprise two distinct components:

   ► estimation of the isotopic composition of the irradiated fuel by means of a suitable depletion code applied to the conditions under which the fuel was irradiated; and

   ► a criticality calculation (or sequence of criticality calculations) applied to the conditions for which the criticality safety assessment is required, using the isotopic compositions of the irradiated fuel (or a subset of nuclides).

3.1.1 Depletion

95. For BUC applications, a depletion code is required to calculate the isotopic composition of irradiated *fissile* and *burnable* materials. These include not only the fuel materials themselves, but also burnable absorbers and structural materials which contain isotopes with significant capture/decay products which will be transferred with the fuel into a system which will be the subject of a subsequent criticality calculation.

96. The depletion or burn-up of a nuclide is described by the following stiff[1] ordinary differential equation (ODE) for the rate of change of the number density for nuclide i:

$$\frac{dN_i}{dt} = -\lambda_i N_i - A_i N_i + \sum_k \alpha_{k,i} C_k N_k + \sum_k \beta_{k,i} \lambda_k N_k + \sum_k Y_{k,i} F_k N_k + \sum_k \gamma_i R_k N_k \,, \quad (3)$$

where

N_i = nuclide number density for nuclide i;

λ_i = decay constant for nuclide i;

$\alpha_{k,i}$ = probability of nuclide i resulting from capture in nuclide k;

$\beta_{k,i}$ = probability of nuclide i resulting from decay of nuclide k;

$Y_{k,i}$ = yield of nuclide i from fission in nuclide k;

$\gamma_{k,i}$ = probability of nuclide i resulting from (n,2n) reactions on nuclide k; and

A_i, C_k, F_k, R_k are the microscopic reaction rates for absorption, capture, fission and (n,2n) in the respective nuclides.

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[1] This ordinary differential equation is classed as "stiff" because some solution schemes may be numerically unstable unless a small time-step is used.

97.   A depletion calculation must solve this equation for each nuclide of interest. In order to do this two types of data are required:

►   case-independent data: decay constants; fission product yields; and branching ratios, i.e. yield probabilities after capture, decay or (n,2n) reactions; and

►   reaction rates for absorption, capture, fission and (n,2n).

98.   The first of these can be considered as nuclear data and will usually be made available to the depletion code as part of a nuclear data library. The second, the reaction rates, must be calculated by the depletion code (or a precursor calculation) taking into account:

►   the geometry;

►   the initial material compositions;

►   the conditions under which the materials were irradiated; and

►   the subsequent post-irradiation cooling time.

99.   The important aspects of the irradiation conditions are discussed in Section 2.2.2.3 and may include:

►   fuel temperature;

►   moderator temperature and density;

►   soluble boron concentration (for PWRs); and

►   specific power and operating history, noting that this may include core shuffling and repeat loading where a partially burned-up assembly is removed from the core and returned to the core in a later operating cycle.

100.   These irradiation conditions must be adequately accounted for in the depletion model. This requirement is in addition to adequately modelling the geometry of the reactor system in which the materials were irradiated, and the initial material compositions.

101.   The absorption reaction rates which are needed to calculate the depletion may be expressed in terms of the flux $\varphi(E)$ and absorption cross-sections $\sigma_{ai}(E)$, as in the following equation (similar equations can be written down for the capture, fission and (n,2n) reaction rates):

$$A_i = \int \varphi(E)\sigma_{a,i}(E)dE \qquad\qquad (4)$$

102.   As both the neutron flux and the cross-sections depend on the neutron energies it is important that the calculation of reaction rates properly accounts for the neutron spectrum. Furthermore both the flux and spectrum have a spatial dependence which must be accounted for.

3.1.1.1   Inventory model

103.   The inventory model (or burn-up chain) refers to the lists of nuclides which are produced by various reactions in a given depletion code and its nuclear data libraries. For BUC applications, this inventory model should be optimised to consider nuclides that are most significant for predicting reactivity effects. The WIMS inventory model is shown as an example in Figure 4.

3.1.1.2 Normalisation

104. The reaction rates calculated by the flux solver in a depletion code may be based on un-normalised, or arbitrarily-normalised fluxes. It is therefore necessary to specify an absolute normalisation. Various normalisation options are available, depending on the code used, and may include one or more of the following:

    ► rating (MW/t);

    ► power (MW);

    ► flux (neutrons $cm^{-2} s^{-1}$); and

    ► fission rate (fissions s^{-1}).

105. To properly account for the specific power and operating history (see Section 2.2.2.3.4) it will generally be necessary to use different normalisations in different time-steps of the depletion calculation.

106. The following regulatory question is proposed:

BUC29. Has the licensee demonstrated the use of appropriate and correct flux normalisations in each time-step of the depletion calculation?

3.1.1.3 Spatial dependence

107. Section 2.2.2.1 discussed the axial and horizontal variation of the burn-up, and this must be accounted for in a depletion calculation. This introduces a requirement to discretise the materials in the model, since materials which have the same composition at the start of the depletion calculation but whose compositions change as a result of burn-up must be represented by unique model materials. In some 3D codes (e.g. recent versions of MONK) this may be automated to some extent by the ability to overlay a spatial mesh to specify 3D spatial material discretisation.

108. Axial variation of burn-up is generally considered to have a greater effect on reactivity because of the lower burn-up at the ends of the rods (see Section 2.2.2.1). In a 3D depletion code this can generally be accounted for by defining unique model materials in each axial zone of the fuel rods. In a 2D lattice code there is no axial dependence so it will generally be necessary to run a separate calculation for each axial slice, using different normalisations to represent the different specific power in each slice.

109. The horizontal dependence is generally considered less important, but where it does need to be accounted for this can be done by: defining different fuel rods in different horizontal regions; identifying rods which should be depleted differentially (e.g. WIMS) or by an overlaid mesh (e.g. MONK). These methods are code-specific.

110. Burnable absorber rods and fuel rods containing burnable absorbers (e.g. gadolinia-bearing fuel rods) should also be discretised radially in order to account for the strong resonance self-shielding effect of the absorbers which causes the outer layers of the rod to burn up more strongly than the centre.

111. In addition to the regulatory questions posed in **BUC9** and **BUC10**, the following additional question relating to the spatial discretisation of materials in the depletion model is proposed:

BUC30. Has the licensee used sufficient spatial discretisation of materials in the horizontal and axial directions, and radially within rods containing burnable absorbers, to adequately represent the spatial variation in the burn-up in the depletion model?

3.1.1.4 Time dependence

112. The solution of Equation (3) in a depletion model can be achieved using standard solvers for stiff ODEs. However, it is necessary to assume some time-step over which the fluxes and reaction rates can be assumed constant, and to adopt some coupling scheme between the flux (and reaction rate) solver and the depletion solver.

113. Typically depletion codes use an explicit Euler scheme in which the fluxes and reaction rates are calculated for the current time-step, and the number densities at a future time-step are calculated from the number densities and reaction rates at the current time-step. Some codes alternatively use a predictor-corrector method which can produce more accurate estimates of isotopic composition at the expense of requiring two flux solutions per burn-up time-step.

114. Both the Euler method and predictor-corrector method are conditionally stable for stiff equations so it is important to ensure that the time-steps used are small enough that numerical instabilities are not seen. This may mean examining the behaviour of the flux solution over the time-steps and checking for spatial oscillations in the flux.

115. The length of the time-steps may also have an effect on the accumulation of certain isotopes and the depletion of burnable absorbers, so care is required to ensure that the time-steps are small enough that this effect is not significant.

BUC31. Has the licensee used sufficiently-small burn-up time-steps to ensure a numerically-stable and converged estimate of the isotopic composition of the irradiated materials?

3.1.1.5 3D Monte Carlo modelling

116. Monte Carlo modelling provides the most accurate method of calculating fluxes and reaction rates for solving the depletion equations, with the fewest approximations. The ability to model full 3D geometry without approximation, use continuous energy cross-section data, and calculate continuous scattering angles results in a very high fidelity calculation the accuracy of which is ultimately limited only by the accuracy of the underlying nuclear data.

117. This accuracy however comes at a high computational cost, particularly for whole-core depletion calculations since the transport calculation must be carried out at least once per time-step. For this reason 2D deterministic lattice physics codes are often employed.

BUC32. Has the licensee demonstrated that the reaction rates in any 3D Monte Carlo calculations have been adequately sampled to give sufficiently-accurate isotopic compositions?

3.1.1.6 2D Deterministic lattice physics codes

118. The lower computational cost of 2D deterministic modelling makes it attractive for depletion modelling. It is, however, important to understand the approximations and limitations associated with this approach. These include:

   ► discretisation in space, angle and energy; and

   ► approximate methods for resonance self-shielding, such as the subgroup method or equivalence theory.

119. Note that the spatial discretisation discussed here is that employed by the 2D deterministic flux solver, as distinct from the discretisation of materials to account for the spatial dependence of the burn-up discussed in Section 3.1.1.3. The following additional regulatory questions are therefore proposed if 2D deterministic methods have been used in the depletion calculation:

BUC33. Has the licensee demonstrated that the spatial, angular and energy discretisation employed by a 2D flux solution give sufficiently accurate isotopic compositions?

BUC34. Has the licensee demonstrated that resonance self-shielding has been adequately modelled?

3.1.1.7 Depletion code packages

120. As noted in the preceding sections a depletion model requires both a neutron transport calculation to determine reaction rates, and a solver for the depletion equations. Some codes (e.g. MONK, WIMS, MCNP) include both components; in other cases the transport code must be coupled to a separate depletion code. Table 4 lists some example code packages (this is not intended to be an exhaustive list).

3.1.2 Criticality

121. Criticality calculations in BUC applications are essentially the same as those used when the fresh fuel assumption is adopted; the only differences being in the isotopic compositions of the fuel materials (and other burnable materials transferred along with the fuel). In most cases 3D Monte Carlo models will be used, since these offer the greatest accuracy with the fewest approximations.

122. The spatial discretisation of the fissile and burnable materials in the depletion code may generate a much larger number of model materials than is typically encountered in criticality calculations based on the fresh fuel assumption. The licensee will therefore have to demonstrate a rigorous approach to ensuring that the discretised material compositions are transferred to the correct geometric locations in the criticality calculations if intact assemblies are to be modelled. Some codes (e.g. MONK) may include user options to facilitate this process.

BUC35. Has the licensee demonstrated a rigorous approach to ensuring that the discretised material compositions are transferred to the correct geometric locations in the criticality calculations?

123. If actinide-only BUC, or limited actinides and fission products BUC, are to be applied then only the nuclides of interest for the chosen approach are transferred from the depletion model to the criticality model. Again the licensee will need to demonstrate a rigorous and consistent approach.

124. The presence of fission products and actinides should be considered when selecting suitable validation benchmarks and estimating the bias in the criticality calculations. Here the use of sensitivity calculations is recommended, and validation benchmarks should be selected on the basis of having similar sensitivities to the application case.

3.2 Benchmarks

125. One of the activities of the OECD/NEA Nuclear Science Committee (NSC) Expert Group on Burn-up Credit Criticality (EGBUC) has been to coordinate international benchmark activities relating to Burn-up Credit. Reference 4 provides an overview of the benchmarks performed and gives conclusions of what the practical outcomes of these studies are. While not providing specific validation of particular codes, since a number of the benchmarks are for notional cases, these studies do provide inter-code comparisons, thus allowing trending analyses to be performed.

126. The benchmark studies have been performed in a number of phases, which are summarised in Table 5. As can be seen the date range of these studies beings in the early 1990s and continues to present day. This is another reason why these benchmarks do not directly provide validation support, since codes and nuclear data have changed much over this time period.

127. From Table 5 it can be seen the various phases were designed to consider specific issues, already described in Section 2. It should also be noted that Reference 4 summarises and discusses conclusions derived from PWR (UOX and MOX) studies. As yet there is no equivalent document covering the three BWR based benchmarks.

128. The various phases are discussed in the following sections.

3.2.1   Phase I: PWR UOX

3.2.1.1   I-A

129. This benchmark considered an infinite array of simple PWR spent fuel rods. The analysis used PWR spent fuels of 30 and 40 GWd/t after one and five years of cooling time. For the nuclides in spent fuel, seven major actinides and fifteen major fission products were considered. These are consistent with those given in Table 1 and Table 2, with the exception of ^{151}Eu which was not used in the benchmark. In the case of 30 GWd/t burn-up, it was found that the major actinides and the major fission products contributed more than 50% and 30% of the total reactivity loss due to burn-up, respectively. Therefore it was shown that more than 80% of the reactivity loss is covered by these twenty two nuclides. For this study the isotopic composition was supplied as part of the benchmark specification, having been calculated by ORNL using the ORIGEN-S/SAS2H sequence of SCALE (although not stated, the age of the report suggests that this would be SCALE-4).

130. The main conclusion, apart from the reactivity loss percentages given above, related to inter-code discrepancies, which were attributed to cross-section data differences. It has to be said that this was the earliest benchmark performed and so from a UK perspective, MONK6B was utilised with UKNDL-based nuclear data. This nuclear data library contained adjusted data to ensure uranium systems over-predicted by ~1%. From a modern re-evaluation of this benchmark closer agreement would be expected, with unadjusted cross-section data and a greater degree of commonality between nuclear data evaluations used in international nuclear data libraries.

3.2.1.2   I-B

131. This benchmark was intended to compare the ability of different code systems and data libraries to perform depletion analysis for the prediction of spent fuel isotopic concentration. While the expected results were based on chemical assay measurements of isotopic concentrations the benchmark specification was simplified to a single infinite pincell, as in Phase I-A, since the purpose of the benchmark was to compare basic methods and data.

132. The main conclusion of this analysis was that most methods were in agreement to within 10% in the ability to estimate the spent fuel concentrations of most actinides, all methods were within 11% agreement about the average for all fission products studied, and most deviations were less than 10%, and many are less than 5%. However, larger discrepancies were noted for the calculated fission product inventories of ^{149}Sm, ^{151}Sm, and ^{155}Gd.

3.2.2 Phase II: PWR UOX with increasingly more realistic geometries

3.2.2.1 II-A

133.   This benchmark retained an infinite array of a simple PWR spent fuel rod geometry from the earlier phase, but in this case an axial variation of fuel composition was employed. The fuel rod was split into nine separate axial fuel locations, end plugs and top and bottom water reflectors. Again, for this study the isotopic composition was supplied as part of the benchmark specification, having been calculated by ORNL using the ORIGEN-S/SAS2H sequence of SCALE. As well as incorporating case using an uniform axial burn-up profile, this study also considered compositions produced using representative axial burn-up profiles provided by CEA Cadarache, France. As well as multiplication factors, participants were requested to provide fission density profiles in the five upper fuel regions, for a selection of the cases.

134.   In terms of system multiplication, again the spread of results obtained between codes was of the order of ~1%, but was again probably somewhat compromised by the inclusion of MONK6B-UKNDL results. In terms of end effect for irradiation up to 30GWd/t, the end effect was found to be less than 1.0% Δk. However, for the 50 GWd/t case, the end effect increases to more than 4.0% Δk when both actinides and fission products were taken into account. For the higher burn-up cases the effect remained less than 1.0% Δk when only actinides were considered. It was recognised that these effects were probably representative but sensitive to the effects of both neutron leakage and axial asymmetry of material composition. Consideration of this issue was made in the later models of this phase.

3.2.2.2 II-B

135.   This benchmark repeated the II-A case but this type utilising a 3D geometry modelling a conceptual PWR spent fuel transport container. Nine basic cases and two additional accident configurations were considered with the following varying parameters: burn-up (0 GWd/t for fresh fuel, 30 and 50 GWd/t), fuel composition (actinides only and actinides with fifteen fission products), axial burn-up discretisation (one or nine zones). Good agreement was found between participants for calculated k-effective, with the dispersion of results, ranging between 0.5% and 1.1% for irradiated fuels and 1.3% for fresh fuel. The reactivity effect of axial burn-up profile for basic cases was consistent with Phase II-A. However for two accident cases the reactivity effect of axial burn-up was dependent on the configuration studied. In these cases the bottom 20 cm of neutron absorbing panels, between fuel assemblies, was replaced with water. For these accident conditions the axially averaged flat distribution was found to be a non-conservative approximation even for low burn-ups (10 GWd/t) and without fission products; the reactivity effect of burn-up profile reached -14000 pcm for 50 GWd/t burn-up and composition including fission products. As a result of this study it was also shown that source convergence can be an issue when considering axial burn-up.

3.2.2.3 II-C

136.   This benchmark was similar to II-B, incorporating the same conceptual transport container geometry but this time the fuel was split into 21 axial regions. This time 34 axial burn-up profiles were studied, incorporating asymmetries derived from real profiles from a German PWR "German Convoy Series Nuclear Power Plant Neckarwestheim II". The aim of this study was to consider the impact of these asymmetries on end effect and fission density. Strong correlations were found between the end effects and the "asymmetry parameters".

3.2.2.4   **II-D**

137. This benchmark was performed in three parts. The first covered depletion calculations, the second k-infinity calculations and the third a series of sensitivity calculations. The aim of the study was to examine the effects of control rod insertion on spent fuel compositions and on reactivity for a PWR-UO$_2$ fuel assembly. The results from the depletion calculations showed good agreement between participants' results with the exception of ^{155}Gd. The absolute difference from the mean value was seen to be less than 10%, but ~50% for the ^{155}Gd. Analysis of effect on k-infinity of CR insertion also showed good agreement between participants. The relative standard deviation was seen to be ~6%, which corresponded to 300 pcm. Analysis of the sensitivity exercise showed, as would be expected, that the most important effect on k-infinity or k-effective are the ^{235}U (+140 pcm/%), ^{238}U (-120 pcm/%), ^{239}Pu (+140 pcm/%), ^{240}Pu (-40 pcm/%) and ^{241}Pu (+50 pcm/%). The effect of other nuclides was seen to be less than 20 pcm/%, with most contributing less than 5 pcm/%.

3.2.2.5   **II-E**

138. This benchmark complemented the earlier phases, in that the purpose of this phase was to study the impact of changes on the spent nuclear fuel isotopic composition, due to control rod insertion during depletion, on the reactivity and the end effect of spent fuel assemblies. Again, realistic axial burn-up profiles were used for different control rod insertion depths, ranging from no insertion to full insertion (i.e. over entire active fuel length). The axial burn-up profiles were extracted from an AREVA-NP-GmbH-owned 17×17-(24+1) PWR UO$_2$ spent fuel assembly burn-up profile database.

3.2.3   **Phase III: BWR UOX geometries**

3.2.3.1   **III-A**

139. This benchmark was intended to confirm the predictive capability of computer code and data library combinations for the neutron multiplication factor of an irradiated BWR fuel assembly within an array. Twenty two benchmark problems were considered, studying the effect on k-effective of cooling time, inclusion/exclusion of fission product nuclides and axial burn-up profile, and inclusion of axial profile of void fraction or constant void fractions during burn-up. Axial profiles of fractional fission rates were also considered for five cases out of the twenty two cases. The relative dispersion of k-effective from the mean value is almost within the band of ±1 %Δk/k. Again, due to the age of this benchmark exercise UKNDL data was employed by some UK contributors. The deviations from the averaged calculated fission rate profiles were found to be within ±5 % for most cases.

3.2.3.2   **III-B**

140. This benchmark was intended to compare the predictability of computer code and data library combinations for the atomic number densities of an irradiated BWR fuel assembly model. The fuel assembly was irradiated under specific power of 25.6 MW/tHM up to 40 GWd/tHM and cooled for five years. The void fraction was assumed to be uniform throughout the channel box and constant, at 0%, 40% and 70%, during burn-up. The calculated atomic number densities of twelve actinides and twenty fission product nuclides were found to be for the most part within a range of ±10% relative to the average, although some results, especially Eu and Gd isotopes, exceeded this level. Pin-wise burn-up results were considered to agree well among the

participants. The results in the infinite neutron multiplication factor k-infinity was also considered to be calculated consistently amongst participants, for void fractions of 0% and 40%; however some results deviated from the averaged value noticeably for the void fraction of 70%. The UK contribution used the WIMS code with JEF2.2 nuclear data.

3.2.3.3 III-C

141. The Phase II-B benchmark was considered to be a landmark that provided general information on burn-up properties of BWR spent fuel. It was based on an 8×8 type fuel assembly, which was prevalent at the time of the benchmark specification, circa 2002. Currently the 9×9 type fuel assembly is more widely used. In the light of the accident at the Fukushima Daiichi Nuclear Power Station it was decided to define a new BWR benchmark along the lines of Phase III-B, but using the 9×9 "STEP-3 BWR fuel" assemblies employed in Japan. This is clearly a very recent exercise (performed in 2015) and the UK did not provide a contribution. The results are therefore focussed largely on SCALE6, CASMO5, SERPENT and a number of MCNP variations. Also ~46% of the contributions used ENDF/B-VII.0.

3.2.4 Phase IV: PWR-MOX fuel

3.2.4.1 IV-A

142. This benchmark considered an infinite array of simple PWR - MOX fuel rods, and consisted of sixty three separate calculations with three different initial MOX compositions (MOX plutonium content and initial plutonium vector), and a range of burn-up periods and cooling times. The reactivity worth of the major actinides for the later generation MOX fresh fuel composition was chosen as it contained the largest plutonium content. It was shown that ^{239}Pu gave the largest positive worth followed by ^{241}Pu. The largest negative worth was due to ^{240}Pu followed by ^{238}U. It was therefore concluded that the extent of burn-up credit gain would be dominated by these four isotopes. However, an important finding of the study was the reactivity worth of the curium isotopes. It was found that if they were included in the spent fuel composition of MOX fuel a positive reactivity worth addition, due to ^{245}Cm isotope was seen. Inclusion of curium isotopes in MOX fuel was shown to increase the reactivity of the system by up to 1000 pcm. It was therefore recommended that curium isotopes should be included in any MOX BUC cases. Clearly this would be a factor for a regulator to consider, possibly as part of ***BUC7***.

3.2.4.2 IV-B

143. This benchmark considered fuel assembly and super-cell (multiple assemblies) geometries. The benchmark considered three different plutonium isotopic vectors related to high, medium and low plutonium content fuel pins so that the plutonium distribution within the MOX fuel assemblies was appropriately represented. As, in reality, the MOX fuel would be irradiated in a mixed UO_2-MOX PWR core, i.e. alongside UO_2 fuel assemblies, the super-cell model included a fuel assembly with a typical UO_2 fuel composition (initial enrichment of 4.3 w/o ^{235}U/U). In all cases, the uranium oxide component of the MOX was assumed to be depleted, as is typical of current MOX fuel fabrication.

3.2.5 Phase VI: VVER-440

144. This phase considers the effects of axial profile on a VVER-440 fuel assembly. As this work is on-going it will not be discussed further here.

3.2.6 Phase VII: Study of spent fuel compositions for long-term disposal

145. The purpose of this study was to consider the predictive capabilities of computer decay code and data library combinations with respect to concentrations of nuclides important to radiological dose assessments and burn-up credit criticality analyses. The study also considered the predictive capabilities of criticality codes to predict k-effective in a generic transport container. Since the vast majority of BUC benchmarks consider the reactivity of systems for a relatively short period (less than 100 years) this study was designed to consider systems of geological periods (10^6 years from discharge). The SNF considered was for a "standard" PWR UO_2 fuel of 4.5 wt.% initial enrichment irradiated to 50 GWd/MTU. Conclusions from this study were that the differences observed could be attributed to a) different decay data, b) the ability of the decay codes to consider relevant precursor decay chains contributing to the formation of a nuclide of interest, and c) approximations related to the number of time steps allowed by the code.

3.2.7 Phase VIII: Reactivity worth benchmark

146. Study of reactivity worth of small isotopic samples (actinides and fission products) in pile-oscillation experiments. As this work is on-going it will not be discussed further here, other than to say that it is interesting to note that the benchmark specification indicates a strong recommendation to use ENDF/B-VII.0 as the source of nuclear data. If this library is not used then the second preferred choice is JEFF-3.1.1. This is likely to attempt to ensure that differences seen are due to the different methods employed rather than some combination of methods and data.

3.2.8 Benchmark Studies Summary

147. Clearly, significant international effort has been expended in these benchmark studies. The output from this work should be of very great interest to licensees and regulators alike, when considering the application of BUC. However, some caution must be raised when considering the vintage of the code and nuclear data combinations in some of these benchmark studies.

3.3 Validation of Burn-up Calculations

148. It is necessary to ensure that the analyses of the reactivity of a system involving burn-up credit have considered all major sources of uncertainty. The methods and data used to achieve this are similar, but different, for depletion and criticality codes. With regards validation of burn-up calculations this essentially involves calculation of SNF compositions for comparison with measured assay data, where the difference, or *bias*, in terms of k-effective in an application environment for calculated and measured compositions is examined. Analysis of multiple experimental data configurations permits estimation of this bias, and experimental uncertainty on this bias (i.e. *bias uncertainty*), associated with the burn-up calculation, using standard statistical methods. This will include a degree of uncertainty arising from basic nuclear data, in terms of fission product yield data, neutron cross-section data used in neutron flux calculations,

etc. For criticality codes the uncertainty analyses usually involves comparison of calculated and experimental (or benchmark) values of k-effective for critical, subcritical or exponential experiments. The experiments considered are required to share neutronic and physical characteristics with the application for which the uncertainty is to apply. With respect to BUC this would ideally require cases involving the actinide and fission products listed in Table 1 and Table 2, and any additional isotopes identified as important for a specific application.

149.   The bias and bias uncertainty associated with the criticality calculation will, again ideally, be calculated on the basis of multiple experimental comparisons and should depend largely on neutron cross-section data. Since experiments are unlikely to exist for all such applications further consideration of nuclear data uncertainties is required to estimate bounding effects on k-effective for missing experimental data. The following sections will describe the various methods available to perform such validation. No recommendations are made as to a preferred option to utilise, as this will be the responsibility of the licensee to demonstrate on a case by case basis.

150.   As has previously been noted assay data are essential to underpin the validation of burn-up calculations. In addition to criticality aspects assay data are also important to many different areas of SNF safety and management. As well as for criticality purposes, SNF compositions are utilised for evaluation of radioactivity, neutron and gamma ray source terms and decay heat. It is therefore necessary for the assay data to be sufficient to validate a variety of design and safety evaluations for the whole nuclear fuel cycle i.e. fuel handling, dry spent fuel storage installations, pool storage, fuel reprocessing facilities and waste repository studies. With respect to criticality safety the essential information is clearly the spent fuel isotopic composition. Assay data, obtained through PIE measurements, provide isotopic composition of the spent fuel along with sufficient data for the fuel design, reactor operating conditions and irradiation history. These latter data are essential to allow precise modelling of the fuel burn-up.

151.   It is worth noting that for validation studies it should be expected that the computational model will use as close a representation to the experiment, in terms of irradiation conditions, in order to minimise bias and bias uncertainty values. However, when the application calculation is performed this is likely to consist of a more generic calculation, utilising worst case conditions for temperature, density, flux profile etc. Although any potential bias and bias uncertainty resulting from the closest representation approach (probably attributable to the nuclear data) will still apply, the generic optimised approach should be demonstrated to provide pessimistic results and not introduce additional uncertainty that may challenge safety margins.

3.3.1   Methodologies

152.   There are a number of methodologies for calculating the bias and uncertainty associated with depletion calculations. These are outlined in Reference 12 and more detail provided on two preferred options in Reference 13. Reference 12 describes the following techniques:

    i.   Bounding method;

    ii.   Monte Carlo uncertainty method;

    iii. Sensitivity/uncertainty method; and

    iv. "Direct difference" method.

153.   All but the first of these methods are described as best-estimate approaches. The bounding method utilises the differences in measured and calculated isotopic compositions for each nuclide and involves reactivity calculations with worst case compositions. That is, all of the nuclides that provide a positive contribution to reactivity will be at the minimum value and all

absorbing nuclides will be at their maximum value. Clearly, the variability of nuclide compositions will not be in the direction that results in the most reactive system. Therefore, this methodology leads to a bounding value that does not really represent an uncertainty in the usual sense, being a one-sided limiting margin with no associated variance. Being bounding, it leads to an overestimation of the system reactivity and hence underestimates the real safety margin of a system. Since one of the purposes of utilising a burn-up credit methodology in preference to the fresh fuel approach is to obtain a more realistic representation of safety margins this approach is less well favoured then best-estimate approaches.

154. The Monte Carlo uncertainty methodology consists, as the name suggests, of a Monte Carlo (probabilistic) sampling approach to compositional uncertainties. To use this methodology multiple calculations are performed with changes to the input parameters reflecting the random uncertainty variation for each parameter, in this case nuclide concentration. The uncertainty attributed to each nuclide concentration is derived from the predicted concentration for a range of PIE samples. As can be assumed, this approach requires substantial computing resource, and automated techniques to apply. The development of modern computer codes allows this type of calculation to be performed. Reference 12 indicates that the KRONOS system has been developed for use with the SCALE sequence and in the UK future releases of MONK will include the SPRUCE tool for such analyses.

155. The sensitivity/uncertainty method involves the use of calculated sensitivity coefficients, in combination with estimated uncertainties (again derived from difference between nuclide composition calculated and measured) on a nuclide-by-nuclide basis. Most modern criticality computer codes have the ability to calculate these sensitivity coefficients. They are essentially the change in k-effective resulting from a 1% change in the cross-section data, and can be calculated over a specified energy range, for a range of reactions (such as capture, fission, inelastic scatter, elastic scatter, 'n,2n', $\bar{\nu}$) and at a material or nuclide level. If the total nuclide sensitivity is calculated then nuclides with a positive sensitivity coefficient show net neutron production (i.e. an increase in the concentration increases the neutron multiplication factor), while negative coefficients show net neutron absorption. Although the calculation of these sensitivity coefficients can involve significant additional overheads, in terms of memory requirements and computing time, the methods can be applied to any model.

156. The final methodology is the direct difference method. In this methodology as close a representation of the irradiated sample conditions are modelled. The compositions from the depletion calculation are then included in a donor model, representing the application geometry, and k-effective calculated. The difference between this calculated k-effective and unity calculated with the measured composition gives a direct indication of the bias associated with the depletion calculation. This will essentially be a reactivity bias associated with all of the uncertainties associated with the parameters used in the burn-up calculation, e.g. power rating, temperature, moderator density, fission yields etc. These components are not separable and may well incorporate a degree of cancellation of effects.

157. Reference 13 provides detailed examples of the application of the direct difference and Monte Carlo uncertainty methodologies to realistic storage configurations. Representative models were developed to cover PWR Spent Fuel Pond (SFP) storage rack, PWR SNF transport cask, and BWR SFP storage rack arrangements. The output from the study reported in Reference 13 is the list of k-effective bias uncertainties provided in Reference 3 for PWR SNF systems. It is noted that positive biases are calculated in Reference 13, for each of the irradiation periods considered. These are not included in the derived criticality safety criterion, in keeping with standard criticality safety practice.

158. Regardless of the specific methodology adopted an interpretation of measured assay data are required. Sources of such data will be considered in the next section.

3.3.2 Sources of Data

159. PIE programmes provide the main source of data for the validation of depletion calculations. There are a number of these that are freely available, and it is to be assumed that operators will have access to their own proprietary assay data.

160. Reference 14 provides details of experimental programmes relating to PIE and fission product validation. Reference 14 is an extensive document, produced by the NEA EGADSNF, and it is not proposed to reproduce significant proportions of that document here. While brief descriptions of the available data are provided the reader is directed to Reference 14 for more detailed information. The programmes described in Reference 14 are known as CERES, ARIANE, MALIBU, REBUS, and PROTEUS. Although some of these programmes are covered by commercial restrictions, there is also the SFCOMPO [Reference 20] database of PIE data.

161. It is noted that no attempt has been made in this review to identify whether duplicate information is present in the following data sources, e.g. whether any of the specific programmes listed below are also included in SFCOMPO, or any other code-specific validation.

3.3.2.1 Experimental Programme: CERES

162. In the early 1990s AEA Technology (UK) and the CEA (France) began a series of experiments and analyses known as the CERES collaboration. The main aim of the programme was to provide experimental data to validate calculation methods and nuclear data used in criticality assessments involving burn-up credit calculations. While these experiments generally consist of reactivity worth measurements, they did involve the use of samples generated from PWR SNF. Therefore, the entire analysis route included the requirement to perform depletion calculations to allow estimation of the reactivity difference between calculated and measured sample composition data.

163. The CERES programme consisted of three distinct phases. Phase I of the programme involved reactivity measurements on irradiated PWR fuel samples in the MINERVE reactor at Cadarache and in the DIMPLE reactor at Winfrith. In Phase II reactivity measurements were made on a set of fission product samples to provide validation of the neutron absorption cross-section data for thirteen of the major fission products relevant to burn-up credit. Phase III, the final part of the programme extended the collaboration to include the participation of the United States through Sandia National Laboratories. In addition, destructive chemical analysis on selected PWR, BWR, CAGR and MOX fuel samples was carried out at Harwell to determine the concentration of the following fission product nuclides: ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{143}Nd, ^{145}Nd, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{153}Eu and ^{155}Gd.

164. While the original work programmes were covered by commercial confidentiality agreements some of the data are being released as an International Reactor Physics Experiment Evaluation Project (IRPhEP) evaluation. This document is currently at draft evaluation status. While previous analyses of CERES reactivity worth measurements incorporated 2D first order perturbation analysis techniques the aim of the IRPhEP evaluation is to provide a 3D representation of the experiments. It is known that the area of 3D Monte Carlo perturbation analyses is currently undergoing development so the value of these experiments in support of methods and validation can only increase.

3.3.2.2   Experimental Programme: ARIANE

165.   Actinide Research in a Nuclear Element (ARIANE) is an experimental programme coordinated by Belgonucléaire (BN). The programme involves extensive measurements covering the nuclides important to burn-up credit, decay heat and radiation source terms. BN act as the programme coordinator, but the programme itself has involved several radiochemical analysis laboratories across Europe, including Studiecentrum voor Kernenergie-Centre d'étude de l'Energie Nucléaire (SCK•CEN) in Belgium, the Institute for Transuranium Elements (ITU) in Germany, the Paul Scherrer Institute (PSI) in Switzerland and the Commissariat à l'énergie atomique et aux énergies alternatives (CEA) in France. Assay measurements have been performed by each of these organisations in order to provide independent cross-check measurements on fuel samples to reduce uncertainty and independently confirm the accuracy of the data. The ARIANE programme involved measurements from three reactors and different fuel assembly designs and types. These were Gosgen, a PWR with samples taken from 15×15 UO_2 assemblies, Beznau-1, a PWR with samples taken from 14×14 MOX assemblies, and Dodewaard, a BWR with samples taken from 6×6 MOX and UO_2 assemblies.

3.3.2.3   Experimental Programme: MALIBU

166.   The "Radiochemical Analysis of MOX and UOX LWR Fuels Irradiated to High Burn-up" (MALIBU) programme is a follow-on programme to ARIANE. Its aim is to extend the available assay data to high burn-up UO_2 and MOX fuel from both PWR and BWR. However, this programme contains proprietary information, and so the details are not freely available. It is however known that the programme will again involve samples from the Gosgen PWR with samples taken from 15×15 UO_2 and MOX assemblies and with a maximum burn-up of ~70 GWd/t. Also, samples will be taken from the Gundremmingen BWR with samples from 9×9 MOX assemblies again with a maximum burn-up of ~70 GWd/t.

3.3.2.4   Experimental Programme: REBUS

167.   The Reactivity Tests for a Direct Evaluation of the Burn-up Credit on Selected Irradiated LWR Fuel Bundles (REBUS) programme is another BN programme. This programme was designed to provide validation of computer codes for criticality calculations involving burn-up credit. The REBUS programme included destructive analysis measurements of the irradiated fuel nuclide concentrations and measurement, in the SCK•CEN VENUS critical facility, of reactivity reduction associated with irradiation. Responsibility for the management of the BN experimental programmes was transferred to SCK•CEN in 2008. The participating countries were Belgium, France, Germany, Japan and the US. Experiments were initially performed for reference configurations using fresh fuel and for irradiated UOX fuel at 54 GWd/t average burn-up. Subsequent experiments were performed for reference configurations using fresh fuel and for irradiated MOX fuels at 20 GWd/t burn-up.

3.3.2.5   Experimental Programme: PROTEUS

168.   The LWR-PROTEUS experimental programme was performed at PSI in Switzerland for the purposes of providing experimental data for validation of computer codes and reactor physics parameters. The isotopic assay phases of the programme involved many series of measurements on spent fuel from the Swiss Leibstadt BWR and Gosgen PWR nuclear power plants. Under Phase II of the programme, a total of thirteen fuel samples were destructively measured; eleven PWR samples and two BWR samples. In total seventeen actinides and forty

fission products were measured in the LWR-PROTEUS experimental programme by PSI. However, the LWR-PROTEUS programme is commercially protected and the data are not currently available.

3.3.2.6   SFCOMPO

169.   The current version of SFCOMPO [Reference 20] is an OECD/NEA maintained database of measured spent fuel composition data that is free from commerciality restrictions. Originally the database was compiled, in paper form, by the Japan Atomic Energy Agency (JAEA) in the later 1990s. The EGBUC recognised the significance of having a centralised framework for the collection and dissemination of assay data from NEA member countries, and so SFCOMPO was transferred to the NEA Data Bank in 2002. The database infrastructure is being developed to enable controlled World Wide Web access. However the current approach requires an underpinning CD/DVD containing the detailed experiment references. At this stage, with SFCOMPO 2.0 available for evaluation purposes on request, this CD/DVD is not available from OECD/NEA. No timescale has been given for a formal release of SFCOMPO 2.0.

170.   It is stated that the current version, SFCOMPO 2.0, contains experimental data coming from 44 different reactors of eight different international types, representing more than 730 fuel samples. The database currently contains more than 15900 isotopic concentration measurements.

171.   It should be noted that the data published by the various contributing organisations is included "as reported". There is currently no independent peer review of the data. These reviews would be necessary to assess the quality of the measurements, identify potential problems with data as determined through inconsistencies in the data and through comparisons of measurement data for other similar fuel types.

172.   It is acknowledged that the task of actually compiling and applying the experimental data for code validation can be very time consuming and frequently requires the user to search the primary reports to confirm details of the experiment and verify reported data. Reference 14 therefore suggests the formation of an evaluation working group to perform the compilation and review functions to ensure the best available data are consistently used in the industry, rather than each organisation having to repeat this exercise. This is something that may be of interest to members of the UK industry interested in BUC, but also to Regulators who, through **BUC23**, will be required to assess the validity of the data used to support code validation.

3.3.2.7   Additional Data

173.   The reader is directed to Reference 14 where further information is provided covering experimental programmes in France, Germany, Japan, Spain, Sweden, UK and US.

174.   However, one of the areas Reference 14 provides further information is with regards codes, and their supporting validation, that have been used within the UK nuclear industry. It is stated that Britain has an active history in performing SNF assay experiments and validation using the UK code FISPIN [Reference 21]. Further, the UK National Nuclear Laboratory holds a large database of measurements, collected over 30 years, covering UK nuclear fuels. These data are mostly for Magnox and AGR fuel, but some BWR and PWR data are included. The database has comparisons of experimental data with calculations using JEF2.2 based nuclear data libraries for the WIMS, TRAIL and FISPIN10 code route [Reference 22]. The FISPIN10 validation database covers major actinides (^{234,235,236,238}U, ^{238,239,240,241,242}Pu), curium isotopes ($^{242,\ 244}$Cm), caesium isotopes (^{134,137}Cs), neodymium isotopes ($^{142-146,\ 148,\ 150}$Nd), europium isotopes (^{154,155}Eu), and noble gases ($^{80,\ 82-86}$Kr and $^{128-132,\ 134,\ 136}$Xe).

175. From a US perspective, as described in Section 3.3.1, Reference 13 provides detailed examples of the calculation of k-effective bias and bias uncertainties for realistic storage configurations. This document utilised Reference 23, as a source of compiled experimental data. In fact this document is another example of applying the experimental data to depletion validation, in this case for an earlier version of SCALE, namely version 5.1. However, this document contains details of other PWR PIE data for reactors Calvert Cliffs, Takahama and Three Mile Island.

176. Details of the validation of the French DARWIN code sequence can be seen in Reference 26. The DARWIN package was developed by the CEA and its French partners (COGEMA, EDF and FRAMATOME). It provides the required parameters necessary for fuel cycle applications and in characterising the irradiated fuels from reactors, i.e. fuel inventory, decay heat, activity, neutron, γ, α, β sources and spectra, and radio-toxicity. The validation of the code system for spent fuel inventory calculations relies on a large experimental programme performed in France, based on SNF chemical assay measurements. The experimental data are based on chemical analysis measurements from fuel rod cuts irradiated in French PWR reactors and from full assembly dissolutions at the COGEMA/La Hague reprocessing plants. The validation data covers a large range of UOX fuels with various enrichments in ^{235}U (3.1% to 4.5%) and burn-up values from 10 GWd/t to 60 GWd/t. In addition MOX fuels have also been investigated, with an initial Pu content in the central zone of 5.6% and a maximum burn-up of 45 GWd/t. Table 7 of Reference 14 provides full details of the experiments covered and the uranium, plutonium, americium, curium and fission product nuclides considered.

3.4 Validation of Criticality Calculations

177. As described above, for criticality codes the uncertainty analyses usually involves comparison of calculated and experimental (or benchmark) values of k-effective for critical, subcritical or exponential experiments. The experiments considered are required to share neutronic and physical characteristics with the application for which the uncertainty is to apply. With respect to BUC this would ideally require cases involving the actinide and fission products listed in Table 1 and Table 2.

178. The bias and bias uncertainty associated with the criticality calculation will, again ideally, be calculated on the basis of multiple experimental comparisons and should depend largely on neutron cross-section data. Since experiments are unlikely to exist for all such applications further consideration to nuclear data uncertainties are required to estimate bounding effects on k-effective for missing experimental data.

3.4.1 Methodology

179. There are a number of methodologies for calculating the bias and uncertainty associated with criticality calculations. From a US perspective the preferred approach is detailed in Reference 17. However, in the UK the ONR do not impose a validation methodology on licensees, merely stating that validation is necessary. Various UK organisations have therefore justified their own particular approach. These are largely based on statistical analysis of comparisons of calculated to experimental (or benchmark) results for a variety of selected application relevant systems.

180. As discussed in Section 2.2.4, the criticality safety criterion generally reduces to:

$$\text{k-effective} + 3\sigma \leq 0.95 - E_{PD} \qquad \text{(2 from Section 2.2.4)}$$

181. E_{PD} is the error in physical data and is usually calculated on the basis of a selection of applicable benchmark experiments. However, it can also be split into additive components, where E_{PD1} would represent the usual typical value calculated from the comparisons between calculation and experiment, and E_{PD2} that would cover additional uncertainty from sources that are outside the available validation range.

182. The vast majority of available validation benchmark experiments cover fresh fuel, be that uranium metal or oxide, mixed oxide, or plutonium metal. Therefore when considering actinide-only burn-up credit some coverage may be afforded by the mixed oxide lattice or solution experiments. The majority of the experiments performed with mixed oxide fuels have been to support applications for use of mixed oxide fuel in nuclear power reactors at the start and end-of-life, including transport. In addition, the mixed oxide cases would be appropriate to support reprocessing of irradiated uranium, with reasonable cooling periods of, typically, five to ten years. Experiments supporting these applications can therefore be selected relatively easily and a suitable value of E_{PD1} derived accordingly. In the UK current methods for selecting suitable experiments are generally through criticality assessor expert judgement. More sophisticated methods are under current development, involving similarity methodologies. This is where an automated approach is adopted, comparing system calculated nuclear data sensitivities with a library of experimental case nuclear data sensitivities. This is part of the methodology described in Reference 17, where the SCALE TSURFER code is used. From a UK perspective ANSWERS is developing a similarity tool that would perform a similar function, with the aim of supporting the criticality assessor's expert judgement rather than replacing it.

183. However, issues with data applicability arise for applications such as long term storage or final disposal where the evolution of the plutonium isotopic composition could take the fuel composition outside the range of applicability of the validation. To resolve this issue would require careful consideration of the evolution of the plutonium isotopic data over a suitable time frame and arguments to be presented concerning validation suitability or the addition of suitably derived value of E_{PD2}. The requirement to consider the evolution of the plutonium isotopic data are covered by Regulatory question ***BUC8*** and clearly this must include the requirement that suitable validation evidence is presented to cover the application (***BUC2***).

184. Similarly, for BUC applications based upon the actinide and major fission product methodology there is the issue that there are very few (and in some cases no) benchmark experiments that cover minor actinides and fission products. These categories would also require consideration of a suitably derived value of E_{PD2}.

185. The derivation of E_{PD2} will likely require an analysis similar to the sensitivity/uncertainty method described previously. Using this method the sensitivity to system reactivity for each isotope would be calculated and then their uncertainty, as derived using nuclear data covariance data, could be estimated. The combination of these calculated uncertainties could then be used to provide a bounding uncertainty on the reactivity worth of the nuclides not covered by conventional validation bias and uncertainties.

186. In summary to derive a suitable value of E_{PD} for a criticality calculation the following components would need to be included.

$$E_{PD} = E_{PD1} + E_{PD2} + E_{PD3} + E_{PD4} \qquad (5)$$

where

E_{PD1} is the bias and uncertainty derived from critical/sub-critical/exponential experiments,

E_{PD2} is the uncertainty derived from the isotopes not covered by E_{PD1},

E_{PD3} is the bias derived from the analysis of suitable depletion experiment analyses, and

E_{PD4} is the uncertainty derived from the analysis of suitable depletion experiment analyses.

187. Regardless of the specific methodology adopted an interpretation of critical/sub-critical/exponential experiments is required. Sources of such data will be considered in the next section.

3.4.2   Sources of Data

188. The main source of validation data are two compendiums compiled by the International Criticality Safety Benchmark Evaluation Project (ICSBEP) and the International Reactor Physics Experiment Evaluation Project (IRPhEP). These projects are operated as part of the OECD/NEA Nuclear Science organisation. They share the main aims of:

► maintaining an inventory of the experiments that have been carried out and documented;

► archiving the primary documents and data released in computer-readable form; and

► promoting the use of the format and methods developed and seek to have them adopted as a standard.

189. The two working groups support the:

► compilation of experiments into a standard international agreed format;

► verification of the data, to the extent possible, by reviewing original and subsequently revised documentation, and by consulting with the experimenters or individuals who are familiar with the experiments or the experimental facility;

► analysing and interpreting the experiments with current state-of-the-art methods; and

► publishing electronically the benchmark evaluations.

190. The result of this final bullet point is Reference 24 and Reference 25.

191. The ICSBEP Handbook [Reference 24] currently contains 4874 experimental descriptions, categorised by fissile media (plutonium, highly enriched uranium, intermediate and mixed enrichment uranium, low enriched uranium, uranium-233, mixed plutonium-uranium and special isotope systems), physical form of the fissile material (metal, compound, solution and miscellaneous systems), and neutron energy range where the majority of the fissions occur (fast, intermediate, thermal and mixed spectra systems). To enable selection of experiments of interest a database front-end, known as DICE (**D**atabase for the **I**nternational **C**riticality Safety Benchmark **E**valuation Project), is maintained by the OECD/NEA. This database provides a summary description of each experimental configuration and a means of selecting experimental configurations of various system parameters. One of the system parameters that is being developed is the sensitivity coefficients for the major nuclides and nuclear processes. This will therefore be an invaluable tool in selecting experiments that match the applications for BUC.

192. The IRPhEP Handbook [Reference 25] currently contains 151 experimental series that were performed at 50 reactor facilities. Only 146 of these experimental series are approved benchmarks, with the remaining five published as draft evaluations. The experiments within the handbook are categorised on the basis of reactor name, reactor type, facility type, and measurement type. There is also a similar front-end database, known as the IRPhEP Database and Analysis Tool (IDAT) which again permits selection of experimental configurations on the basis of system parameters, again including sensitivity coefficients.

193. Using the DICE tool it is possible to extract the experiments currently within the database that involve fission products. These are one involving ^{149}Sm (LEU-COMP-THERM-050), another involving ^{103}Rh (LEU-COMP-THERM-079), and a third involving elemental Sm, Cs, Rh, and Eu (LEU-MISC-THERM-005). Clearly this scarcity of information will require the use of other techniques, as described above, to adequately consider bias and uncertainty for BUC applications.

194. Additional experiments such as the reactivity worth measurements performed in DIMPLE and MINERVE as part of the CERES analyses will also be of use once formally published via IRPhEP. The results of the CERES analysis will be discussed in the following nuclear data section of this report.

3.5   Nuclear Data

195. This section considers nuclear data of importance to burn-up credit.

3.5.1   Nuclear Data of Relevance to BUC

196. The term 'nuclear data' covers several types of physical data. Some data types are only important for determining the composition of spent nuclear fuel (SNF), whilst others are also important in determining the reactivity worth of SNF of known composition.

- ▶ Neutron cross-sections;
  - ▶ Neutron capture – important for SNF composition and reactivity worth;
  - ▶ Fission – important for SNF composition and reactivity worth;
  - ▶ Scattering (plus scatter matrices) – important for SNF composition;
  - ▶ Threshold reactions - (n,2n), (n,3n) etc. – important for SNF composition; and
  - ▶ Other reactions – not of great significance to the BUC nuclides considered.
- ▶ Neutron production;
  - ▶ Mean neutrons per fission ($\overline{\nu}$) – important for neutron production in SNF;
  - ▶ Fission spectra – BUC assessment will be insensitive to variations in this data; and
  - ▶ Delayed neutron data – BUC assessment will be insensitive to variations in this data.
- ▶ "Burn-up" data;
  - ▶ fission product yields – important for determining SNF composition;

- ► half-lives or decay constants– most important for determining SNF composition (during reactor operation and short-term cooling). Important for the short-lived nuclides considered for burn-up credit. (Note: decay constant = 0.6931 / half-life);

- ► decay branching ratios – relevant to the SNF concentration of a small subset of nuclides considered for burn-up credit; and

- ► capture branching ratios – potentially relevant to the SNF concentration of some actinides (due to capture in ^{241}Am to form ^{242}Am/^{242m}Am) and Eu or Sm isotopes (due to capture in ^{151}Eu to form ^{152}Eu/^{152m}Eu).

► Spontaneous fission sources – important in operation, more so in SNF (see BUC36 below); and

► Covariances for cross-sections, $\overline{v}$ and fission spectrum.

BUC36. Has the licensee demonstrated that spontaneous fission effects will not be significant, i.e. there are negligible quantities of nuclides such as ^{252}Cf. Otherwise, does the application discuss the worth of spontaneous fission in the SNF?

197.   To investigate the suitability of the above 12 actinides and 16 fission products for burn-up credit, the consistency of the underlying nuclear data for these nuclides has been assessed. The neutron cross-sections, fission product yields and half-lives have been compared for three nuclear data evaluations (see Section 3.5.3).

198.   However, it must be noted that the nuclear data relevant to determining spent fuel compositions is substantial and impractical to consider in full detail. It is difficult to determine the major production paths of the nuclides of interest, as these are system dependent, and therefore what nuclear data are the most important.

3.5.2   Nuclides under Consideration

199.   Reference 3 (see also Reference 13 Section 3.1) recommends 12 actinides and 16 fission products for burn-up credit, with reference to commercial PWR spent nuclear fuel.

200.   Several studies have been performed to identify the nuclides that have the most significant effect on the calculated value of k-effective as a function of burn-up and cooling time. These results are summarised in NUREG/CR-6665 [Reference 1].

201.   This report concludes that the actinides and fission products are candidates for inclusion in burn-up credit analyses for storage and transportation systems, based on their relative reactivity worth at the cooling times of interest. The report notes that relative reactivity worth of the nuclides will vary somewhat with fuel design, initial enrichment, and cooling time, but the important nuclides remain the same and have been substantiated by numerous independent studies. These nuclides have the largest impact on k-effective and there is a sufficient quantity of applicable experimental data available for validation of the analysis methods. Accurate prediction of the concentrations for the nuclides requires that the depletion and decay calculations include nuclides beyond those listed. Additional actinides and fission products are needed to assure the transmutation chains and decay chains are accurately handled.

202.   The nine nuclides for "actinide-only" burn-up credit are given in Table 1 and a further nineteen nuclides (three minor actinides and sixteen fission products) for "limited actinide and fission product" burn-up credit are given in Table 2. To reiterate Paragraph 10 (Section 2.1), these nuclides have been chosen by the international community for very specific reasons; firstly due to large neutron capture or fission cross-sections with large reactivity worth. The actinides

contribute a significant positive reactivity. The selected fission products, being non-volatile, non-gaseous and stable/long-lived, provide a significant proportion of negative reactivity. Secondly, there are sufficient experimental data available for validating the calculation of the concentrations and reactivity worths of these nuclides in SNF. Note that there are significantly more validation data for the nuclides in Table 1 compared to those in Table 2.

BUC37. Is the licensee claiming nuclides other than those recommended in Reference 3? Does the licensee present validation for each of the nuclides being claimed?

BUC38. Is the licensee claiming BUC for MOX fuels? Are additional nuclides such as ^{242m}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm being claimed? Does the licensee present validation for these nuclides?

BUC39. Are there additional fissionable actinides in the system being evaluated which are not considered in the analysis, and which could result in underestimation of k-effective?

3.5.3 Nuclear Data Evaluations under Consideration

203. The nuclear data used in application codes originates from evaluated data files produced by a number of international nuclear data projects, primarily, JEFF (Europe), ENDF/B (US) and JENDL (Japan).

204. Improvements in the accuracy of nuclear data lead to increased accuracy in calculations. This may in turn enable a reduction in the error allowances. In some cases, it is not just that data are improved but data entirely absent are provided. These improvements occur incrementally as new evaluations for individual nuclides are produced. New evaluations are collected together to form periodic releases of the main evaluated data projects, typically every 3-5 years.

205. Key evaluations are:

   ► JEF-2.2 (1992);
   ► JEFF-3.1.2 (2012);
   ► JEFF-3.2 (2014, decay data not updated);
   ► JEFF-3.3 (expected 2017);
   ► ENDF/B-VII.0 (2006);
   ► ENDF/B-VII.1 (2011); and
   ► ENDF/B-VIII.0 (expected 2017).

206. JEF-2.2 nuclear data are still widely used in UK industry, with JEFF-3.x data experiencing increased levels of usage. There have been numerous updates to the JEFF evaluations since JEF-2.2, therefore, the latest public release, JEFF-3.2 is compared to JEF-2.2 below. To offer some degree of independence of the JEFF project, the ENDF/B-VII.1 evaluation is also compared.

207. As might be expected, successive evaluations within the same project (e.g. JEFF or ENDF/B), will often 'inherit' some datasets from preceding evaluations. What may be less obvious is that different projects may adopt new data from a common source or from a different project (e.g. JEFF-3.3 may take a dataset from ENDF/B-VII.1 if it is considered better quality than the JEFF-3.2 equivalent).

208. Modern nuclear data evaluations are significantly more complete than older evaluations in terms of the coverage of nuclides and types of data. The assembly of more complete datasets allows for an improved level of self-consistency within the dataset as well as the ability to

validate the dataset as a whole rather than individual components. Historically (up to the early 1990's), evaluations suffered more from incomplete coverage, leading to a more frequent requirement for the end-user to assemble the necessary data for specific applications.

BUC40. Does the licensee provide details of the nuclear data set(s) used in supporting calculations? If the nuclear data are from a recognised source, are references to supporting documentation and validation work provided? If the nuclear data are taken from multiple sources, are secondary sources documented?

BUC41. Is evidence provided that the nuclear data set is validated for the reactor type under consideration? Is there evidence of validation for burn-up credit assessment? If not, what allowances are made for use of un-validated data?

3.5.4 Review of Neutron Cross-sections

209. The neutron absorption cross-sections of a nuclide affects its abundance during operation (in terms of removal from the system) as well as the reactivity worth for a fixed concentration of the nuclide in SNF. Therefore, reliable cross-section data are required in consideration of burn-up credit.

210. Table 6 provides thermal average absorption (= capture + fission + (n,2n) + (n,3n) etc.) cross-sections for the nuclides. The table gives the relative difference between the evaluations (with differences >5% highlighted), or indicates where evaluations are the same. Similarly, Table 7 provides absorption cross-section integrals for the nuclides. They are calculated as the integral of the cross-section over the energy range 0.5 eV to 100 keV, weighted over a 1/E spectrum. As above, the table gives the relative difference between the evaluations. For the actinides, Table 8 gives thermal average fission cross-sections and Table 9 gives fission cross-sections integrals. In the tables, the relative difference between the given evaluation and JEFF-3.2 is denoted by Δ.

211. The differences in the cross-sections for each nuclide are summarised as follows;

   ► ^{95}Mo: reasonable consistency between the JEF/JEFF and the newer evaluation provided in ENDF/B-VII.1;

   ► ^{99}Tc: reasonable consistency between the JEFF-3.2 and ENDF/B-VII.1 datasets from separate evaluations performed 2008-2010;

   ► ^{101}Ru: the ENDF/B-VII.1 capture cross-sections are significantly higher at thermal energies than those for JEFF-3.2. The difference of >50% is much larger than typical nuclear data uncertainties;

   ► ^{103}Rh: reasonable consistency between the JEFF-3.2 and ENDF/B-VII.1 datasets from separate evaluations;

   ► ^{109}Ag: JEFF-3.2 maintains the data provided in JEF-2.2 but is in good agreement with the 2010 re-evaluation provided in ENDF/B-VII.1;

   ► ^{133}Cs: good consistency between the JEF/JEFF and the newer evaluation provided in ENDF/B-VII.1;

   ► ^{143}Nd: JEFF-3.2 maintains the data provided in JEF-2.2 but is in reasonable agreement with the more recent (1999) evaluation provided in ENDF/B-VII.1;

   ► ^{145}Nd: JEFF-3.2 maintains the data provided in JEF-2.2 but is in reasonable agreement with the 2010 re-evaluation provided in ENDF/B-VII.1;

   ► ^{147}Sm: JEFF-3.2 maintains the data provided in JEF-2.2 but is in good agreement with

the 2006 re-evaluation provided in ENDF/B-VII.1;

► ^{149}Sm: All three datasets are in good agreement. Given the magnitude of the thermal average cross-section, the 5% discrepancy may not have a significant effect on the overall ^{149}Sm neutron capture rate;

► ^{150}Sm: JEFF-3.2 maintains the data provided in JEF-2.2 but is in good agreement with the 2006 re-evaluation provided in ENDF/B-VII.1;

► ^{151}Sm: JEFF-3.2 maintains the data provided in JEF-2.2 but is in good agreement with the 2006 re-evaluation provided in ENDF/B-VII.1;

► ^{152}Sm: JEFF-3.2 maintains the data provided in JEF-2.2 but is in good agreement with the 2006 re-evaluation provided in ENDF/B-VII.1;

► ^{151}Eu: JEFF-3.2 maintains the data provided in JEF-2.2 but is in reasonable agreement with the 2004 re-evaluation provided in ENDF/B-VII.1;

► ^{153}Eu: the JEFF-3.2 data have minor changes from JEF-2.2. The ENDF/B-VII.1 thermal absorption is significantly higher than JEFF-3.2; the 14% difference is at the upper end of typical nuclear data uncertainties;

► ^{155}Gd: JEFF-3.2 maintains the data provided in JEF-2.2 (with minor changes) but is in very good agreement with the 2006 re-evaluation provided in ENDF/B-VII.1;

► ^{234}U: JEFF-3.2 and ENDF/B-VII.1 share a 2006 re-evaluation which significantly reduced the small fission cross-section from JEF-2.2 (therefore reducing the thermal absorption cross-section);

► ^{235}U: one of the most measured and evaluated nuclides, the cross-sections are consistent across the evaluations;

► ^{236}U: the evaluation appear generally consistent except for the thermal average fission cross-section. Here, the JEFF-3.2 is ~23% higher than the other two evaluations. However, ENDF/B-VII.1 states a 20% uncertainty on this small (~50 mb) cross-section;

► ^{238}U: one of the most measured and evaluated nuclides, the cross-sections are consistent across the evaluations, though there is variation in the very low fission cross-section;

► ^{237}Np: the absorption cross-sections are generally consistent with JEFF-3.2 and JEF-2.2 being derived from the same source. However, the small fission cross-section is re-evaluated in ENDF/B-VII.1 resulting in a large change to the fission cross-section;

► ^{238}Pu: Discrepancies exist between JEF-2.2, JEFF-3.2 and ENDF/B-VII.1 with ENDF/B-VII.1 taking the most recent re-evaluation (2007) from the JENDL project. Note the short half-life of ^{238}Pu leads to difficulties measuring it's cross-sections;

► ^{239}Pu: one of the most measured and evaluated nuclides, the cross-sections are consistent across the evaluations;

► ^{240}Pu: absorption cross-sections are consistent across evaluations, however, re-evaluation of the small fission cross-section has produced discrepancies;

► ^{241}Pu: Evaluations are consistent as JEFF-3.2 and ENDF/B-VII.1 share a re-analysis of the data provided in JEF-2.2. However, measured cross-section data are likely scarce due to the very short half-life;

► ^{242}Pu: similar to ^{240}Pu, absorption cross-sections are consistent across evaluations, however, re-evaluation of the small fission cross-section has produced discrepancies;

► ^{241}Am: Discrepancies exist between JEF-2.2, JEFF-3.2 and ENDF/B-VII.1 with JEFF-3.2 and ENDF/B-VII.1 being more recent re-evaluations; and

► ^{243}Am: Discrepancies exist between JEF-2.2, JEFF-3.2 and ENDF/B-VII.1 with JEFF-3.2 and ENDF/B-VII.1 being more recent re-evaluations.

3.5.5   Review of Half-lives

212. The half-life of an unstable nuclide affects its abundance (in terms of removal from the system) both during and post operation. Nuclides with a short half-life are not of interest for burn-up credit whilst those with an uncertain half-life will be more difficult to account for in considering burn-up credit.

213. Table 10 compares the half-lives of the 12 actinides and 3 unstable fission products for different evaluations. The remaining 13 fission products are considered stable (meaning they are theoretically stable and/or are so long-lived as to have no measurable decay rate). JEFF-3.1.1 is the most current release of the JEFF decay data library. It is noted that the ENDF/B-VII.1 data are, in the majority, unchanged from ENDF/B-VII.0.

214. Consideration of the values in Table 10 shows that;

► ^{99}Tc is long-lived so the variation in evaluated half-life values (range ~5%) will have little effect on the ^{99}Tc concentration (<0.5%);

► ^{151}Sm: there is some variation in the ~90 year half-life. This could be accounted for by assuming a lower concentration of ^{151}Sm in the SNF, thereby giving a conservative reactivity worth;

► The majority of the actinides are sufficiently long lived that the choice of data source will have negligible impact on BUC assessment: the evaluations give consistent values and the half-life uncertainties are better than 1%. This also applies to the long-lived ^{147}Sm fission product;

► ^{238}Pu: the short half-life (87.7 y) is the same (within rounding error) for all evaluations with an uncertainty better than 0.4%, which is based on a number of experimentally determined values. Therefore, the choice of data source will have no impact on BUC assessment;

► ^{241}Pu is short-lived (~14.3 year half-life) and there is some variation in the evaluated half-lives. This variation will affect the calculated ^{241}Pu and ^{241}Am content of cooled SNF; and

► Whilst ^{241}Am is relatively short-lived (~433 y), the evaluations give half-lives consistent to 0.2% with uncertainties of better than 0.2%. The choice of evaluation in the BUC assessment will not be significant compared to the half-life of ^{241}Pu which decays to ^{241}Am.

3.5.6   Review of Fission Product Yields

215. The fission yields of the fission products affect their production during operation. Nuclides with a low or uncertain yield will be more difficult to account for in considering burn-up credit.

216. The fission product yields for thermal neutron induced fission in ^{235}U are compared in Table 11 and those for ^{239}Pu are compared in Table 12. The latest JEFF release (JEFF-3.1) is compared to ENDF/B-VII.1 and JEF-2.2.

217. The evaluations contain two yield values for each fission product; the independent yield, being the yield coming directly from the fission event, and the cumulative yield, being the independent yield plus the independent yields of all parent fission products. For this comparison, the cumulative yield is more relevant to post-shutdown end-of-life fuel, as all short-lived parent nuclides will have decayed to the longer-lived daughter products being considered here. However, the cumulative yield cannot account for the neutron absorption in parent nuclides during reactor operation (and thereby an effective reduction in the cumulative yield).

218. The largest discrepancy in the fission yield data is for ^{150}Sm. Additionally, the yields are small with large uncertainties (35-64%). However, the major production route for ^{150}Sm is not directly from fission but from neutron capture in ^{149}Sm which is a significant neutron poison. The fission yield data for ^{109}Ag also suffers from discrepancies (around ±10%) with uncertainties up to 64% as does ^{155}Gd (for ^{239}Pu fission) with uncertainties of 11-17% JEF-2.2 being discrepant by +19%.

219. Overall, the cumulative fission product yields for end-of-decay-chain nuclides (which include the nuclides considered for BUC) are consistent across evaluations with discrepancies for specific nuclides rather than systematic differences.

3.5.7   Consideration of Uncertainty Data

220. The uncertainties provided with the evaluated nuclear data (covariance matrices for cross-sections and $\bar{v}$ can be used to quantify the effect of nuclear data uncertainties on quantities of interest. It is recognised internationally that the covariance data are sparse and all the major nuclear data projects are planning to increase the quantity and quality for future releases, and also to address notable gaps.

221. It is useful to consider the nuclear data uncertainties from the covariance data for these nuclides. It is difficult to state an overall uncertainty from the covariance matrices given in the evaluations. However, sensitivity methods such as those discussed in Paragraph 155 (Section 3.3.1) can be combined with the covariances to provide an estimate of the overall uncertainties for each nuclide in the system of interest.

BUC42. Does the licensee support their application with an estimate of uncertainty on reactivity worth (arising from nuclear data and other uncertainties)? If not, are conservatisms employed to allow for nuclear data uncertainties?

3.5.8   Review of Benchmark Exercises for Spent Fuel Compositions

222. The OECD BUC benchmark Phase I-B (Reference 18), already discussed in Section 3.2.1.2, summarises the results of a comparison of physics codes and data libraries in predicting spent fuel isotopic concentrations. To expand on the nuclear data aspects, the report concluded that most methods are within 10% agreement in their estimate of isotopic concentrations for actinides, with two exceptions; ^{238}Pu and ^{243}Am. The ^{238}Pu deviation was due to incomplete data in some cases. The ^{243}Am deviation (11%) appeared to also be the result of poor data, but the source of the deficiency was not clear. The 11% agreement for the fission had three significant exceptions; ^{149}Srn, ^{151}Sm, and ^{155}Gd. The variation was attributed to inconsistencies in cross-section and fission product yield data, particularly as there are a significant number of production paths for these isotopes, and all are sensitive to the fraction of ^{239}Pu fission. The large differences, for ^{155}Gd, were believed to be due to inadequate cross-section data for ^{155}Eu (though this may not be the only source of deficient data).

223. Further examination of the information given in Reference 18 highlights the following;

  ► It considers all 12 actinides and 15 of 16 fission products (not ^{151}Eu) of interest to BUC. It additionally considers ^{135}Cs;

  ► The 'incomplete' data for ^{238}Pu includes data for the Cm isotopes. This will influence the concentrations of ^{238}Pu and other Pu isotopes, and potentially ^{234}U as well, being the daughter of ^{238}Pu decay;

  ► The ^{155}Eu cross-sections are indeed significantly varied for earlier data evaluations. This discrepancy has been identified in several studies, including Reference 28. Modern evaluations, such as JEFF-3.2 and ENDF/B.VII.1 include improved data for ^{155}Eu;

  ► The benchmark calculations were performed in 1993-94, with the underlying nuclear data being derived before 1990; and

  ► Some calculations took data from one primary source with some additions, whereas other calculations used compilations of data from multiple sources. The major evaluations used were JEF-1, JEF-2.2, UKNDL, ENDF/B-IV, ENDF/B-V, ENDF/B-VI, JENDL-2 and JENDL-3. Other sources of data were also used. It is difficult to source the underlying datasets from this period. Therefore, a direct comparison of the data sets, including determination of the pedigree of the data, is impractical.

224. Reference 19 summarises the results the OECD/NEA Burn-up Credit Phase IV-B benchmark. As already discussed in Section 3.2.4.2, the exercise was conducted in a similar fashion to Phase I-B but considering MOX fuel assemblies. It considers all 12 'major' actinides and 15 of 16 fission products (not ^{151}Eu) of interest to BUC. It additionally considers the 'minor' actinides ^{242m}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm and ^{245}Cm.

225. The report concluded that excellent agreement between the calculations can be achieved for the major actinides, with three general exceptions. Relatively large differences were found for ^{234}U, ^{238}Pu and ^{237}Np for the case using MONK8A with JEF-2.2 data. It observes relative differences in the results for minor actinides of ±30%. For the fission products, relatively large differences were observed for ^{95}Mo, ^{103}Rh, ^{109}Ag, and ^{155}Gd.

226. Examination of the information given in Reference 19 highlights the following;

  ► Decay chains for ^{238}Pu and Cm isotopes are omitted from the MONK8A data. This may have significant bearing on the deviation in the MONK8A results for ^{234}U and ^{238}Pu;

  ► The differences in the results for minor actinides, may have a small impact on the concentrations of Pu isotopes in cooled fuel;

  ► The differences in ^{155}Gd concentrations are likely due to the variation in the ^{155}Eu data, noted above;

  ► The cause of the differences in ^{95}Mo, ^{103}Rh and ^{109}Ag concentrations is not clear, though one calculation set does use a JEF-1 based dataset which may be relatively incomplete compared to later datasets, such as those based on JEF-2; and

  ► The calculations were performed circa 2000. It may be difficult to source all the underlying datasets for these calculations. Therefore, a direct comparison of the data sets, is likely impractical.

3.5.9 Review of Validation Data for Fission Product Worth

227. The CERES experiments (References 28 and 30) are summarised in Section 3.3.2.1. Reference 31 reports relatively recent WIMS10 calculations of fission product worths

measured in the CERES experiments, with the WIMS10 reactor physics code and JEF-2.2 and JEFF-3.1 nuclear data.

228. Reference 31 found that 12 fission product reactivity worths are predicted, with JEFF-3.1 data, to within 11% of measured values, of which four are accurate to within the 4% estimated benchmark uncertainty (^{143}Nd, ^{145}Nd, ^{152}Sm and ^{155}Gd). The data for some nuclides (^{109}Ag, ^{145}Nd, ^{147}Sm, ^{152}Sm and ^{155}Gd) was maintained from JEF2.2 to JEFF3.1 so there was no improvement in the worth predictions for these nuclides. For others, including ^{95}Mo, ^{103}Rh and ^{153}Eu, new evaluations in JEFF-3.1 improved the results.

229. It is noted the results in this paper informed the re-evaluation of some nuclides, such as ^{95}Mo, ^{99}Tc- 95, ^{103}Rh, ^{109}Ag, ^{133}Cs and ^{145}Nd, which were included in the ENDF/B-VII.1 dataset, as noted under Paragraph 211.

3.5.10   Summary

230. Nuclear data relevant to determining spent fuel compositions is substantial and impractical to consider in detail. It is difficult to determine the major production paths of the nuclides of interest as these are system dependent.

231. Some comparison of the nuclear data itself has be done, however, validation cases can be looked at to identify nuclide production routes that show discrepancies due to inadequate nuclear data.

232. For determining the reactivity worth of a known composition of fuel, the nuclear data set is much more limited so this can be compared in better detail. Validation data for known fuel compositions can also be considered.

4 Assessment of Technology for Confirmation of Burn-Up or Reactivity

4.1 Introduction

233. In the application of burn-up credit as part of a criticality safety case, the criticality safety case will include a 'loading curve' or 'safe domain' condition that separates, for the particular fuel in question, the combinations of initial enrichment and burn-up that are acceptable from those that are not. Only SNF assemblies that fall within the 'safe domain' may be fed forward to the next stage in the handling of the SNF assembly (e.g. for loading into a transport cask or for addition to a dissolver vessel in a reprocessing plant). This relates to regulatory question BUC5.

234. Clearly, a crucial part of the safety case is how to determine whether or not a SNF assembly falls into the safe domain. There are essentially two main approaches that are adopted to make this determination.

235. The first approach is essentially to base the decision on the information on fuel burn-up that will have been generated during the operations of the reactor, i.e. to base the decision on the information on the fuel assembly supplied by the reactor operator. If this approach is adopted then, in order to ensure criticality safety, as well as the burn-up information the safety argument will include additional requirements such as visual examination of the assembly to confirm its identity and/or additional controls on fuel movements. Uncertainties in the reactor record values should be provided and justified to ensure confidence that the burn-up value being used for the determination of the assembly falls in the 'safe domain' [Reference 32]. It is recommended that this justification include comparisons with in-core measurements as in References 33 and 34. To date, reliance on reactor records for burn-up confirmation has mainly been applied to at-reactor storage of SNF.

BUC43. Has the licensee sufficiently justified the level of uncertainty assumed in the reactor data?

236. It is noted that such cases may be accepted by regulatory bodies in other countries, e.g. in the US and Germany. However, the requirements that are then placed on the systems to control fuel movements may make it unsuitable for safety cases following the approach set out in ONRs Safety Assessment Principles (SAPS) [Reference 35].

BUC44. If a BUC safety case has been presented without out-of-core verification methods is there sufficient information available from reactor records/measurements to confirm the burn-up is at least the limiting value required for a given assembly type and enrichment to the required degree of confidence?

237. The second approach is to base the decision on whether or not a SNF falls into the safe domain on the results of out-of-reactor measurements of the fuel assembly, generally using some measurement of gamma or neutron emissions from the SNF which can be related to fuel burn-up and/or the residual fissile content of the assembly. Several different types of measurement may be carried out, potentially at several different points of the SNF assembly in order to obtain a sufficient characterization of the assembly. Examples of measurement systems to support burn-up credit include the FORK detector examined in the US [Reference 36], the THORP Feed Pond Fuel Monitor (FPFM) in the UK [Reference 37], and in the measurements carried out at the COGEMA reprocessing plant in France [References 38 and 39]. Out-of-reactor independent measurements of the characteristics of the SNF are more often used in the

context of safety cases for SNF reprocessing, transportation and for geologic repository operations.

238. It should be noted that although the two methods have been identified separately for clarity of discussion, in practice there will generally be some overlap between the two. For example, even in a situation in which the sentencing of the fuel assemblies is based on reactor burn-up records, there may be a requirement for some independent measurements which, though incapable of giving a precise measurement of fuel burn-up would detect fuel that was grossly outside the safe domain (e.g. un-irradiated fuel). Conversely, the calibration of independent instruments used for fuel sentencing is generally based on the reactor records of the (particularly well characterised) fuel assemblies used in the calibration.

239. Out-of-reactor verification measurements incur costs, notably they have the potential to increase reactor downtime if measurements are performed on spent fuel at the reactor. Risks are also incurred, including increased dose to personnel, risk of fuel assembly damage and mishandling events due to increased fuel assembly movement [Reference 34]. Reference 34 states that these costs and risks must be weighed against the cost and risks of not performing the measurements. Delay-related costs associated with burn-up credit may be smaller for reprocessing applications, where reactor downtime is not a factor.

BUC45. Has the licensee identified the risks associated with burn-up verification (for example, but not limited to, increased operator dosage), and are they justified?

240. This section is structured as follows:

   i. A review of applications of burn-up credit in practice in the context of burn-up verification is presented;

   ii. Approaches to burn-up verification for burn-up credit by application is presented. This includes discussion of particular examples, where relevant;

   iii. Information on the individual verification method techniques are presented; and

   iv. Background is provided on specific individual verification systems that have not otherwise been discussed in the preceding sections.

4.2 Review of Applications of Burn-up Credit in Practice

241. Taking credit for the burn-up of individual fuel assemblies may also increase the potential and consequence from misloading events. Consequently, misloading events should be one of the key considerations in burn-up credit criticality safety assessments.

BUC46. Have sufficient steps been taken to address the potential for misloading for a reactor of the specific type in question, in particular if this is based only on the assembly identifier? Have redundant steps for assembly identification been identified? Have reactor records established that misloading is credible?

242. Generally, the operational limits and conditions for ensuring sub-criticality in spent fuel storage using burn-up credit are based on a conservative combination of the fuel's initial enrichment and the burn-up history (in which the amount of burn-up is an important parameter). This approach is commonly known as the 'safe loading curve' approach [Reference 40]. "In such circumstances, the criticality safety assessment should determine the operational measures necessary to ensure compliance with this curve during operation; for example, the measurements that are necessary to verify the initial enrichment and burn-up" [Reference 41].

243. Specific guidance on criticality safety using a burn-up credit approach is commonly provided for the handling and storage of spent fuel at nuclear power plants [Reference 42] independently from spent fuel storage facilities [Reference 43].

4.2.1   At Reactor

244. The most common use of burn-up credit at reactor is in spent fuel pools. This is because BUC offers advantages of high density storage compared with the older practice of assuming the fuel has not been irradiated. In this case, access to reactor records is easily achievable as all fuel is from the single reactor operator who maintains responsibility for the spent fuel pools. Measurement methods in this application must be compatible with fuel reloading operations and have a minor impact on the overall refuelling outage time. More than eleven countries have approved and implemented burn-up credit at some level for PWR and VVER fuels. These countries include: Belgium, Brazil, Germany, Korea, Netherlands, Russia, Slovenia, South Africa, Spain, Switzerland and the United States (US) [Reference 39].

245. The situation for dry storage at reactors is similar to that for pool storage. Taking credit for fissile depletion and some degree of neutron absorption in actinides and perhaps fission products results in increased cask capacities. Dry storage casks are loaded at reactors directly from the spent fuel pool after the fuel assemblies have cooled appreciably (usually about five years at minimum). Burn-up credit for dry storage is approved in the US and in use in Armenia, Germany and Ukraine [Reference 39]. Differences between pool storage and dry storage casks may be related to the increased cooling time and the increased possibility of misloading due to multiple fuel movements since removal from the reactor increasing reliance on configuration management for access to the correct reactor records. The cask design itself may have an impact on potential accident scenarios. This is particularly of concern where neutron absorbers may be axially distributed in the cask structure making the burn-up distribution in the fuel more significant and the need for verification by measurement more important.

246. The International Atomic Energy Agency (IAEA) has developed international guidelines for the use of burn-up credit for the storage of SNF [Reference 43]. The most significant consequence addressing this increase in risk was a requirement for the verification of burn-up by measurement in the initial release of IAEA SSG-15. The 2012 version of IAEA SSG-15 requires that "all fuel should be assumed to be at a burn-up and enrichment value that results in maximum nuclear reactivity, unless credit for burn-up is assumed on the basis of an adequate justification. Such justification should include an appropriate measurement or evaluation that directly or indirectly confirms the calculated values for the content of fissile material or depletion level. For application of burn-up credit in long term storage, possible changes in the nuclide composition of the spent fuel with storage time should be taken into account."

4.2.2   Away from Reactor

247. Similar to dry cask storage, transportation regulations usually require a minimum cooling time. These minimum cooling times can vary depending on the capacity and design of the transportation cask and if the cask is designed to operate dry or water-filled. These design differences could impact the possible accident scenarios. The uncertainty in fuel parameters (both burn-up and cooling time in this case) and/or the probability of a misload since fuel assemblies may have been moved multiple times increase the importance of measurement verification for these applications of BUC. One advantage may be that if the SNF operations to load the transportation cask are independent of the re-fuelling outage, an independent

measurement with accurate results might be more achievable. Burn-up credit for transportation has been approved in the US and is already in use in Armenia, France, Germany, Netherlands, Russia and Switzerland for VVER and PWR fuels [Reference 39].

248.   It is necessary to ensure that the methods and procedures that are in place to transfer the data between locations where the fuel is handled (e.g. between the reactor and a reprocessing plant) are robust and exclude the possibility of data file and transcription errors.

BUC47.   Has the licensee ensured that sufficient quality assurance procedures are in place to ensure the accurate transfer of operator data between plants/organizations involved in the management of spent fuel?

249.   The most significant distinction in the application of burn-up credit for at-reactor applications versus away-from-reactor applications is the potential to impact the public. Thus, the IAEA has developed guidance for its member countries related to the transportation of SNF. This IAEA guidance is given in IAEA Specific Safety Requirements SSR-6 [Reference 44]. Paragraph 677(b) requires "A conservative estimate of the neutron multiplication for the package assessments. After irradiation but prior to shipment, a measurement shall be performed to confirm the conservatism of the isotopic composition." Further advisory material is also contained in Reference 53.

250.   In the case of facilities reprocessing SNF, the need for verification by measurement remains a priority since the fuel processed at these facilities will be from a variety of plants and operators. The advantage of performing verification measurements at these facilities is that interference with reactor operations is no longer a concern. Both France and the UK use some form of burn-up credit in reprocessing facilities [Reference 39].

251.   At different stages of reprocessing, destructive analysis is also a possibility allowing the direct measurement of isotopes of neodymium to establish burn-up and uncertainty.

252.   Future disposal facilities will likely be established on a national or international basis and will therefore receive fuel from a range of reactor operators potentially including different reactor and fuel types, limited access to original reactor operating records and wide variability in the uncertainties in reported burn-up values. Belgium, the Czech Republic, Germany and Korea have incorporated burn-up credit in their analysis of disposal facilities. The US and Sweden have also pursued burn-up credit for disposal [Reference 39].

4.2.3   Special Considerations for Fuel Type

253.   Pressurised Water Reactor (PWR), PWR-Mixed Oxide (MOX) and Water-Water Energetic Reactor (VVER[2]) systems are fundamentally similar in geometry and operating parameters. The primary difference with PWR-MOX fuel is the relative importance of additional actinide nuclides included which may impact the requirements for independent measurement of burn-up. The at-reactor spent fuel pools for these fuels generally include soluble boron for criticality control and some have even integrated neutron absorber plates into the structure of the storage racks themselves. Any verification measurement system involving neutron detection needs to consider the potential impact of boron as a neutron absorber.

254.   In the case of Boiling Water Reactor (BWR) fuel, there is increased complexity to account for spatial variation in the moderator density (void fraction) in addition to fuel and moderator temperatures and the variation of initial enrichment throughout a single fuel assembly. In the case of enrichment variations, making conservative assumptions such as using the highest enrichments throughout and neglecting to credit physical neutron absorbers in the fuel may be

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[2] Vodo-Vodyanoi Energetichesky Reaktor (transliteration from Cyrillic), literally 'Water-Water Power Reactor'

options. While these simplifications may be appropriate for analysis, they will continue to be concerns contributing to uncertainties in the measurement of fuel assembly burn-up. Benchmarking measurements of burn-up in BWR fuel and establishing the overall uncertainty in reactor records will need to account for these additional sources of uncertainty.

BUC48. Are the measurement techniques used to verify the burn-up appropriate for the different types of fuel/type of operation covered by the assessment?

4.3 Approaches to Burn-up Verification by Application

4.3.1 At Reactor Applications (Pool and Dry Storage)

255. Both the United States and Germany have actively been developing guidance for the use of burn-up credit for at reactor storage of SNF.

256. The use of burn-up credit in spent fuel pools is well established in the US and has been in practice for decades [Reference 32]. In the US, criticality monitoring is required by 10 CFR 70.24 [Reference 45]. Demonstrating subcriticality by analysis was formalised as an alternative methodology in reactor regulations in 1998 [Reference 46]. An NRC internal memorandum often referred to as the "Kopp Memorandum" [Reference 8] provided the first guidance on acceptable methods for performing criticality analyses at LWR plants. Because of the lack of alternative formal guidance, the Kopp memorandum was quickly adopted by industry for use in performing criticality analyses and provided regulatory clarity and stability for many years. In 2010, the NRC issued an Action Plan to develop new interim staff review guidance followed by a durable Regulatory Guide that would replace the Kopp Memorandum, and would better reflect the staff positions on acceptable criticality analysis methods that evolved through interactions with licensees since 2005.

257. This NRC Interim Staff Guidance (ISG) DSS-ISG-2010-01, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," [Reference 47] was issued in 2011. Development began on more permanent and durable guidance in the form of NEI 12-16 [Reference 32] which when endorsed by the NRC, will supersede all previous guidance documents. The latter document requires consideration of the burn-up measurement uncertainty in the criticality safety assessment and defines this parameter as equivalent to the reactor record burn-up uncertainty. No specific regulatory requirement for burn-up verification by measurement has been established for spent fuel pool applications of BUC. However, it is required that "If burn-up is credited, a process should be implemented to ensure that the fuel was depleted in a manner consistent with the assumptions in the criticality analysis" [Reference 32].

258. Dry storage at-reactor is regulated on the basis of licensing the type of storage cask to be used. Interim guidance relative the use of BUC is given in ISG-8 [Reference 3]. Additional guidance is provided in NUREG-1536 [Reference 48].

259. German practice established a regulatory document for pool storage [Reference 49] and for dry storage & transportation [Reference 50]. In both cases, the misloading event must be excluded as a design basis event through application of the double contingency principle [Reference 39]. For pool storage, quantification and verification of burn-up is based on evaluation of the reactor records [Reference 51] and hence there is no requirement for a verification measurement. Reference 50 requires a burn-up measurement, "since stipulated in paragraph 674 of IAEA TS-R-1" [Reference 39], i.e. IAEA Regulations for the Safe Transport of Radioactive Material [Reference 52]. Hence for dry storage a consistency check by means

of measurement is also required, for example gamma scanning, measurement of passive neutron emission or both in combination [Reference 51].

4.3.2   Away From Reactor applications (transportation, reprocessing, disposal)

4.3.2.1   Transportation

260.   IAEA Regulations for the Safe Transport of Radioactive Material [Reference 52 and Reference 44] stipulate that for irradiated nuclear fuel, "after irradiation but prior to shipment, a measurement shall be performed to confirm the conservatism of the isotope composition". It is further stated in associated Advisory Material [Reference 53] that when burn-up credit is used in package assessment, "operational and administrative controls are needed to ensure that the INF[3] being loaded in the package is within the characteristics used to perform the safety evaluation" and that it is appropriate to link the required measurement to this assessment. The measurement technique should be appropriate to the intended purpose, for example:

   i.   In France, a simple gamma detector measurement is used for burn-up credit allowances less than a certain threshold, with more direct measurement being used for a higher irradiation allowance (as discussed further below); and

   ii.   If the axial burn-up profile is identified as being important in the safety analysis, measurements could be used to determine that the profile is within the defined limits.

261.   In the US, guidance for the use of BUC for the transportation of SNF is based on NRC interim staff guidance. This guidance has been developed and modified over the past two decades and is currently on revision 3. "ISG-8, Rev. 3, includes two major changes in the recommendations to staff reviewing burn-up credit applications for SNF transportation and storage systems: (1) optional credit for fission product and minor actinide neutron absorbing isotopes in the SNF composition, and (2) misload analyses and additional administrative procedures in lieu of a burn-up measurement at the time of loading" [Reference 3]. This change provides a regulatory option for burn-up verification measurements rather than requiring the license applicant to pursue exemptions from regulations. Further guidance for the use of burn-up credit for the transportation of SNF is given the NRC Standard Review Plan for Transportation Packages for Spent Nuclear Fuel, NUREG-1617.

262.   As discussed above, German practice established a regulatory document for dry storage & transportation [Reference 50]. For the case of transportation, covered by Reference 50, the misloading event must be excluded as a design basis event through application of the double contingency principle [Reference 39]. A burn-up measurement is additionally required, "since stipulated in paragraph 674 of IAEA TS-R-1" [Reference 39], i.e. IAEA Regulations for the Safe Transport of Radioactive Material [Reference 52]. Hence for transportation a consistency check by means of measurement is also required, for example gamma scanning, measurement of passive neutron emission or both in combination [Reference 51].

263.   A system known as PYTHON which uses a single neutron measurement and a total gamma measurement has been installed in the spent fuel storage ponds of various European reactors [Reference 38]. The system is used to provide a profile of burn-up along the axis of an irradiated fuel assembly and supports the BUC safety cases for the transport of spent fuel to COGEMA in France and its subsequent storage in the ponds at La Hague. However, the validation requirements themselves depend upon the degree of burn-up that is claimed in the safety case as follows:

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[3] Irradiated Nuclear Fuel

    i.   If the burn-up requirement is ≤3.2 GWd/t a simple gross gamma measurement is sufficient (or an equivalent method approved by the regulators may be used to confirm the burn-up); and

    ii.   If the burn-up requirement is >3.2 GWd/t a more detailed burn-up measurement must be performed, one which can demonstrate that the burn-up in the 50cm (axial) length of the assembly that is least irradiated is greater than the safety case requirement. The measurement method must be approved by the regulators.

BUC49. Are the number and type of misloading scenarios considered in the misloading analysis sufficient and appropriate? Have the scenarios been assessed correctly?

BUC50. Are the safety measures that are proposed to control fuel movements adequate and sufficient? Have the initiating event frequencies of misloading faults been justified adequately and are they acceptably low?

4.3.2.2   Reprocessing

4.3.2.2.1 France

264.   At the COGEMA reprocessing plant at La Hague fuel rods are sheared and the fuel pieces are dropped into the buckets of the rotary dissolver and lowered into hot (un-poisoned) nitric acid. The BUC safety assessment for the dissolver leads to a set of loading curves (one for each value of the initial enrichment) which specify a maximum permissible mass per dissolver bucket as a function of fuel burn-up [Reference 39]. Each curve gives a burn-up limit above which criticality safety is ensured by the geometry of the bucket (without any need to limit the fuel mass in the bucket). A PWR fuel assembly is normally spread over three or four buckets.

265.   Since the 1990s the criticality safety assessments for spent fuel storage in ponds and fuel shearing and dissolution at the COGEMA plant have been based on actinide-only BUC [Reference 38]. In addition, actinide-only BUC was used in the safety cases for the transport of irradiated fuels from Germany, Holland, Switzerland and Belgium to La Hague. Subsequent development (since the year 2000) has included the effect of fission products in the BUC approach used at La Hague. Since the introduction of BUC COGEMA has taken the view that out-of-reactor measurements to verify the fuel burn-ups are necessary. They have required reactor operators to carry out verification measurements at their spent fuel storage ponds before consigning the spent fuel to La Hague and have themselves carried out measurements to verify fuel burn-up prior to consigning spent fuel assemblies to reprocessing.

266.   Compliance with the safe loading curves for the dissolver buckets and the determination of the number of buckets required for a fuel assembly is obtained from a combination of checks against reactor data and the measurements made on the assembly. The measurements consist of gamma scanning (for peaks from ^{137}Cs, ^{134}Cs and ^{154}Eu) on two sides of the assembly and passive neutron count measurements on the other two sides [Reference 39 and Reference 38]. The CESAR depletion code [Reference 54] is used to interpret the measurements.

267.   To take into account the presence of MOX fuels in the pool, a blind identification of the fuel type (UO_2 or MOX) is first performed. Radiation transport calculations are utilised to determine k-effective as a function of initial enrichment, burn-up and conditions. The count rate is expressed as a function of detector yield, burn-up and k-effective. Given the count rate, the average burn-up is therefore determined to within 5% [Reference 55] . Some of the operator data used in the interpretation of the neutron measurement (e.g. loading history, fuel type) is then checked using the gamma scanning results.

268.   Based on a comparison between the measured average burn-up for the assembly and the value given for that quantity in the reactor data the instrument makes a "Go" / "No Go" decision as to whether or not the fuel assembly is fed forward for shearing into the dissolver bucket.

269.   Both measurement systems (i.e. PYTHON and the instruments installed in the reprocessing plants) have been approved by the French authorities, initially for UO_2 fuel assemblies. Subsequently, a validated methodology has been developed for MOX fuel assemblies [Reference 55].

4.3.2.2.2 UK

270.   The original criticality safety assessment for the THORP Head End plant, which did not utilise BUC, led to a requirement to include a certain concentration of dissolved gadolinium nitrate neutron poison in the dissolver acid, in order to achieve compliance with the criticality safety criteria. The presence of these gadolinium concentrations resulted in substantial active waste volumes. It was recognised that a reduction in these active waste volumes would provide reductions in the overall risks from storage and handling, as well as reductions in operator dose and in the cost of operations. Actinide only BUC was adopted to obtain a less conservative criticality assessment to reduce the required gadolinium concentration [Reference 56].

271.   In the Head End area of the THORP plant, measurements of the cooling time, Residual Enrichment (RE), burn-up and initial enrichment in the spent fuel of each fuel assembly are performed prior to shearing. Since RE is closely related to spent fuel reactivity this provides a practical way of verifying compliance with the safe limits derived in the criticality safety assessment. The measurements are made by the THORP FPFMs.

272.   The primary role of the FPFMs [Reference 37] is to ensure that all fuel entering the Head End of Thorp for reprocessing conforms to the plant flow sheet acceptance criteria without reference to operator-declared fuel data (other than fuel type) for the particular assembly under investigation. For initial THORP base load fuels these acceptance criteria were:

   i.   For LWR fuel: Irradiation ≤ 40 GWd/t(U); Cooling Time ≥ 5 years;

   ii.   For AGR fuel: Irradiation ≤ 25 GWd/t(U); Cooling Time ≥ 3.8 years; and

   iii. For both LWR and AGR fuels, pre-irradiation enrichment ≤ 4w/o ^{235}U.

273.   In order to achieve the planned plant throughput two FPFMs were installed. Each has a high purity germanium gamma detector and five fission chamber neutron detectors, split into two modules arranged at 90 degrees to each other. A neutron source transfer system, controlled by the FPFM moves a ^{252}Cf source between an exposed and shielded position to allow both active and passive neutron measurements. The exposed source is also used during pre-measurement standardisation to check the counting efficiency of the neutron detectors.

274.   The position of the detector and source modules are adjustable to accommodate a range of fuel sizes.

275.   The gamma detector is standardised using a mixed ^{137}Cs and ^{134}Cs source that can be moved between a shielded and exposed positions.

276.   While a fuel item is being removed from its storage container the monitor to be used is standardised. Measurements are then made at each of up to four measurement heights, during which the fuel is rotated in the measurement station. Local values of cooling time, irradiation and RE are calculated at each measurement height. In cases in which the fuel sentencing does not take account of BUC, initial enrichment is also calculated.

277. Weighted averages for the whole assembly are also calculated for each parameter. The weighted average values are used to generate a "Go" or "No Go" signal. If the outcome is "Go" the assembly is delivered to the shear elevator for transfer to the shear cave where it is sheared and the fuel pieces are then delivered to the dissolver. If the outcome is "No Go" the assembly is returned to its storage container for further investigation.

278. A combination of three radiometric techniques are used to measure the required parameters:

    i.   High Resolution Gamma Spectroscopy (HRGS);

    ii. measurement of passive neutron emissions; and

    iii. measurement of active neutron emissions from fissions induced by the ^{252}Cf neutron source.

279. In the HRGS the net photo-peak areas of the selected radionuclides (^{137}Cs, ^{134}Cs, ^{106}Ru and ^{154}Eu) are corrected for background activity and normalised for relative detection efficiency using the multiple photo-peaks of ^{134}Cs.

280. The required parameters are determined as follows:

    i.   Cooling Time: Two independent values of cooling time are determined by HRGS using the activity ratios ^{134}Cs/^{154}Eu and ^{106}Ru/^{137}Cs. The final cooling time for a particular measurement position is then obtained from a weighted mean of the two values;

    ii. Irradiation: Three separate values of irradiation are determined at each measurement position: two use HRGS and the third uses passive neutron measurements. In the HRGS determinations the absolute ^{137}Cs count rate and the activity ratio (^{106}Ru $\times$ ^{137}Cs/(^{134}Cs)^2) are used. The irradiation has a linear relationship with ^{137}Cs activity and a logarithmic relationship with the activity ratio. The third method for irradiation determination makes use of the relationship between passive neutron emission (principally from ^{244}Cm) and irradiation. The final irradiation value is calculated by taking a weighted mean of the three values;

    iii. Residual Enrichment: The final enrichment, expressed as a ^{235}U 'equivalent enrichment' is determined from a combination of measured irradiation and a neutron multiplication parameter from the active neutron measurements using the ^{252}Cf neutron source; and

    iv. Initial Enrichment: The initial enrichment is calculated from a combination of the final enrichment and measured irradiation via FISPIN [Reference 21] calibration data.

281. Typical measurement uncertainties (from early BWR campaigns) were as follows:

    i.   Cooling time: standard deviation 83 days (for values ranging from 1000 – 8000 days);

    ii. Irradiation: standard deviations 1 and 0.8 GWd/t for east and west monitoring stations, respectively for BWR calibration data with values ranging from about 7 to 35 GWd/t, standard deviation 0.48 GWd/t for the first THORP BWR campaign; and

    iii. Initial enrichment: standard deviation 0.12% ^{235}U.

4.3.2.2.3 Other activities

282. A method of verifying burn-up in reprocessing applications using dissolver off-gas has been evaluated in Japan [Reference 57] with potential application to burn-up credit applications. Xenon isotopic ratios can be evaluated using gas chromatography/mass spectroscopy, and these are correlated with fuel type and burn-up. In the study described in Reference 57, which

considered six batches of dissolver off-gas, declared and evaluated burn-ups were in agreement to within ~7% when using this method.

4.3.2.3   Proposals for Disposal Facilities

283.   In the US, regulatory agencies are mandating a measurement system that verifies fuel burn-up and/or characterises the composition of spent nuclear fuel. It is evident that in many cases reactor records are sufficiently reliable to eliminate the need for at-reactor burn-up measurements. The case for away-from-reactor applications is less clear. Access to reactor records may be limited if even obtainable for fuel assemblies that have long since been removed from the reactor. Also, fuel assemblies at a facility for permanent disposition such as a geologic repository will be from more than a single reactor/operator making the assessment of operating history and uncertainties more complicated.

BUC51.   In cases where records from multiple reactors are being used, is the level of reliance being placed on these records justified? Are uncertainties identified and accounted for?

284.   Under the auspices of the US geologic repository program, the Multi-Detector Analysis System (MDAS) is being developed to provide some understanding of the fissile mass, radiation source term, and fissile isotopic content information from recorded energy spectra, without reliance on initial fuel composition or historic plant operating records. It can verify burn-up calculations or be used directly for DOE fuel and waste characterization [Reference 36].

285.   It is intended to apply burn-up credit to the Swedish geologic repository [References 58 and 59]. Reference 59 states that "IAEA transport regulations… require measurements before transport to verify the fuel composition. This would be covered during the encapsulation operations." The quality of the reactor record is highlighted as being important.

286.   As discussed above, a measurement prior to transportation to the repository is consistent with IAEA regulations. If a measurement prior to transport is then used to support use of BUC at the repository, then this could place additional requirements on ensuring that packaged material is accurately tracked.

287.   Burn-up verification through measurement could in theory be complicated if the spent nuclear fuel is in its final packaging prior to disposal and burn-up of the individual assemblies were not verified by measurement prior to packaging. This could require reliance on records and measurements obtained previously by independent organizations. Reference 60 states that it may be preferable to perform verification measurements prior to future loading of assemblies to prevent having to unload assemblies to perform these measurements at a later date.

4.3.3   Challenges for Verification Measurements

288.   This section addresses approaches to providing operational measures to ensure compliance with the "loading curve". As discussed in Reference 34 the burn-up of a fuel assembly that is reported in reactor records is not (and cannot be) the result of a direct measurement of that quantity. Rather the burn-up is inferred from quantities which can be measured using numerical models of the reactor performance. It is generally necessary to visually confirm the identity of the assembly when using assembly-specific information from the reactor record.

289.   Local measurements within the reactor (e.g. of neutron flux) will only be available for a subset of the fuel channels. Other quantities used in the inference of fuel burn-up, such as the thermal output of the reactor, are averages over the whole reactor. However, because of the importance of a knowledge of fuel burn-up in the overall management of the reactor (from the perspective of both operational efficiency, safe operation and regulatory compliance), in

practice the means of characterising reactor conditions are well developed, as can be seen, for example, from Figure 4.3 of Reference 34 which shows very good agreement between the predicted and measured axial power distribution or burn-up, respectively, at an instrumented core location in a PWR. As discussed in Reference 34, various US studies have indicated uncertainties in burn-up within 5% and in many cases rather less.

290. One issue that requires consideration in any BUC safety case is the axial variation of burn-up along fuel assemblies, the fact that, as discussed in Reference 34 and Reference 61, the end portions of the fuel assemblies experience less burn-up, and so remain more reactive than the central parts of the assemblies.

BUC52. Has it been demonstrated that the issue of axial burn-up variation has been adequately addressed in the interpretation of fuel data (both reactor derived data and those from out-of-reactor verification measurements)?

4.4 Calibration of Systems for the Measurement of Burn-up Parameters

291. Systems used to determine burn-up are generally calibrated by measuring burn-up indicators from a representative sample of fuel assemblies with well-defined irradiation histories. This approach has the advantage that the calibration assemblies have the same geometry as the fuel to be measured (and as discussed above, there may be other factors that need to be consistent between calibration and verification measurement, e.g. position of the detector relative to the assembly in the case of passive neutron counting). In addition, other fuel parameters such as cooling time can be determined independently to provide validation of the operator-declared parameters for the reference assemblies.

292. This approach to calibration using a representative sample population is the approach that has been used, for example, in THORP [Reference 37]. Calibration measurements with sample populations of the different types of fuel to be processed (PWR, AGR etc.) are carried out in order to identify appropriate calibration parameters for each fuel type. The accuracy of the calibration is dependent on the accuracy with which the irradiation history of the sampled fuel assemblies are known. These are typically based on reactor records, themselves derived from in-core measurements with corresponding uncertainties. This factor tends to lead to burn-up verification measurements being less accurate overall than burn-ups derived directly from reactor records [Reference 34].

293. Another possibility is to seek methods of calibration that are independent of operator declared data, for example by determining the correlation between burn-up indicators and burn-up by using fuel inventory codes such as ORIGEN [Reference 62] and FISPIN [Reference 21]. These codes have been validated by comparison with data from destructive assay and provide calculated inventories for a wide range of fission products and actinides.

294. In order to use such an approach it is necessary to identify burn-up indicators that can be reliably calculated by the inventory code and which can be measured reliably. As indicated in Section 4.5.2, there are several gamma activity ratios that are insensitive to the system geometry and which could be used. Conversely however, these ratios often depend on relatively short-lived radionuclides and so cannot be used for long cooled fuels. They may also depend on the initial enrichment.

295. In principle, it is also possible to envisage correlating activity with criticality directly, through calibrating measured activity with an activity/criticality relationship derived computationally.

The relationship between activity and criticality is itself a function of burn-up. In practice, activity is linked to burn-up in the first instance, with burn-up then used to derive criticality.

296. Some additional confidence in/validation of an instrument used for burn-up measurements can be obtained if it can be shown that the two approaches to calibration indicated above (i.e. based on operator-supplied data and based on the use of inventory codes) lead to consistent results.

BUC53. Is the approach to calibration of the measurements adequate and appropriate for the type of fuel/type of operation covered by the assessment?

4.5 Background: BU Verification Measurement Techniques

297. The main out-of-core measurement techniques that are used are as follows [References 34 and 63]:

    i.   Measurement of the gamma activity of a single radionuclide;

    ii.   Evaluation of the ratio of the gamma activities of two or more radionuclides;

    iii.   Measurement of passive neutron emissions;

    iv.   Measurement of active neutron emissions (neutron multiplication) produced by a neutron source; and

    v.   Measurement of Cerenkov in fuel pond water.

298. These techniques and their advantages and disadvantages are discussed in more detail in the remainder of this section. In practice, different methods are often used together (e.g. gamma scanning and passive neutron measurements) in order to enable several parameters (e.g. cooling time and irradiation) to be obtained together from the measurements. The issue of the calibration carried out to relate the raw measurements to the fuel parameters of interest is also discussed.

299. There exists practical operational experience of the use of these methods for spent fuel monitoring in both wet and dry conditions, as appropriate.

300. Examinations of irradiated assemblies have been performed in several cases in order to compare reactor records with out-of-core measurement data and hence verify the reactor records and evaluate the effectiveness of the verification techniques [Reference 34].

301. Systems have been deployed to verify cooling time, burn-up and enrichment of fuel assemblies in dry fuel handling facilities, fuel storage ponds and reprocessing plants. In some cases, such as the FORK system, there is no direct connection between the measurement instrument and the plant control system. When FORK was used, for example, for in-pond measurements to verify fuel burn-up prior to cask loading at the Arkansas Nuclear One facility in the US [Reference 8] the measurement had no direct influence on the ability of a plant operator to move fuel assemblies within the pond or to a transport cask. Any restrictions on fuel movements relied on managerial controls. However instruments used within reprocessing plants, such as the FPFM, form part of the plant control system. The values measured by the FPFM are used to generate a "Go" or "No Go" signal depending on whether or not they fall within the safe domain defined in the plant safety case. In the case of the FPFM if the outcome is "Go" the assembly is delivered to the shear elevator for transfer to the shear cave where it is sheared and the fuel pieces are then delivered to the dissolver. If the outcome is "No Go" the assembly is returned to its storage container for further investigation. A similar approach is applied at the La Hague reprocessing plant.

302.   It is noted that many burn-up verification systems were developed for safeguards applications rather than for BUC, and hence are not optimised for this application [Reference 34].

4.5.1   Measurement of the Gamma Activity of a Single Radionuclide

303.   Measurements of the gamma activity of a single radionuclide are used to give a direct measure of fuel burn-up. Ideally, a single radionuclide whose activity is used to provide a measure of burn-up would have a high fission yield, a relatively long half-life and a strong gamma peak well separated from those of other fission products. These requirements are well met by ^{137}Cs which has a 30 year half-life (making it relatively insensitive to variations in reactor power rating and errors in cooling time corrections), very low neutron absorption and very similar yield from ^{235}U and ^{239}Pu fission [Reference 34]. Its activity thus has a linear relationship to burn-up. In addition, the 662 keV gamma peak of ^{137}Cs is the main gamma emitter measured after five years cooling (because of the short half-life of primary gamma emitters).

304.   These advantages have led to measurements of the ^{137}Cs gamma peak being used as a means of verifying fuel burn-up in BUC safety cases in reprocessing plants in both the UK and France. The addition of the ability to measure the ^{137}Cs gamma peak is what distinguishes the FORK+ detector from the original FORK instrument. A further advantage of measuring ^{137}Cs activity is that the relatively long half-life of ^{137}Cs means that (unlike the gamma activity ratios discussed in the next section) its measurement can still be used to verify the burn-up of older legacy fuels with long cooling times. Furthermore, because gamma rays can be collimated, gamma counting can be used to measure axial burn-up profiles [Reference 39].

305.   The main disadvantages of measuring the ^{137}Cs activity follow directly from the fact that the data are absolute measurements and so are very sensitive to factors such as the source geometry, the distance from the source to the detector, detector sensitivity and contributions from background radiation sources ("background noise").

306.   In addition, because of self-absorption of gamma-rays in the assembly, the measurements tend to be most sensitive to the outer pins in the assembly rather than the assembly as a whole. However, there are several ways in which this effect can be mitigated. In the THORP FPFM, for example, the fuel assembly is rotated as it is measured so giving results more representative of the whole assembly. At La Hague gamma measurements are made on two sides of an assembly.

307.   Averaging results from the two opposing sides of the assembly then gives a result which is not very sensitive to horizontal burn-up gradients in the assembly. In addition at La Hague, passive neutron measurements are made on the other two sides of the assembly. Using the FORK+ detector measurements can, in principle, be made on all sides of the assembly, although this will increase the measurement time and the amount of in-pond handling that will be required, which may have implications for operator dose [Reference 34].

308.   The absolute ^{137}Cs activity is a very useful burn-up indicator for fuel with a broad range of enrichment, burn-up and cooling times in practice. The activity of this radionuclide in spent fuel has been shown to be predicted consistently by different inventory codes and validated by destructive analysis. If the measurement geometry and detection efficiency are well known and reproducible ^{137}Cs can be used to provide a calibration that does not rely on the operator-supplied irradiation history. It is important to ensure that the measurement conditions are consistent with those assumed for the calibration and a measurement process that uses this approach therefore needs to include suitable checks to eliminate the possibility of systematic errors in the measurement conditions (e.g. in geometry or detection efficiency).

4.5.2 Measurement of the Ratio of the Gamma Activities of Two or More Radionuclides

309. Measurements of activity ratios [Reference 34 and Reference 39] are used to provide values of fuel cooling time and fuel burn-up. Ideally, the chosen radionuclides should be in a state of equilibrium when the spent fuel is released from the reactor, the two radionuclides in the ratio should have significantly different half-lives and their gamma emission peak energies must be measurable and not interfered with by other peaks in the fuel gamma emission spectrum.

310. Ratio methods have the advantage that they are insensitive to the source/detector geometry and (in some cases) to fuel enrichment and rating. However, there may still be a requirement to correct for the relative detection efficiency at the different gamma energies. Their main disadvantage follows from the requirement for the radionuclides in the ratio to have significantly different half-lives. This results in the use of radionuclides with quite short half-lives (e.g. ^{106}Ru, with a half-life of 373 days, ^{134}Cs with a half-life of 2.06 years). As, in practice, measurements of the gamma emissions from these radionuclides become impractical after a few half-lives, this limits the use of ratio methods to fuels with cooling times of less than ten to twenty years (depending on the particular ratio that is used).

311. Examples of radionuclide activity ratios that are used in practice [References 34 and 39] are:

    i. ^{134}Cs/^{154}Eu and ^{106}Ru/^{137}Cs which are used in the THORP FPFM to measure cooling time;

    ii. (^{106}Ru $\times$ ^{137}Cs/(^{134}Cs)$_2$) which is used in the THORP FPFM and in the Magnox Fuel Handling Plant (FHP) to measure burn-up; and

    iii. ^{140}La/^{95}Zr and ^{95}Zr/^{144}Ce which are used in the FHP to measure cooling time.

312. The ratios listed in bullet iii) involve radionuclides with relatively short half-lives. The half-life of ^{140}La is 1.68 days (although the relevant half-life in irradiated fuel is essentially the 12.7 day half-life of its ^{140}Ba parent); the half-life of ^{95}Zr is 64 days and that of ^{144}Ce is 285 days. These values imply that the ratios are only useful for cooling times less than about 160 days and 500 days, respectively. Thus, it is clear that these values are too short for the ratios to be relevant to BUC verification, rather they are used to confirm that the fuel cooling time is greater than the 150 day (radiological) limit applied to de-canning operations at FHP. However, they again illustrate the principles outlined above that apply to the choice of an appropriate activity ratio.

4.5.3 Measurement of Passive Neutron Emissions

313. Measurement of passive neutron emission, primarily from ^{244}Cm can also be used to determine the fuel irradiation (using the relationship between the passive neutron count rate and the fuel irradiation, [References 39 and 63]). This method has the following advantages. First, the count rate has a very strong dependency on burn-up, cooling time and initial enrichment so that if the enrichment, cooling time and the system geometry are well known it is a very sensitive and accurate indicator of burn-up (the ^{244}Cm content is proportional to the approximately the fourth power of the burn-up). Second, for an assembly in a fuel pond the initial neutrons cascade through the assembly as a result of induced fissions. This amplifies the flux and effectively samples all of the pins in the assembly so that, unlike gamma measurements, the measurement is representative of all of the pins in the fuel assembly, not just those in the outer layers. Third, the relatively long half-life of ^{244}Cm (18.1 years) means that the method is relatively insensitive to errors in cooling time corrections.

314. The disadvantages of the method are: First, that the ^{244}Cm activity is strongly dependent on the initial enrichment of the assembly, so that reliable information on the initial enrichment will be required to correctly interpret the results of the measurement. Second, as an absolute measurement of count rate the measurement is sensitive to the source geometry (including the geometry of any water gap between the source and the detectors) and the distance from the source to the detector. Third, that the measurement can be affected by neutron multiplication. Although this may increase the signal strength, it can also complicate the interpretation of the measurement. The signal can also be affected by the presence of any neutron poisons in the fuel or in the pool water. Accurate calibration of the measurements will therefore be very important.

4.5.4   Measurement of Active Neutron Emissions.

315. Unlike the other methods outlined in the previous section active neutron measurements using a neutron source are not used to measure the fuel burn-up but rather to measure the 'Residual Enrichment' (RE) of the fuel.

316. The RE is expressed as a ^{235}U 'equivalent enrichment' which includes the contribution from the Pu content of the irradiated fuel, which is related to an 'equivalent' amount of ^{235}U. For ^{239}Pu and ^{241}Pu factors based on the relative thermal fission cross-sections are used to determine their equivalent ^{235}U mass (equivalent in terms of reactivity). In some BUC safety cases, such as that for THORP the safe domain is defined in terms of RE values, rather than in terms of the burn-up itself.

317. RE, expressed as w/o ^{235}U equivalent is determined from a combination of the measured irradiation (obtained from the methods described in previous sections) and a neutron multiplication parameter [Reference 37]. The latter is determined by measuring the neutron flux produced when the fuel assembly is exposed to a known neutron source (usually ^{252}Cf) and comparing it to the passive neutron flux produced by the assembly in the absence of the source. Prior to the active measurement the source will also have been used to standardise the neutron detectors. The RE is linearly related to the neutron multiplication. The gradient of the relationship is irradiation dependent. Hence an alternative to deriving the RE may be to assess the reactivity of the SNF package based on measurements and verify this value with respect to the calculated reactivity.

318. A 'self-interrogation' technique has been considered where instead of utilising an external neutron source, the flux produced by the intrinsic source of the assembly itself and the fissions resulting from it is measured [Reference 39].

319. In the initial calibration exercise carried out prior to each fuel campaign values of RE derived from operator's data are used to relate measurements of neutron multiplication to RE.

320. A different active measurement method called Differential Die Away (DDA) is used by the THORP Hulls Monitor [Reference 39] in order to measure the mass of fissile material that remains in the pieces of empty fuel cladding that remain in the dissolver basket after fuel dissolution. DDA is used for a variety of fissile material measurement applications. In the hulls monitor it is used together with passive measurements of neutron and gamma emissions to give several measures of the fuel content.

321. Fast neutron pulses of a known energy, frequency and width are generated by a deuterium-tritium neutron generator into the measurement cavity containing the hulls. If there is no fissile material in the cavity the fast neutron flux in the cavity quickly dies away because the neutrons are thermalised, absorbed or escape. However, when fissile material is present in the cavity neutrons produced from induced fissions in the fissile material slow the rate of decay. The

degree to which the rate of decay is reduced allows an estimate of the fissile mass (generally expressed as grams ^{235}U equivalent) in the cavity to be obtained. When combined with passive neutron and gamma measurements, and information from the FPFM the total mass in the hulls and thus the RE of the material in the hulls can also be estimated.

322.   With these methods an axial section of the fuel several centimetres in length contributes to the signals and so the measurement yields an average burn-up over that region [Reference 39]. The method is therefore unsuitable for the measurement of axial burn-up profiles which, as indicated in Section 4.5.1, can be obtained using collimated gamma measurements.

323.   Active neutron counting could in principle be used to measure k-effective (and hence sub-criticality) of the system directly, rather than burn-up (which is then used to calculate sub-criticality). The directly measured k-effective would only be correct for the configuration in which it was measured. Reference 39 indicates that there has been 'limited development' in this area.

4.5.5   Measurement of Cerenkov Radiation

324.   The movement of a charged particle through a medium at a velocity greater than the phase velocity of light in that medium leads to the production of Cerenkov radiation. Irradiated fuel in a storage pond leads to the production of electrons (either directly from beta decay or indirectly from electrons produced by Compton scattering of gamma-rays) with velocities above the phase velocity of light in water and so to the generation of Cerenkov radiation with wavelengths ranging from ultraviolet to infrared. The intensity of the Cerenkov radiation generated by the irradiated fuel increases with fuel burn-up and reduces with increased fuel cooling time [Reference 34].

325.   Therefore, in principle, viewing the intensity of the Cerenkov radiation generated by spent fuel in a storage pond gives a relatively straightforward and nonintrusive method for verifying fuel burn-up, with no requirement to move the fuel in order to make the measurement. Detectors used for measuring Cerenkov radiation are generally based on measuring the ultraviolet emissions from the fuel and filter out light at normal optical frequencies so that the measurement is unaffected by the facility lights which can be left in their normal state when the measurements are being made.

326.   Cerenkov radiation is produced within the assembly, in the gaps between the fuel pins and is highly collimated along the axis of the assembly. This is ideal for the normal vertical arrangement of fuel assemblies in a storage pond although it also implies that careful positioning of the detector above the assembly is required in order to detect the maximum intensity of the Cerenkov radiation produced by the assembly.

327.   As described in Reference 34, trials of instruments for the measurement of Cerenkov radiation in Japan, Sweden and Canada have shown that the Cerenkov radiation from spent fuel can be measured and that it may be possible to determine the cooling time of an assembly if its burn-up is known (from operator records). However, the detectors used in the trials discussed in Reference 34 were developed in the context of nuclear safeguards work, rather than to support BUC arguments and so had not been optimised to measure burn-up. Thus, in the context of BUC assessments their use would be limited to screening out fuel assemblies that lay well outside the safe domain of a BUC safety case (e.g. un-irradiated or very lightly irradiated assemblies) and any safety case using the detection of Cerenkov radiation as a form of verification would therefore have to incorporate very large safety margins to allow for the relatively crude level of screening that it provides.

328.   In principle, the technique has significant advantages compared to those in Sections 4.5.1 – 4.5.4: The measurements can be performed in-situ in the storage pond with no requirement to handle the fuel assembly. Potential hazards associated with fuel handing (e.g. from dropping or otherwise damaging the fuel or from radiation hazards to personnel) therefore do not arise with this method. However, this has to be balanced against the relatively imprecise nature of the information obtained from the measurement.

329.   In addition, there are several factors that could influence the measurements that would require more precise quantification if the method was to be used as part of a BUC safety case. One of these is the effect of the fuel pond water on the signal. It is known, for example, that neutron absorption resulting from the presence of boron in the pond water can substantially reduce the intensity of the Cerenkov radiation, compared to a pond with no added boron. As well as taking account of the potential effects of neutron poisons, it would be necessary to quantify any effects from other chemicals that may be routinely added to pond water (e.g. to control the pH of the pond).

330.   In addition, it is necessary to consider the potential contribution to the measured Cerenkov radiation from assemblies adjacent to that being measured because gamma radiation from adjacent assemblies can travel into the one being measured and generate Cerenkov radiation there, potentially (for example) increasing the burn-up inferred for the measured assembly. It would be necessary to demonstrate that this effect had been quantified and taken into account in any proposal to use measurements of Cerenkov radiation in support of a BUC safety case.

331.   Finally, it should be noted that, because the intensity of the Cerenkov radiation is directly related to the intensity of the ionising radiation flux from the fuel assembly then based solely on the intensity of the Cerenkov radiation it will be very difficult to discriminate between long-cooled well-irradiated fuel and short-cooled lightly-irradiated fuel. As the REs of these fuels could be very different a BUC safety case will require the ability to tell the difference between these two cases and so it is unlikely that the verification required by such a case could be based on Cerenkov radiation measurements alone. Some other source of verification will be required. It is notable that no BUC safety case has made use of Cerenkov radiation measurements as a verification requirement.

4.6   Background: Specific Additional BU Verification Measurement Systems

4.6.1   FORK/FORK+

4.6.1.1   Background

332.   The FORK system [References 34 and 36], as originally developed, was designed to perform a Safeguards role and so was simply required to confirm that the fuel had been irradiated. It was intended that any more detailed information about the assembly would be obtained from the operator records. It was subsequently used (and its capability extended) to provide verification that the burn-ups of fuel assemblies that were held in fuel storage ponds or were to be loaded into casks for transport were compliant with the conditions specified in the criticality assessments of the storage ponds or transport casks which took credit for the fuel burn-up.

333.   However, it was considered by the developers of the instrument that the data provided by reactor operators would generally be very reliable (essentially for the reasons outlined

previously) and so the aim of the instrument was to detect unexpected random, relative variations among assembly burn-ups and to identify any anomalous values, rather than to provide a detailed characterisation of the properties of an individual assembly.

4.6.1.2 The FORK/FORK+ Detector

334. The original FORK system measured passive neutron emissions (via two fission chambers, one in in each arm of a fork [Reference 36]) and had an ion chamber positioned between the two fission chambers to measure the gross gamma-ray count rate.

335. One fission chamber in each arm is designed to detect epithermal neutrons. They are each enclosed in a polythene cylinder that is surrounded by a thin sheet of cadmium which absorbs thermal neutrons. The other fission chamber has no such cover and so is also sensitive to thermal neutrons.

336. In the FORK+ instrument a cadmium-zinc-telluride detector was added into the back of the device. The purpose of that detector is to measure the 662 keV gamma peak of ^{137}Cs and so to provide a direct measure of the fuel burn-up.

337. When the instrument is in use, the detector head is moved in the storage pond to the location of the fuel assembly to be measured. The assembly is raised out of its storage location until the point at which the measurement is to be made is adjacent to the detector head. The detector is moved into contact with the assembly and neutron and gamma counts are collected. Measurement of each assembly (at the single measurement point selected) takes about one minute.

338. Calibration of the detector is required in order to be able to interpret the measured neutron and gamma fluxes in terms of the fuel burn-up. In principle, this could be achieved using a reference fuel assembly of known burn-up B_r and cooling time t_r with the same geometry as the assemblies that will subsequently be measured. The measured count rate from the reference assembly C_r and its known burn-up and cooling time can be used to interpret the measured count rate C from a second unknown assembly in terms of its burn-up C and cooling time t using a relationship such as:

$$\left(\frac{C}{C_r}\right) = \left(\frac{B}{B_r}\right)^p \left(\frac{t}{t_r}\right)^{-m} \tag{6}$$

339. Where p and m are coefficients obtained from burn-up models for a particular isotope. When considering the neutron count rate it is necessary to take account of variations across the assembly population of cooling time and initial enrichment. The correction for cooling time is made by extrapolating the neutron data back to the date on which the assembly was discharged from the reactor, using the 18.1 year half-life of the main neutron emitter ^{244}Cm. A correction for initial enrichment is required because ^{244}Cm is produced by activation of ^{238}U and so its abundance depends on the reactor flux rather than the fission rate. The correction for variations in initial enrichment is based on the reported values of initial enrichment in the reactor records.

340. In practice, the approach that was adopted when the system was deployed was not actually based on the use of a particular reference assembly but rather on producing a 'self-calibration' from the data by using the measured values from many assemblies and their reactor records to generate a best fit to the observations. Comparison of the result for each assembly with the best fit curve enables variations in assembly burn-ups to be characterised and would reveal any individual assemblies that had an anomalous count rate or anomalous reactor data.

341. As discussed in Reference 34 and Reference 36 measurement campaigns using the FORK and FORK+ detectors (and a similar Westinghouse system) have been carried out at several different spent fuel storage ponds (initially to verify fuel burn-ups in connection with IAEA Safeguards inspections). Each campaign involved carrying out measurements on several tens of fuel assemblies (from 34 – 93 assemblies). Taken together over the several campaigns measurements have been carried out for assemblies with cooling times ranging from 0.18 – 17.6 years and burn-ups from 14.4 GWd/t to 58.3 GWd/t. It was found that the deviation of the FORK burn-up measurements from the best fit curve of reactor record burn-up against neutron counts was generally less than 5% and in most cases within 2-3% random variation. The maximum deviation seen for a standard fuel element was 9.1%. It is also notable that the presence of neutron sources in two assemblies was clearly identifiable from the FORK measurements, indicating the ability of the system to identify anomalous assemblies (though it should be noted that the neutron sources were located near the midpoint of the assemblies, i.e. at the location where the measurements were made on all of the assemblies).

342. The measurements made at different sites showed that the dependence of the neutron counts on burn-up varied with the assembly design. For the Babcock and Wilcox fuel assemblies measured at Arkansas Nuclear One Unit 1, the neutron count was proportional to the burn-up to the power of 3.83 whereas for Combustion Engineering fuel assemblies measured at Arkansas Nuclear One Unit 2, the neutron count was proportional to the burn-up to the power of 4.35. This factor would need to be taken into account in situations in which assemblies of different designs could be present (e.g. in the feed pond of a reprocessing plant).

343. The FORK systems have the advantage that the measurement time for an individual fuel assembly is short (about one minute) so that the impact of a measurement campaign on other operations in the storage pond is limited. The main operational disadvantage of the instrument is the requirement to lift the fuel assembly out of storage to make the measurement. Ideally this would be done when the assembly has to be moved anyway (e.g. for loading into a flask) in order to minimise the additional handling of the assembly and its associated risks (e.g. damage from dropping the fuel, impact on other items in the fuel pond, etc.)

344. The FORK system is typically used to take a single measurement at one axial position on the assembly (generally at its midpoint) and so no information on the axial variation of burn-up along the assembly is obtained. Although measurements from the mid-plane alone may be used to produce an estimate of the overall assembly-average burn-up (e.g. based on more detailed measurements of a reference assembly, specific modelling of the burn-up variation or information on axial burn-up variations such as that given in Reference 6) the lack of measured information on the axial variation will introduce additional uncertainty into the value of assembly-average burn-up. In principle, there is no reason why more FORK measurements at different axial positions could not be made. However doing so would increase the overall measurement time for a given assembly and therefore the time for which the fuel assembly is being handled. In any particular case it will be necessary to assess the balance between the value of additional measurements, the impact on facility operations of the extra measurement time that would be required and the potential risks associated with additional handling of the fuel assemblies.

345. The NRC guidance on the use of burn-up credit in criticality safety assessments of PWR casks [Reference 3] indicates that burn-up verification should include a measurement that confirms the reactor record for each assembly (although it also indicates that measurement of a sample of fuel assemblies would be considered if the sampling method can be justified). Although no specific measurement systems are mentioned in the guidance it appears that appropriate FORK measurements could satisfy the requirements in Reference 3, at least for LWR fuels, particularly if combined with information such as that in Reference 6.

4.6.2   The NAJA Device

346.   The NAJA device has been designed to determine automatically whether fuel is UO_2 or MOX; whether fresh or irradiated; the presence and type of any absorbers; the assembly enrichment; and the assembly identification number. Validation of fuel loading and burn-up credit are target applications [Reference 64]. NAJA is intended to be located between the pond and the reactor building, and to have no influence on the fuel loading and unloading schedule [Reference 65], and hence has the potential to mitigate some of the economics penalty associated with burn-up credit application.

347.   NAJA combines active neutron counting; passive neutron counting; gamma spectroscopy; a video for capturing the fuel identification number (including in the case where the fuel assembly is corroded); an ultrasonic probe for locating the assembly position relative to the detector; and an online depletion code. Active neutron measurement is used to determine whether the fuel type is as intended. Hence if the fuel type is already known, then the active neutron measurement is no longer necessary.

348.   A feasibility study of NAJA has been performed, with comparison to PYTHON and to experimental results. Accuracies of up to 2.5% in burn-up and 0.5% in UO_2 enrichment (for enrichments of 3-4%) are reported [Reference 39].

349.   It is claimed that the NAJA device improves safety through elimination of the misloading event (i.e. through determining/confirming assembly type and checking of the spent fuel. Operational benefits are also discussed in Reference 65.

5   Future Work

350.   Burn-up credit is still the subject of research and development activities around the world and it is therefore expected that improvements to methodologies and supporting data will be forthcoming. The following paragraphs indicate some examples of where ongoing work may have an impact on future burn-up credit critical safety assessments, but clearly this brief list should not be considered exhaustive.

351.   Developments to depletion and criticality codes are often driven by user requirements and the increasing application of burn-up credit arguments is therefore leading to demand for improved burn-up credit capabilities in the major depletion and criticality codes. For example, with particular relevance to the UK, the recent release of MONK10B was significantly motivated by improvements to the burn-up credit capabilities [Reference 66] and further developments are planned for future versions. It is therefore expected that future releases of depletion and criticality codes will include improvements to the accuracy, efficiency and practicality of burn-up credit calculations.

352.   Benchmark experimental data for irradiated fuels are still scarce, and there is a general move towards using commercial reactor critical data for validating reactivity calculations in irradiated nuclear fuel. Any such data, or results of computational benchmarks for irradiated fuels, which are published in future will contribute to the much-needed validation data for burn-up credit assessments.

353.   The major nuclear data projects, such as JEFF (Europe) and ENDF/B (US) continue to improve the accuracy and coverage of their nuclear data evaluations. Both JEFF-3.3 and ENDF/B-VIII.0 are expected to be released in 2017. In addition to improving data for nuclear cross-sections and fission product yields, future developments will also address covariance data. It is recognised internationally that the current covariance data are sparse and all the major nuclear data projects are planning to increase the quality and quantity of their covariance data in future releases, which will facilitate better uncertainty analysis in future burn-up credit calculations.

354.   Sensitivity, uncertainty, and data assimilation methods are all currently areas of research in criticality modelling. These approaches might be expected to lead to increasingly less conservative methods being used for burn-up credit calculations, and in particular to the use of best estimate plus uncertainty (BEPU) methods.

6   Conclusions

355.   It is difficult to quantify accurately the effect that BUC will have on criticality safety and it is also difficult to link the measured radiation from spent nuclear fuel with the degree of burn-up for compliance.

356.   These difficulties demonstrate that BUC is a complex technical area and indeed is one that is still undergoing significant development around the world. In order to support ONR inspectors receiving applications utilising BUC in the UK, this report has provided an overview of the methodologies and data involved in BUC currently. This report provides information covering:

> i.   regulatory practices around the world (largely concentrating on the US, as the major adopter of BUC);
>
> ii.   the methods which can be used to model burn-up in spent fuel for criticality safety assessment;
>
> iii.   the validation methods and data available for the codes and nuclear data used to calculate the models (for both depletion and criticality);
>
> iv.   the available technology for the calculation and confirmatory measurement of burn-up or reactivity, and the reliability of such technology; and
>
> v.   areas of potential difficulty, where appropriate.

357.   In order to highlight specific pieces of key information this report includes a series of "Regulator Questions". These questions are intended as triggers for an ONR inspector when reviewing a safety case to ensure areas of known concern are addressed. No attempt has been made to provide "an appropriate answer" to each of the fifty-three questions presented. Appropriate answers may be application specific and would require suitable justification by a licensee, coupled with subsequent consideration by the regulator to confirm that they satisfy all necessary requirements. Information has been included in this report to aid in the formation of such justifications and for regulatory considerations to be made.

Tables

Table 1 Nuclides considered in actinide-only burn-up credit

Nuclides		
^{234}U	^{235}U	^{238}U
^{238}Pu	^{239}Pu	^{240}Pu
^{241}Pu	^{242}Pu	^{241}Am

Table 2 Additional nuclides considered in actinide and fission product burn-up credit

Nuclides in addition to Table 1			
^{236}U	^{237}Np	^{243}Am	
^{95}Mo	^{99}Tc	^{101}Ru	^{103}Rh
^{109}Ag	^{133}Cs	^{143}Nd	^{145}Nd
^{147}Sm	^{149}Sm	^{150}Sm	^{151}Sm
^{152}Sm	^{151}Eu	^{153}Eu	^{155}Gd

It is noted that ^{236}U, ^{237}Np and ^{243}Am are not positive contributors to system reactivity so their exclusion from actinide-only burn-up credit is conservative. The main reason for their exclusion is that they are not significant contributors to the reactivity of the system and have limited validation data available.

Table 3 List of Regulator Questions.

No.	Has the Licensee Addressed this Question Satisfactorily
BUC1	What specific steps have been undertaken to validate the computer code system used to calculate isotopic concentrations in SNF created during burn-up in the reactor core and subsequent decay in this application?
BUC2	What specific steps have been undertaken to validate a computer code system to predict the subcritical multiplication factor, k-effective, of SNF in this particular application?
BUC3	Have suitable measures been applied to ensure the bounding conditions for the isotopic concentration and criticality calculations have been established?
BUC4	Have validated codes and bounding conditions been used to adequately generate application specific acceptance criteria (e.g. BUC loading curves)?
BUC5	What measures have been identified that allows verification that SNF assemblies meet the acceptable criteria and confirm that proper fuel assembly selection has occurred prior to use in the particular application?
BUC6	Has the licensee provided sufficient measurement data and/or justified any extrapolation techniques adopted such that the established bias and bias uncertainty information can be justified as reasonable and bounding?
BUC7	Has the licensee ensured that all isotopes considered in the BUC approach are non-volatile, non-gaseous, relatively stable and that the effect of the presence of the isotopes have been considered in the estimation of bias and bias uncertainty?
BUC8	Has the licensee considered the evolution of the reactivity of the application system over a suitable time period?
BUC9	Has the licensee taken into account the potential effects of axial variation in burn-up and ensured that sufficient evidence has been provided to justify that a suitably representative and bounding axial burn-up profile has been used?
BUC10	Has the licensee adequately considered the potential effects of horizontal variation in burn-up?
BUC11	Has the licensee adequately considered the potential for increased reactivity due to the presence of burnable absorbers or control rods during irradiation?
BUC12	Has the licensee adequately considered the potential effects of atypical control rod insertions during irradiation?
BUC13	Has the licensee ensured that the bounding axial burn-up profile incorporates assemblies exposed to control rod insertions?

Table 3 List of Regulator Questions. (continued)

No.	Has the Licensee Addressed this Question Satisfactorily
BUC14	Has the licensee adequately considered the potential effects of fuel temperature during irradiation and used a suitable bounding value in the depletion calculations?
BUC15	Has the licensee adequately considered the potential effects of moderator temperature and density during irradiation and used a suitable bounding value in the depletion calculations?
BUC16	Has the licensee adequately considered the potential effects of soluble boron concentration during irradiation, where appropriate, and used a suitable bounding value in the depletion calculations?
BUC17	Has the licensee adequately considered the potential effects of specific power and operating history during irradiation and used a suitable bounding approach in the depletion calculations?
BUC18	Has the licensee adequately considered the effect of computational methods on the bias and uncertainty associated with the computer code system used to calculate isotopic concentrations?
BUC19	Has the licensee adequately considered the effect of nuclear cross-section and decay data on the bias and uncertainty associated with the computer code system used to calculate isotopic concentrations?
BUC20	Has the licensee adequately considered the effect of input parameters on the bias and uncertainty associated with the computer code system used to calculate isotopic concentrations?
BUC21	Has the licensee adequately considered the effect of modelling on the bias and uncertainty associated with the computer code system used to calculate isotopic concentrations?
BUC22	Has the licensee adequately considered the effect of experimental data on the bias and uncertainty associated with the computer code system used to calculate isotopic concentrations?
BUC23	Has the licensee employed appropriate radiochemical assay (RCA) or post-irradiation examination (PIE) data to support depletion code validation?
BUC24	If the licensee has utilised proprietary data to support depletion code validation has the licensee supplied sufficient information to allow regulator verification of the claimed calculation accuracy?

Table 3 List of Regulator Questions. (continued)

No.	Has the Licensee Addressed this Question Satisfactorily
BUC25	Has the licensee defined suitable acceptable domain loading curves, demonstrating both the range of applicability and the anticipated population of fuel inventory?
BUC26	Has the licensee ensured that all appropriate sources of bias and uncertainty have been incorporated into the definition of the acceptable domain loading curves?
BUC27	Has the licensee considered the implications of misloading a single, severely-underburned (or even potentially fresh) assembly as well as multiple, moderately-underburned assemblies?
BUC28	Has the licensee defined robust administrative procedures to ensure that only acceptable SNF will be utilised in the application under consideration?
BUC29	Has the licensee demonstrated the use of appropriate and correct flux normalisations in each time-step of the depletion calculation?
BUC30	Has the licensee used sufficient spatial discretisation of materials in the horizontal and axial directions, and radially within rods containing burnable absorbers, to adequately represent the spatial variation in the burn-up in the depletion model?
BUC31	Has the licensee used sufficiently-small burn-up time-steps to ensure a numerically-stable and converged estimate of the isotopic composition of the irradiated materials?
BUC32	Has the licensee demonstrated that the reaction rates in any 3D Monte Carlo calculations have been adequately sampled to give sufficiently-accurate isotopic compositions?
BUC33	Has the licensee demonstrated that the spatial, angular and energy discretisation employed by a 2D flux solution give sufficiently accurate isotopic compositions?
BUC34	Has the licensee demonstrated that resonance self-shielding has been adequately modelled?
BUC35	Has the licensee demonstrated a rigorous approach to ensuring that the discretised material compositions are transferred to the correct geometric locations in the criticality calculations?

Table 3 List of Regulator Questions. (continued)

BUC36	Has the licensee demonstrated that spontaneous fission effects will not be significant, i.e. there are negligible quantities of nuclides such as 252Cf. Otherwise, does the application discuss the worth of spontaneous fission in the SNF?
BUC37	Is the licensee claiming nuclides other than those recommended in Reference 3? Does the licensee present validation for each of the nuclides being claimed?
BUC38	Is the licensee claiming BUC for MOX fuels? Are additional nuclides such as ^{242m}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm being claimed? Does the licensee present validation for these nuclides?
BUC39	Are there additional fissionable actinides in the system being evaluated which are not considered in the analysis, and which could result in underestimation of k-effective?
BUC40	Does the licensee provide details of the nuclear data set(s) used in supporting calculations? If the nuclear data are from a recognised source, are references to supporting documentation and validation work provided? If the nuclear data are taken from multiple sources, are secondary sources documented?
BUC41	Is evidence provided that the nuclear data set is validated for the reactor type under consideration? Is there evidence of validation for burn-up credit assessment? If not, what allowances are made for use of un-validated data?
BUC42	Does the licensee support their application with an estimate of uncertainty on reactivity worth (arising from nuclear data and other uncertainties)? If not, are conservatisms employed to allow for nuclear data uncertainties?
BUC43	Has the licensee sufficiently justified the level of uncertainty assumed in the reactor data?
BUC44	If a BUC safety case has been presented without out-of-core verification methods is there sufficient information available from reactor records/measurements to confirm the burn-up is at least the limiting value required for a given assembly type and enrichment to the required degree of confidence?
BUC45	Has the licensee identified the risks associated with burn-up verification (for example, but not limited to, increased operator dosage), and are they justified?

Table 3 List of Regulator Questions. (continued)

BUC46	Have sufficient steps been taken to address the potential for misloading for a reactor of the specific type in question, in particular if this is based only on the assembly identifier? Have redundant steps for assembly identification been identified? Have reactor records established that misloading is credible?
BUC47	Has the licensee ensured that sufficient quality assurance procedures are in place to ensure the accurate transfer of operator data between plants/organizations involved in the management of spent fuel?
BUC48	Are the measurement techniques used to verify the burn-up appropriate for the different types of fuel/type of operation covered by the assessment?
BUC49	Are the number and type of misloading scenarios considered in the misloading analysis sufficient and appropriate? Have the scenarios been assessed correctly?
BUC50	Are the safety measures that are proposed to control fuel movements adequate and sufficient? Have the initiating event frequencies of misloading faults been justified adequately and are they acceptably low?
BUC51	In cases where records from multiple reactors are being used, is the level of reliance being placed on these records justified? Are uncertainties identified and accounted for?
BUC52	Has it been demonstrated that the issue of axial burn-up variation has been adequately addressed in the interpretation of fuel data (both reactor derived data and those from out-of-reactor verification measurements)?
BUC53	Is the approach to calibration of the measurements adequate and appropriate for the type of fuel/type of operation covered by the assessment?

Table 4 Examples of depletion codes and code packages.

Code	Comments	Originators
MONK	General 3D Monte Carlo criticality and reactor physics code with integrated depletion solver and burn-up credit functionality.	ANSWERS (UK)
FISPIN	Point depletion code with reactor and burn-up dependent cross-sections supplied by WIMS/TRAIL.	ANSWERS (UK)
WIMS	General 2D/3D deterministic / Monte Carlo reactor physics code with integrated depletion solver, or coupled to FISPIN. Can be combined with PANTHER nodal, whole-core performance code.	ANSWERS (UK)
DARWIN	Coupled code package comprising PEPIN2 depletion solver utilising cross-section data calculated by APOLLO2 or TRIPOLI.	CEA (France)
CASMO	2D lattice code for PWR/BWR fuel with integrated depletion solver. Can be coupled to SIMULATE nodal performance code.	Studsvik (Sweden)
MCNP	General 3D Monte Carlo radiation transport with integrated depletion solver (in MCNP6).	LANL (US)
SCALE	Coupled code system including 2D lattice code (TRITON), 3D Monte Carlo code (KENO) and depletion solver (ORIGEN).	ORNL (US)
SERPENT	3D Monte Carlo reactor physics code with integrated depletion solver.	VTT (Finland)

Table 5 Status of the Benchmarks on Burn-up Credit performed by EGBUC.

Phase	Application	Description	Status
I-A	PWR UOX	Multiplication factor, spectra and reaction rates calculations for an infinite PWR fuel rod lattice with varying compositions: different enrichments; burn-ups; cooling-times; presence or absence of actinides (major and minor), and of fission products (major and minor).	Report published in June 1996
I-B	PWR UOX	Depletion calculations for a simple infinite PWR pin-cell lattice.	Report published in January 1994
II-A	PWR UOX	Multiplication factor calculations of an infinite array of PWR fuel with finite axial height. The aim being to study the effect of axial burn-up profile on criticality calculations of PWR fuel storage.	Report published in February 1996
II-B	PWR UOX	Multiplication factor and spatial fission distribution calculations of a realistic PWR spent fuel transport cask including accidental situations. The aim being to further study the effect of axial burn-up profile on criticality calculations.	Report published in May 1998
II-C	PWR UOX	Multiplication factor and fission distribution calculations of a realistic PWR fuel transport cask. The aim being to study the sensitivity to the axial burn-up shape.	Report published in September 2008
II-D	PWR UOX	Multiplication factor and fission distribution calculations of a realistic PWR fuel transport cask. The aim being to study control rods effects on spent fuel composition.	Report published in December 2006
II-E	PWR UOX	Study on the impact of changes in the isotopic inventory due to control rod insertions in PWR UO_2 fuel assemblies during irradiation on the end effect	Report published in June 2015
III-A	BWR UOX	Criticality calculations of an infinite array of BWR spent fuel assemblies with emphasis on axial burn-up and void profiles.	Report published in September 2000
III-B	BWR UOX	Depletion calculations of an array of BWR fuel.	Report published in February 2002
III-C	BWR UOX	Compositions of Spent Fuel Assembly for BUC and Criticality control of damaged fuel.	Report published in March 2016
IV-A	PWR MOX	Reactivity Prediction Calculations for Infinite Arrays of PWR MOX Fuel Pin Cells	Report published in May 2003
IV-B	PWR MOX	Inventory MOX Fuel Depletion Calculations	Report published in May 2003
VI	PWR UOX	Burn-up profile in a VVER-440 assembly	Analysis of results on-going
VII	PWR UOX	Study of spent fuel compositions for long-term disposal	Report published in February 2012
VIII	PWR	Reactivity worth of small isotopic samples (FP and actinides) in pile-oscillation experiments	Specifications Distributed

Table 6 Comparison of thermal absorption cross-sections.

Nuclide	JEFF-3.2	ENDF/B-VII.1	JEF-2.2	Δ(E7.1)	Δ(JEF-2.2)
^{95}Mo	12.0 b	11.9 b	12.4 b	-1%	+3%
^{99}Tc	18.7 b	17.8 b	17.1 b	**-5%**	**-9%**
^{101}Ru	3.03 b	4.64 b	3.04 b	**+53%**	+0.1%
^{103}Rh	130 b	129 b	133 b	-1%	+3%
^{109}Ag	80.9 b	80.5 b	80.9 b	-1%	<0.1%
^{133}Cs	25.8 b	25.8 b	25.9 b	+0.1%	+0.4%
^{143}Nd	297 b	287 b	297 b	-3%	<0.1%
^{145}Nd	37.1 b	37.3 b	37.2 b	+0.3%	+0.1%
^{147}Sm	50.8 b	50.6 b	50.8 b	-0.4%	+0.1%
^{149}Sm	63412 b	60107 b	61524 b	**-5%**	-3%
^{150}Sm	91.5 b	88.6 b	91.6 b	-3%	+0.1%
^{151}Sm	12501 b	12466 b	12510 b	-0.3%	<0.1%
^{152}Sm	183 b	183 b	183 b	<0.1%	<0.1%
^{151}Eu	7331 b	7710 b	7332 b	**+5%**	<0.1%
^{153}Eu	274 b	313 b	274 b	**+14%**	<0.1%
^{155}Gd	45504 b	45503 b	45535 b	<0.1%	+0.1%
^{234}U	88.7 b	88.7 b	90.9 b	=	+3%
^{235}U	594 b	594 b	591 b	=	-0.4%
^{236}U	4.76 b	4.61 b	4.63 b	-3%	-3%
^{238}U	2.39 b	2.39 b	2.42 b	<0.1%	+1%
^{237}Np	159 b	153 b	159 b	-4%	+0.1%
^{238}Pu	474 b	358 b	478 b	**-24%**	+1%
^{239}Pu	972 b	974 b	973 b	+0.2%	+0.1%
^{240}Pu	260 b	262 b	263 b	+1%	+1%
^{241}Pu	1271 b	1271 b	1274 b	=	+0.2%
^{242}Pu	16.8 b	19.0 b	16.6 b	**+13%**	-1%
^{241}Am	670 b	616 b	555 b	**-8%**	**-17%**
^{243}Am	69.1 b	72.4 b	67.9 b	**+5%**	-2%

Table 7 Comparison of absorption cross-section integrals.

Nuclide	JEFF-3.2	ENDF/B-VII.1	JEF-2.2	Δ(E7.1)	Δ(JEF-2.2)
^{95}Mo	111 b	104 b	110 b	**-6%**	-0.3%
^{99}Tc	312 b	322 b	304 b	+3%	-3%
^{101}Ru	111 b	112 b	111 b	+1%	-0.1%
^{103}Rh	968 b	968 b	1035 b	<0.1%	+7%
^{109}Ag	1473 b	1467 b	1474 b	-0.4%	<0.1%
^{133}Cs	421 b	405 b	439 b	-4%	+4%
^{143}Nd	132 b	130 b	132 b	-2%	<0.1%
^{145}Nd	231 b	223 b	231 b	-4%	<0.1%
^{147}Sm	794 b	777 b	794 b	-2%	+0.1%
^{149}Sm	3486 b	3432 b	3486 b	-2%	<0.1%
^{150}Sm	339 b	333 b	339 b	-2%	=
^{151}Sm	3464 b	3438 b	3465 b	-1%	<0.1%
^{152}Sm	2978 b	2977 b	2978 b	<0.1%	<0.1%
^{151}Eu	3351 b	3289 b	3352 b	-2%	<0.1%
^{153}Eu	1409 b	1419 b	1411 b	+1%	+0.2%
^{155}Gd	1544 b	1538 b	1545 b	-0.4%	+0.1%
^{234}U	632 b	632 b	660 b	=	+4%
^{235}U	409 b	409 b	404 b	<0.1%	-1%
^{236}U	349 b	346 b	351 b	-1%	+0.3%
^{238}U	275 b	275 b	278 b	+0%	+1%
^{237}Np	658 b	664 b	658 b	+1%	<0.1%
^{238}Pu	177 b	166 b	166 b	**-6%**	**-6%**
^{239}Pu	478 b	474 b	474 b	-1%	-1%
^{240}Pu	8487 b	8499 b	8452 b	+0.1%	-0.4%
^{241}Pu	741 b	740 b	740 b	-0.1%	-0.1%
^{242}Pu	1130 b	1123 b	1117 b	-1%	-1%
^{241}Am	1832 b	1594 b	1454 b	**-13%**	**-21%**
^{243}Am	1788 b	2053 b	1814 b	**+15%**	+1%

Table 8 Comparison of actinide thermal fission cross-sections.

Nuclide	JEFF-3.2	ENDF/B-VII.1	JEF-2.2	Δ(E7.1)	Δ(JEF-2.2)
^{234}U	0.059 b	0.059 b	0.407 b	=	**+591%**
^{235}U	507 b	507 b	505 b	=	-0.4%
^{236}U	0.054 b	0.042 b	0.042 b	**-23%**	**-24%**
^{238}U	0.000024 b	0.000015 b	0.000010 b	**-37%**	**-56%**
^{237}Np	0.016 b	0.018 b	0.016 b	**+12%**	+0.1%
^{238}Pu	15.2 b	14.8 b	14.8 b	-3%	-3%
^{239}Pu	697 b	699 b	698 b	+0.3%	+0.2%
^{240}Pu	0.054 b	0.058 b	0.061 b	**+7%**	**+13%**
^{241}Pu	940 b	940 b	946 b	=	+1%
^{242}Pu	0.0023 b	0.0122 b	0.0130 b	**+436%**	**+468%**
^{241}Am	2.91 b	2.83 b	2.93 b	-3%	+0.5%
^{243}Am	0.073 b	0.073 b	0.044 b	-1%	**-39%**

Table 9 Comparison of actinide fission cross-section integrals.

Nuclide	JEFF-3.2	ENDF/B-VII.1	JEF-2.2	Δ(E7.1)	Δ(JEF-2.2)
^{234}U	0.707 b	0.707 b	0.620 b	=	**-12%**
^{235}U	269 b	269 b	272 b	<0.1%	+1%
^{236}U	4.34 b	4.46 b	4.38 b	+3%	+1%
^{238}U	0.00125 b	0.00234 b	0.00216 b	**+88%**	**+74%**
^{237}Np	0.216 b	0.637 b	0.208 b	**+195%**	-4%
^{238}Pu	27.6 b	20.0 b	22.8 b	**-27%**	-17%
^{239}Pu	298 b	293 b	289 b	-2%	-3%
^{240}Pu	3.37 b	2.72 b	3.74 b	**-19%**	**+11%**
^{241}Pu	561 b	561 b	571 b	<0.1%	+2%
^{242}Pu	0.255 b	0.234 b	0.944 b	**-9%**	**+270%**
^{241}Am	7.87 b	8.43 b	9.78 b	**+7%**	**+24%**
^{243}Am	2.20 b	2.00 b	1.20 b	**-9%**	**-46%**

Table 10: Comparison of half-lives.

Nuclide[4]	JEF-2.2	δ[5]	JEFF-3.1.1	δ	ENDF/B-VII.1	δ
^{99}Tc	212,860 y	2.35%	214,000 y	3.74%	211,100 y	0.57%
^{147}Sm	$1.06{\times}10^{11}$ y	1.9%	=J2	=J2	=J2	=J2
^{151}Sm	88.7 y	10.7%	90 y	6.7%	90 y	8.9%
^{234}U	245,700 y	0.12%	=J2[6]	=J2	245,500 y	0.24%
^{235}U	704 My	0.07%	=J2	=J2	=J2	=J2
^{236}U	23.4 My	0.17%	23.7 My	0.84%	=J2	=J2
^{238}U	4,500 My	0.11%	=J2	0.07%	=J2	0.07%
^{237}Np	2.1 My	0.47%	=J2	=J2	=J2	0.33%
^{238}Pu	87.7 y	0.34%	=J2	=J2	=J2	0.11%
^{239}Pu	24,113 y	0.17%	=J2	0.05%	24,110 y	0.12%
^{240}Pu	6563 y	0.08%	=J2	=J2	6561 y	0.11%
^{241}Pu	14.4 y	0.69%	14.33 y	0.28%	14.29 y	0.04%
^{242}Pu	373,500 y	0.29%	=J2	=J2	=J2	=J2
^{241}Am	432.7 y	0.12%	432.8 y	0.16%	432.6 y	0.14%
^{243}Am	7365 y	0.30%	=J2	=J2	7370 y	0.20%

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[4] All the evaluations consider ^{95}Mo, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{143}Nd, ^{145}Nd, ^{150}Sm, ^{152}Sm, ^{151}Eu, ^{153}Eu and ^{155}Gd to be stable. The ENDF/B evaluations consider ^{149}Sm to be stable whilst the JEFF evaluations consider it to be extremely long lived (half-life = $2{\times}10^{15}$ y).

[5] δ is the percent uncertainty assigned to the evaluated half-life value

[6] The notation "=J2" indicates the value is the same as in the JEF-2.2 evaluation.

Table 11: Comparison of ^{235}U fission product yields.

	JEFF-3.1	ENDF/B-VII.1	JEF-2.2	Δ(E7.1)	Δ(JEF-2.2)
^{95}Mo	6.50E-02	6.50E-02	6.56E-02	<0.1%	+1%
^{99}Tc	6.13E-02	6.11E-02	6.18E-02	-0.4%	+1%
^{101}Ru	5.17E-02	5.17E-02	5.14E-02	+0.1%	-1%
^{103}Rh	3.10E-02	3.03E-02	3.02E-02	-2%	-3%
^{109}Ag	2.88E-04	3.12E-04	2.60E-04	**+9%**	**-10%**
^{133}Cs	6.60E-02	6.70E-02	6.61E-02	+2%	+0.3%
^{143}Nd	5.95E-02	5.96E-02	5.94E-02	<0.1%	-0.2%
^{145}Nd	3.94E-02	3.93E-02	3.93E-02	-0.3%	-0.3%
^{147}Sm	2.23E-02	2.25E-02	2.27E-02	+1%	+2%
^{149}Sm	1.05E-02	1.08E-02	1.05E-02	+3%	-0.5%
^{150}Sm	6.13E-07	3.00E-07	7.56E-07	**-51%**	**+23%**
^{151}Sm	4.20E-03	4.19E-03	4.16E-03	-0.4%	-1%
^{152}Sm	2.53E-03	2.67E-03	2.59E-03	**+6%**	+2%
^{151}Eu	4.20E-03	4.19E-03	4.16E-03	-0.4%	-1%
^{153}Eu	1.48E-03	1.58E-03	1.51E-03	**+7%**	+2%
^{155}Gd	3.08E-04	3.21E-04	3.24E-04	+4%	+5%

Table 12: Comparison of ^{239}Pu fission product yields.

|	JEFF-3.1	ENDF/B-VII.1	JEF-2.2	Δ(E7.1)	Δ(JEF-2.2)
^{95}Mo	4.95E-02	4.82E-02	4.92E-02	-3%	-1%
^{99}Tc	6.18E-02	6.21E-02	6.18E-02	+0.4%	-0.1%
^{101}Ru	6.18E-02	6.02E-02	6.19E-02	-3%	+0.2%
^{103}Rh	6.95E-02	6.99E-02	6.95E-02	+1%	+0.1%
^{109}Ag	1.67E-02	1.48E-02	1.50E-02	**-12%**	**-11%**
^{133}Cs	6.99E-02	7.02E-02	6.90E-02	+0.3%	-1%
^{143}Nd	4.48E-02	4.41E-02	4.47E-02	-1%	-0.2%
^{145}Nd	3.04E-02	2.99E-02	3.05E-02	-2%	+1%
^{147}Sm	2.04E-02	2.00E-02	2.05E-02	-2%	+0.4%
^{149}Sm	1.26E-02	1.22E-02	1.25E-02	-4%	-1%
^{150}Sm	2.27E-05	1.15E-05	1.64E-05	**-49%**	**-28%**
^{151}Sm	7.76E-03	7.38E-03	7.62E-03	-5%	-2%
^{152}Sm	6.08E-03	5.76E-03	5.86E-03	-5%	-4%
^{151}Eu	7.76E-03	7.38E-03	7.62E-03	-5%	-2%
^{153}Eu	3.80E-03	3.61E-03	3.95E-03	-5%	+4%
^{155}Gd	1.74E-03	1.66E-03	2.08E-03	-5%	**+19%**

Figures
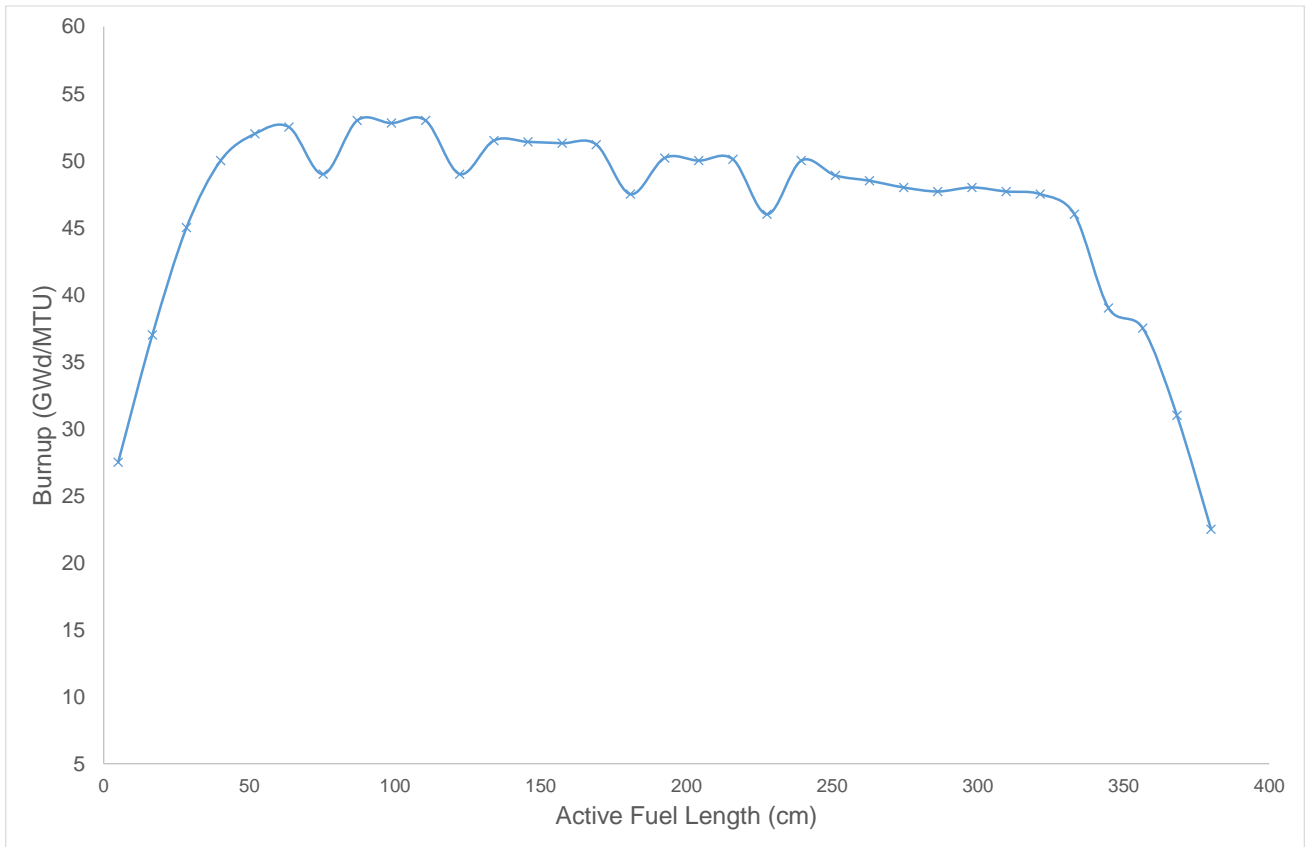
Figure 1 Sample Axial Burn-up Profile.

> Note: The 'dips' seen in the flat section of the axial profile are the location of the fuel assembly grid support plates. The suppression of the neutron flux caused by absorption of neutrons in these plates causes the reduced burn-up in the fuel rods.
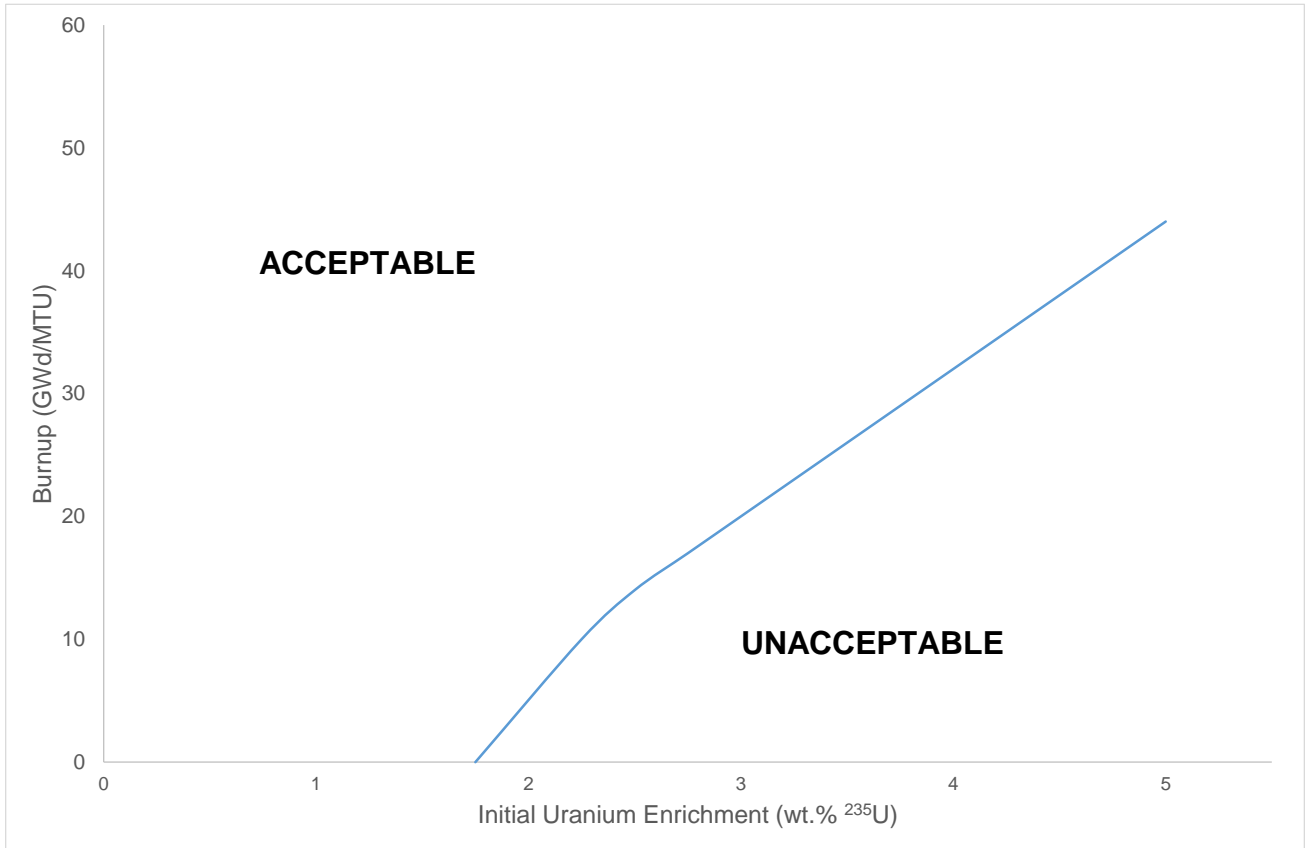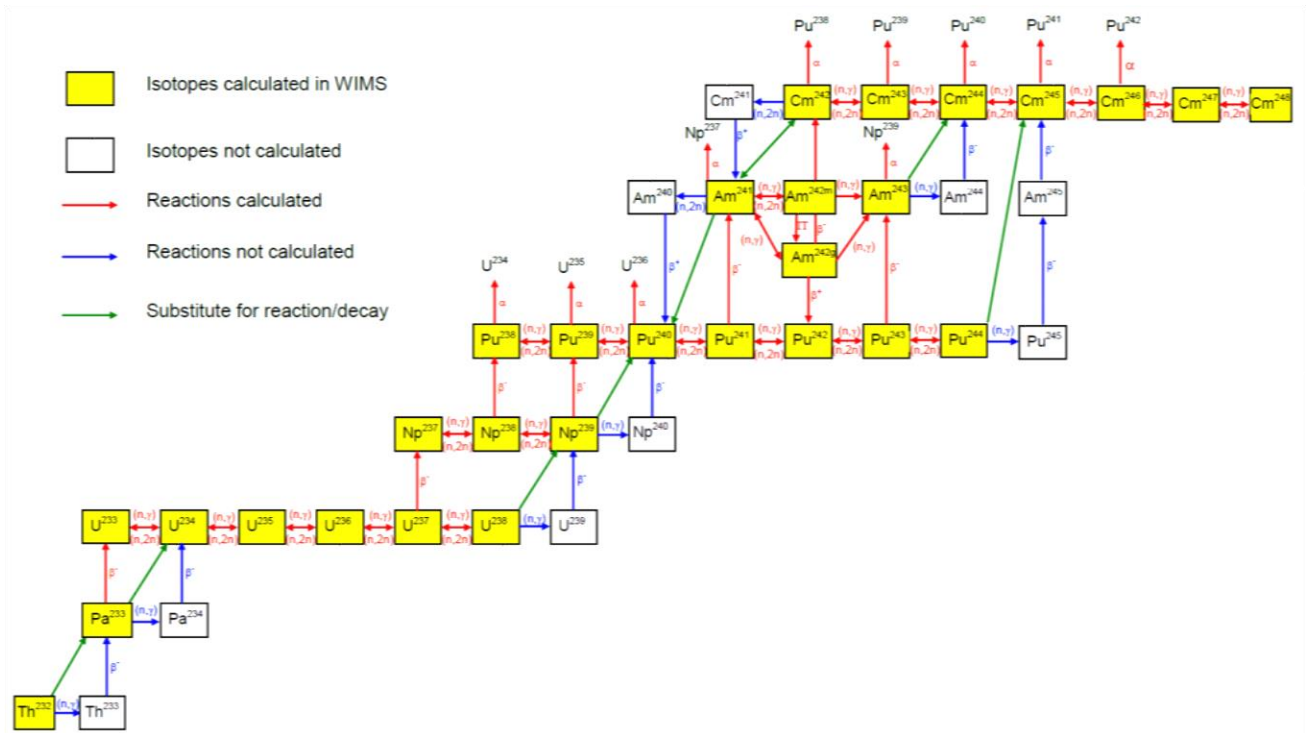
Figure 2 Sample Loading Curve.
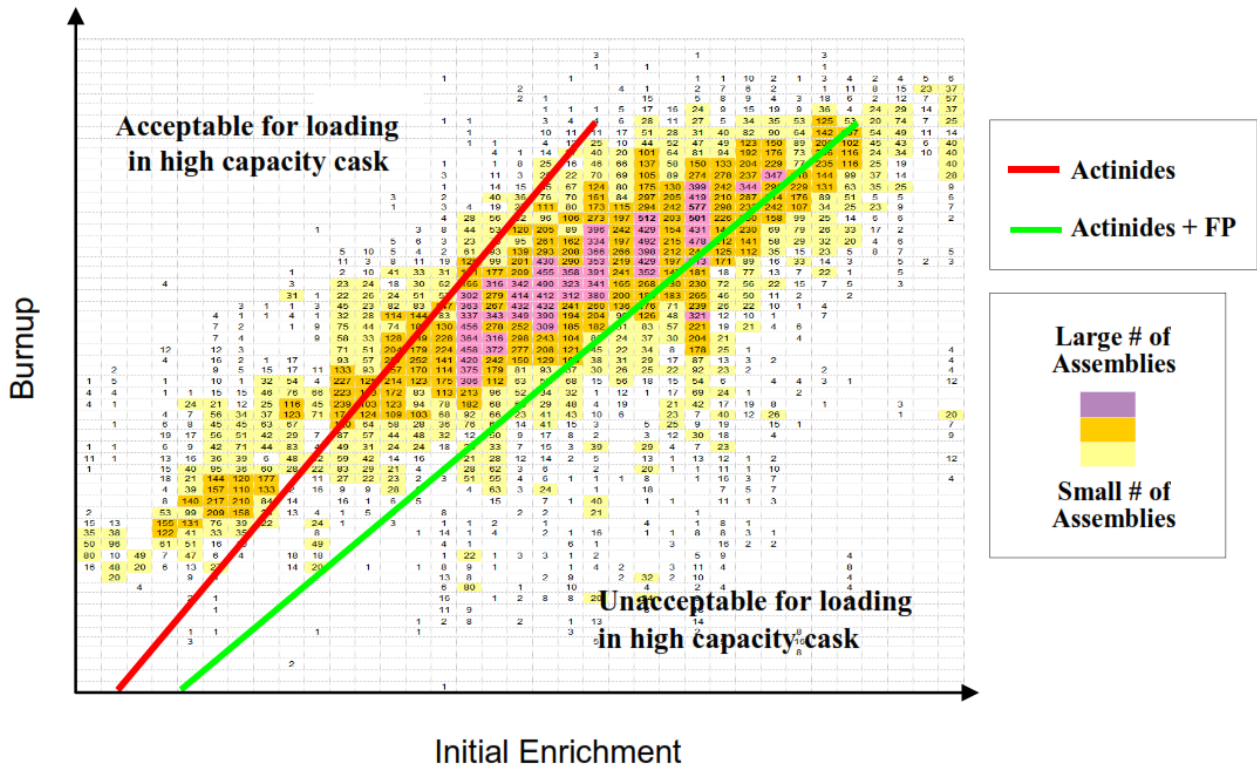
Figure 3 Further Sample Loading Curve.

Note this figure is a reproduction of Figure A-8 in Reference 3.

Figure 4 The WIMS inventory model used in MONK and WIMS depletion calculations.

This figure is reproduced from Reference 27.

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